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Revision 0

MARVEL 90% Final Design Report

September 2023

Idaho National Laboratory

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Microreactor Program
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EXECUTIVE SUMMARY

This report provides documentation of the Microreactor Applications Research Validation and Evaluation (MARVEL) project's 90% Final Design, as required by U.S. Department of Energy (DOE) Standard-1189, "*Integration of Safety into the Design Process.*"[1]

Per DOE-STD-1189-2016, the 90% Final Design documentation focuses on design completion, at a level capable of supporting procurement, construction, testing, and operation. At this phase, the design organization finalizes the hazards and accident analyses, fire hazard analysis (FHA), security vulnerability assessments, and other supporting analyses for design completion.

The objective of this report is to provide a high-level summary of the design thus far and provide references including, but not limited to, the following design deliverables:

- Complete final drawings, specifications and commercial grade dedications that may be released for bid and/or construction.
- Clearly defined testing plans for the safety and functionality of all subsystems.
- Quality Assurance Program for Design, Testing and Procurement.
- Software Quality Assurance Plan.
- Code of Record (COR), applicable design requirements including codes and standards.
- Final design that meets all the requirements stipulated in the COR.
- Final design review, consisting of final validation of comment resolution from previous reviews, and a review of any additional developments since the last review.
- Updated Safety Design Strategy.
- Hazard Analysis.
- Fire Hazard Analysis.
- Accident analysis.
- Security vulnerability assessment.
- Current and detailed cost estimate.
- Current construction schedule, and
- Risk & Opportunities Assessment.

The complete list of all engineering deliverables for the 90% final design are listed in Appendix D.

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TABLES

ACRONYMS

ADL	Affected Documents List, a substantial part of the EC package
ARDC	Advanced Reactor Design Criteria
ANS	American Nuclear Society
ANSI	American National Standards Institute
AR	augmented requirements
AS	Asset Suite
ASME	American Society of Mechanical Engineers
Be	beryllium
B_4C	boron carbide
BeO	beryllium oxide
BDBA	beyond deign basis accident
CD	control drum
CFD	computational fluid dynamics
CFR	*Code of Federal Regulations*
CIA	central insurance absorber
CM	configuration management
COR	code of record
CSS	Core Structures System
DBA	design basis accident
DC	design criteria
DCR	document change request
D&D	decontamination and decommissioning
DFM	Design for Manufacturability
DFS	Drum Forcing System
DHR	decay heat removal
DID	defense-in-depth
DNS	Drum Neutronics System
DOE	Department of Energy
DOE-ID	Department of Energy Idaho Operations Office
DPMS	Drum Position Measurement System
DSA	documented safety analysis
DSS	Drum Structures System
e	eutectic
EC	engineering change
ECS	Engine Cooling System
EDMS	Electronic Document Management System
EG	evaluation guideline
EIS	Environmental Impact Statement
EM	electromagnetic
FCS	Fuel Core System
FHA	fire hazards analysis
FONSI	Finding of No Significant Impact
FOR	Functional and Operating Requirements
FSAR	final safety analysis report
FS	fuel system
FSF	fundamental safety function
Ga	Gallium
GDC	general design criteria

GVS	Guard Vessel System
HALEU	high-assay low-enriched uranium
HC	hazard category
He	helium
HFP	hot full power
HMI	Human Machine Interface
HRU	Heat Rejection Unit
HX	heat exchanger
I&C	instrumentation and control
ICS	Instrumentation and Control System
IHX	intermediate heat exchanger
IGS	Inert Gas System
In	Indium
INL	Idaho National Laboratory
IRF	inherent reactivity feedback
kW	kilowatt
kWe	kilowatt-electric
kWth	kilowatt-thermal
LBE	lead-bismuth eutectic
LEU	low-enriched uranium
LMP	Licensing Modernization Project
LOCA	loss of coolant accident
LOF	loss of flow
LOHS	loss of heat sink
LOOP	loss of off-site power
LOP	loss of power
MARVEL	Microreactor Applications Research, Validation and Evaluation
MFC	Materials and Fuels Complex
MJ	megajoules
MR	management reserve
MRP	Microreactor Program
MRS	MARVEL Reactor Structure System
MSR	Molten Salt Reactor
MWth	megawatt thermal
NaK	sodium-potassium alloy
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NPH	natural phenomenon hazard
NRC	Nuclear Regulatory Commission
NSR	nonsafety-related
NSR-AR	nonsafety-related with augmented requirements
NUREG	Nuclear Regulatory Commission Regulation
PCB	PCS boundary
PCMS	Primary Coolant Management System
PCS	primary coolant system
PDC	principal design criteria
PEP	project execution plan
PGS	Power Generation System

PLM	Product Lifecycle Management
R&D	research and development
RCS	Reactivity Control System
RG	Regulatory Guide
RIA	reactivity insertion accident
RPP	Radiation Protection Program
RPS	reactor protection system
RSF	Reactor Support Frame
RSS	Reflector Support System
SAR	safety analysis report
SBE	safety basis event
SCB	secondary coolant system boundary
SCGS	Secondary Cover Gas System
SCMS	Secondary Coolant Management System
SCR	Stationary Core Reflector System
SCS	secondary coolant system
SDC	seismic design category
SDS	safety design strategy
SFR	sodium fast reactor
SFR-DC	sodium fast reactor design criteria
SHLD	Reactor Shielding System
Sn	Tin
SNAP	Systems for Nuclear, Auxiliary Power
SOS	Secondary Output Structure
SR	safety-related
SSC	structures, systems, and components
SSE	safe shutdown earthquake
SSS	Secondary Support Structure
T&FR	technical and functional requirements
TOP	transient overpower
T-REXC	TREAT Micro-Reactor Experiment Cell
TREAT	Transient Reactor Test (TREAT) facility
TRIGA	Training, Research, Isotope, General Atomics
TS	technical specifications
U	uranium
UCS	upper confinement structure
U-ZrH	uranium zirconium hydride
VM	verification matrix
WC	WindChill, a PLM program
ZrH	zirconium hydride

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MARVEL 90% Final Design Report

1. INTRODUCTION

The Microreactor Applications Research Validation and Evaluation (MARVEL) project, funded by the United States Department of Energy (DOE) via the Microreactor Program (MRP), represents an innovative milestone in the field of nuclear power generation. Aimed at developing a test microreactor at the Idaho National Laboratory (INL), the project sets out to redefine our understanding and utilization of nuclear energy systems. At the heart of the MARVEL project is the quest to create an operational nuclear applications test bed, purposed to generate combined heat and power. This infrastructure not only facilitates integration of Research & Development (R&D) with end-user technologies but also provides an enabling platform for microreactor technologists to test advanced control systems.

The microreactor, a thermal reactor employing Uranium Zirconium Hydride (UZrH) fuel, is currently under review for authorization by the Department of Energy Idaho Operations Office (DOE-ID). This encompasses compliance with the National Environmental Policy Act (NEPA), safety assessments, and supplementary readiness evaluations for startup and operation. The MARVEL reactor is strategically planned to reside in the Transient Reactor Test (TREAT) Facility to expedite deployment. Leveraging the existing operating Category B reactor facility, approved facility safety basis, operating crews, and recent re-start experience, MARVEL is designed for rapid implementation.

The MARVEL team is composed of multidisciplinary experts from Idaho National Laboratory (INL), Argonne National Laboratory, Los Alamos National Laboratory, Walsh Engineering, Qnergy, Munro & Associates, and Creative Engineers Inc. The diverse team brings a wealth of knowledge and innovation to the project, striving to shape the future of microreactor technology.

Investigating a near-term concept is central to the MARVEL project. Its aim is to probe the feasibility and utility of the microreactor as a future research and development test bed. Key pursuits include the integration of microreactors with electric and non-electric applications and the demonstration of the technology to essential end-users. Additionally, the project may stimulate infrastructure development and capabilities to support future microreactor demonstrations.

The U.S. DOE MRP is at the forefront of technological advancement in the nuclear energy sector. Its focus on R&D is designed to address the existing economic and energy security challenges faced by decentralized generation in civilian, industrial, and defense energy sectors. The program verifies that microreactor concepts can be licensed and deployed by commercial entities to meet specific use case requirements. These very small, factory-fabricated, transportable reactors are designed to produce tens of megawatts of thermal (MWth) energy. Microreactors, by virtue of their self-regulating characteristics, low maintenance requirements, and sustainable and affordable heat and power generation, present an effective solution to remote communities and industrial users. Their unique and novel nature underpins the objectives of the MARVEL project as we aim to accelerate the research, development, and application of these next-generation nuclear reactors.

1.1 Design Goals

The driving tenet of the design philosophy central to the MARVEL project is expeditiously achieving criticality. Contrary to a conventional focus on technology readiness level, the key performance metric adopted within this project is time. This metric, however, does not overlook the imperatives of safety and quality. Rather, it pushes the boundaries of project progression without jeopardizing these fundamental considerations.

Three principal factors inform the team's technology selection process: (i) the availability of off-the-shelf technologies, (ii) the feasibility of technology implementation given existing resources and expertise, and (iii) the potential for swift development to elevate technology and manufacturing readiness levels. The resultant operational ethos is characterized by agility, proactive problem-solving, and a solution-oriented mindset. Innovation and creativity undergird the project's team culture, propelling the exploration of new methodologies and tools outside the conventional realm of nuclear technology. Specifically, the Scrum Agile Product Development framework has been integrated into the project's management structure. Its iterative and incremental nature aligns with the project's philosophy of marrying rapid progression with safety and quality imperatives.

Moreover, the Design for Manufacturability (DFM) modeling technique, a mainstay in the automotive industry, has been adopted within the project's design processes. The DFM method prioritizes product designs amenable to simplified manufacturing, thereby facilitating reductions in production costs and time.

In essence, the design philosophy of the MARVEL project encapsulates a delicate balance between speed, safety, and quality, all while nurturing an environment conducive to innovation and creativity. The strategic adoption of tools and methodologies from a variety of industries catalyzes project progression, edging the promise of microreactor implementation closer to fruition.

1.1.1 Target Application

The MARVEL project stands at the nexus of research, technological development, and practical application, aimed in particular at serving (i) researchers, (ii) technology developers, and (iii) potential microreactor end-users. With its passive design and strategic approach, MARVEL seeks to enable microreactor deployment by developing a dynamic, applications focused, nuclear test bed. This infrastructure is tailor-made to facilitate advanced research, promote hands-on education, train operators, and support the rigorous testing of state-of-the-art control systems. By actively bridging the gap between theoretical research and practical utility, MARVEL endeavors to simplify and expedite the deployment process for microreactors. In doing so, the project not only paves the way for cutting-edge advancements in nuclear energy but also stimulates a more widespread adoption and understanding of microreactor technologies, including seamless integration with both electric and non-electric applications from a microreactor operating within a microgrid. Through this holistic approach, MARVEL positions itself as a premier resource for researchers and technologists aiming to be at the forefront of next-generation nuclear innovations.

1.1.2 Installation, Startup, and Operation

The MARVEL project is steadfastly committed to refining the processes integral to the installation, startup, and seamless operation of microreactors. One of its paramount goals is to streamline the complexities traditionally associated with siting and environmental reviews, thereby accelerating the deployment of these next-generation reactors. MARVEL is the first reactor to achieve a NEPA approval based on a shorter Environmental Assessment with a finding of no significant impact (FONSI) and not an Environmental Impact Statement (EIS).

Understanding the vulnerabilities of such first-of-a-kind technologies, MARVEL underscores the importance of leveraging both cyber and physical security frameworks of the existing hazard category 2 facility of the TREAT reactor. These existing enhanced security measures are meticulously devised to guard against potential threats and ensure the uninterrupted and safe operation of the test microreactor.

Equally significant is MARVEL's ambition to ensure that these microreactors can be flawlessly integrated into a net-zero electrical and/or thermal microgrid, underpinning its commitment to sustainable energy solutions. One of the project's high-value objectives revolves around the demonstration of both high and low-grade heat extraction. This focus is instrumental in driving the twin goals of optimized efficiency and maximizing the productivity potential of the microreactors. Collectively, these endeavors chart a comprehensive roadmap for the microreactor's lifecycle, from its installation and startup to its sustainable operation. It is important to note that the high-grade heat extraction system will be pursued separately from this design effort and thus will not be included in this final design report.

1.1.3 Enabling Future Control Technologies

One of the pivotal components of this venture is the emphasis on introducing radiation and temperature-resistant sensors and instrumentation. The robustness of these tools is of paramount importance, given their role in safe and reliable reactor operation. Furthermore, they serve a dual purpose by also facilitating advanced tests for sensor reliability and ensuring their online calibration remains precise.

In the future, the horizon of nuclear reactor operation will be redefined by the MARVEL project through its capability of testing advanced control systems. The ambition here is twofold: firstly, to significantly minimize human intervention by automating operator functions, and secondly, to ensure that this automation doesn't encroach upon the sanctity of reactor safety. While these ambitions are not the primary objectives of the MARVEL project today, the existence of MARVEL reactor can enable these demonstrations with an operational microreactor. Potential future enabling development and demonstration opportunities include, but not limited to, the following:

1. Digital Twin: An innovative application of the live data procured from these sensors is in the creation of a digital twin for the reactor. This virtual replica is not just a static model; it is envisioned to be the training ground for an advanced, artificial intelligence-driven control system. The potential here is vast: enhanced operational precision, improved efficiency, and the capability to predict and respond to potential disruptions even before they manifest.
2. Remote Monitoring: Additionally, the incorporation of wireless data transmission technologies means that crucial information on the microreactor's electrical and thermal power output can be constantly relayed in real-time throughout its entire lifecycle – from startup to shut down. This continuous stream of data serves as a foundation for predictive maintenance strategies, ensuring not just optimal performance but also significantly prolonging the reactor's operational lifespan.
3. Autonomous Operation: And finally, embarking on the journey to achieve autonomous operation in microreactors represents a visionary pursuit in nuclear research and development. The endeavor requires an amalgamation of nuclear science, cutting-edge sensor technology, and advanced AI algorithms. The R&D landscape of the future envisions a microreactor that can independently manage its functions, adapt to varying external conditions, and preemptively address potential disruptions. The path to this autonomy demands a symbiotic relationship between real-time data analytics and predictive modeling. As nuclear technology interfaces with the realms of artificial intelligence and machine learning, the objective is to create reactors that are not only self-sufficient but also embody the pinnacle of safety and efficiency. This revolutionary R&D focus underscores a commitment to reimagining the boundaries of nuclear energy while fostering an environment of innovation and forward-thinking solutions.

1.1.4 Scope of 90% Final Design of the MARVEL Project

Inclusion: The scope of the 90% final design includes the complete reactor, including the design, operability and maintainability of the five major reactor systems: (i) Fuel and Core System, (ii) Reactivity Control System (iii) MARVEL Reactor Structure, (iv) Instrumentation & Control System, and (v) Power Generation System. The scope of this design also includes the primary and secondary coolant loading system.

Exclusion: The current scope does not include (i) the high-grade heat extraction system, and (ii) interfacing with the TREAT facility, which will be conducted through the TREAT Micro-Reactor Experiment Cell (T-REXC) project, whose concept is briefly described in this report. The objective of the T-REXC project includes transforming the North storage pit into a critical experiment facility that can be used by MARVEL and other critical experiments in the future within the TREAT facility. The structures, systems, and components (SSCs) for T-REXC includes:

- North storage pit shield structures (to prevent neutron activation of the concrete)
- Pit lid, with integrated top shielding
- I&C infrastructure (to capture and display T-REXC facility data and demonstrator data)
- Electrical power infrastructure – including electrical supply panel near the pit, standby power generator, etc.
- Control room infrastructure - including signal and data transfer between MFC-720 and MFC-724
- Ventilation, including HEPA filter and exhaust monitoring,
- Fire detection, including sodium and sodium-potassium alloy (NaK) fires
- Fire mitigation systems, in accordance with the TREAT fire hazards analysis
- Neutron source for startup
- Radial static neutron reflectors
- Reactivity control materials, in the form of beryllium oxide (BeO) control drums for neutron population control
- A system to preclude water intrusion into the pit
- Radiation monitoring.

1.2 Risks and Schedule

PLN-6384, "Microreactor Applications Research Validation and Evaluation (MARVEL) Project, Project Execution Plan,"[2] has been prepared for the project. The plan is a living document that defines the scope, responsibilities, and methodology for identifying, assessing the impacts of, and managing risks that could affect successful and timely completion of the project.

Risk management processes are used to predict the uncertainties in the project and minimize the occurrence or impact of those uncertainties that are threats, such that there are minimal and acceptable impacts on the project's cost, schedule, and operational performance. Conversely, opportunities that represent a good chance of significantly improving the project's cost, schedule, or operating performance may be exploited. (Where threats and opportunities are considered collectively, the term "risk" will be used. Otherwise, they will be referred to individually as threats or opportunities.) Rather than develop a stand-alone risk management plan, the risk management processes for MARVEL will be managed as described in PLN-6384.

Risk management is a necessary and ongoing management process. The MARVEL project will consider all potential sources of project-related threats and opportunities, such as programmatic and technical risks, risks to the facility performance, and risks from interfaces with other programs. The risk management process described here will be conducted to closure of the risk or close-out of the project. The objectives of this section are to:

- Evaluate the project scope and assumptions in the development of risk management responsibilities.
- Describe the risk assessment process to identify, qualify, quantify, respond to, and track project risks.

- Ensure the detail, scope, timing, and risk analysis are commensurate with the complexity of the project.
- Identify when, during the project lifecycle, the risk assessment (identification, qualification, quantification, and response) is performed and updated.

The outcomes of the risk processes are:

- Recommended levels of Management Reserve and Contingency for the MARVEL project.
- Recommended levels of schedule reserve for the MARVEL project.
- A prioritized risk register that identifies discrete risk events, risk-handling strategies, risk-handling actions, as well as pre-and post-mitigation risk likelihoods and impacts.

Risks have been identified, evaluated qualitatively, and described in the risk register. DOE STD-1189 guidance for developing a risk and opportunity matrix was used in the development of the risk register, and it has been updated as the project design matured. The current Risk Register is shown in Table 1. The project risk register will be reviewed as needed, but at least quarterly, to reevaluate and update risks and their handling strategies. Emerging risks will be added to the register and evaluated for consequence, impact, and planned mitigation strategies. Detailed review of risk handling results in the design phase were reviewed at the project's 90% design review. Similarly, the 90% design review will include forward-looking risk identification and handling strategies for construction. The risk register will be maintained as a living document through the rest of the project.

The risk register identifies the potential cost impact for the purpose of assessing (i.e., scoring and ranking) the risks as low, medium, or high. The residual risk cost and schedule impacts in the risk register serve as the basis for recommending contribution to management reserve (MR), schedule reserve, DOE-held contingency, and DOE schedule contingency. The project risk mitigation strategies focus on identifying actions to eliminate or mitigate the likely and high impact threats identified in the risk register. Exploitation of opportunities will focus similarly on those opportunities with high potential for exceptional impact. The probability and impact chart used in the MARVEL risk register is shown below.

Quantitative risk analysis determines the probability and consequence of overall project risk by applying a numerical model to the individual risks identified in the risk register. A Monte Carlo analysis tool, Safran, may be used that is capable of statistically evaluating individual risks and opportunities as they relate to cost and schedule. In this tool, the threats and opportunities will be tied to individual activities or items in the resource loaded schedule and cost estimate to be analyzed. The threats and opportunities will be correlated in the model to define overall project risk and refine MR, schedule reserve, and contingency. The quantitative risk model and analysis will be documented, maintained, and updated throughout the life of the project.

Table 1. MARVEL risk register.

Risk ID	Risk Title	Risk Statement	Risk Assumptions	Type	Risk Category	Initial Risk Rating	Handling Strategy	Residual Risk Rating
T1	T1- unexpected thermal expansion around drum	If unexpected thermal expansion around drum occurs, then CD cannot be turned as designed or reduced startup rate.	The gaps and tolerances between moving and non-moving parts are insufficient to accommodate thermal expansion.	Threat	Technica l	Moderate	Mitigate	Moderate
T4	T4- Natural circulation in the Primary Coolant Loops	If there are no pumps in the system and project is relying on natural circulation and it does not offer adequate coolant mass flow rates, then project does not meet its objective to deploy a functional microreactor.	Assumes natural circulation will not occur as modeled.	Threat	Technica l	High	Mitigate Accept	Low
T6	T6- Delayed Fuel Fabrication	If there are delays in fuel fabrication at TRIGA International (late start, low production rate), then there will be delays in delivery of the fuel to TREAT and ultimately delay MARVEL start-up.	Fuel cannot be fabricated at TRIGA International timely.	Threat	Project	High	Share	Low
T7	T7- Supply Chain Delays	If there are delays in the supply chain, then there may be delays of receipt of components such as the primary coolant system or beryllium metal (non-fuel).	Long lead procurement increases the threat that technical errors can be made in the procurement. Approval of long lead procurement may be delayed.	Threat	Project	Moderate	Mitigate	Low
T8	T8- Procured Components Fail Acceptance Test (Non-Fuel components)	If acceptance testing of MARVEL components (non-fuel) fails, then rework or re-order is required.	Early inspection are not incorporated in individual procurements.	Threat	Project	Low	Mitigate	Low

Table 1. MARVEL risk register.

Risk ID	Risk Title	Risk Statement	Risk Assumptions	Type	Risk Category	Initial Risk Rating	Handling Strategy	Residual Risk Rating
T9	T9- Start-up Delays	If there are equipment issues or unexplained data at start-up, then there will be a delayed approach to criticality and/or physics testing.	MARVEL does not perform physics testing per the test plan.	Threat	Technical	Moderate	Mitigate Accept	Moderate
T10	T10- Component Constructability	If as designed, some components may not be able to be constructed, then it will result in redesign, rework and cost increases/schedule delays.	Many of the components in the MARVEL primary system are new designs and unique.	Threat	Technical	Low	Mitigate	Low
T12	T12- TREAT Facility not ready for MARVEL Construction	If there is inadequate integration with the TREAT facility and organization, then schedule delay will result due to the facility and/or staff not ready to support MARVEL construction, start-up, and/or operation.	MARVEL does not integrate with TREAT to ensure coordination with TREX-c.	Threat	Project	Moderate	Mitigate	Low
T14	T14- Limited Resource Pool	If there is a limited resource pool from which to draw scientists, engineers, and operations resources to perform MARVEL scope, then schedule delays will result.	MARVEL is limited to use INL resources or issue minimal subcontracts for comparable resources.	Threat	Project	Moderate	Transfer	Low
T16	T16- Programmatic Conflicts	If there are programmatic conflicts at TREAT for space and resources (PELE, SAR Upgrades, Net Zero, etc.), schedule delays will result.	MARVEL is not prioritized at TREAT.	Threat	Project	Moderate	Mitigate	Low

Table 1. MARVEL risk register.

Risk ID	Risk Title	Risk Statement	Risk Assumptions	Type	Risk Category	Initial Risk Rating	Handling Strategy	Residual Risk Rating
T17	T17- Funding Delays	If there are delays in funding due to continuing resolution or delayed allotments, then delays in performance may occur. (Contracting is particularly susceptible.)	Limitation of funding clauses cannot be used in all contracts.	Threat	Project	Moderate	Mitigate	Low
T22	T22- MARVEL Operation may require 2 shifts	If MARVEL operations is increased to two shifts, then TREAT may not be prepared to support the increased resource need and delays to operations may be realized.	Two shifts will be required to operate MARVEL successfully.	Threat	Project	High	Mitigate	Moderate
T24	T24- Bowing/twisting of the fuel reduces heat transfer	If bowing occurs, then azimuthal temperature of the fuel cladding would exceed the cladding temperature criteria resulting in the need for wire spacers to be used in between the fuel rods.	Heat transfer will be reduced.	Threat	Technica l	Moderate	Mitigate	Low
T27	T27- Adequate design of engine tube blow-out mitigation	If the design for the engine tube blow-out is inadequately designed, then rework of mitigation tactics will be required.	design assumptions are incorrect/ mitigation tactics fail; results do not meet criteria impacting primary pressure vessel and schedule(s)	Threat	Safety	High	Mitigate	Moderate

Table 1. MARVEL risk register.

Risk ID	Risk Title	Risk Statement	Risk Assumptions	Type	Risk Category	Initial Risk Rating	Handling Strategy	Residual Risk Rating
T28	T28- verification of corrosion induced failure	If corrosion rates are significantly different than acceptable/planned rates and effects on the sacrificial liner would exceed allowable amounts, then MARVEL operation run-times will be shortened and the IHX (primary boundary) would be compromised.	maintenance times will be increased, run-time shortened to impractical levels, & a leak in the primary boundary is found	Threat	Safety	Moderate	Mitigate	Low
O1	O1- Collaboration with Research Reactor Infrastructure (RRI) Program	If MARVEL can collaborate with RRI on HALEU material purchase, transportation, and TN-BGC-1 licensing, then economies of scale and opportunities to leverage RRI experience can be offered.	TRIGA International will re-license the TN-BGC-1 container for TRIGA and MARVEL fuel. HALEU will be procured from Y12.	Opportunity	Project	Moderate	Exploit Enhance	High
O2	O2- Additional demonstrations during Operations increase MARVEL Utility	If opportunities arise to work with other programs for demonstration of MARVEL, then collaborations will be developed using existing infrastructure.	Other programs are aware of MARVEL capabilities and are part of the collaboration meetings.	Opportunity	Project	High	Exploit	High

2. MARVEL REQUIREMENTS

2.1 Requirements Structure

Functional and Operating Requirements (FOR)-868, "Microreactor Applications Research Validation and Evaluation (MARVEL) Project,"[3] contains the Level 1 and Level 2 requirements for the MARVEL project. Level 1 requirements are the highest-level mission requirements that apply to all SSCs in the MARVEL design and are derived from unique project objectives. Level 2 requirements are either derived from regulatory sources (e.g., the MARVEL principal design criteria or DOE Orders & Regulations) or refine/decompose Level 1 requirements (and are linked to them in the project requirements management database) in terms of how they specifically apply to the five primary systems in the MARVEL design:

- Fuel and Core System (FCS)
- MARVEL Reactor Structure (MRS)
- Reactivity Control System (RCS)
- Instrumentation and Control System (ICS)
- Power Generation System (PGS).

The relationship between these systems is shown in Figure 1. Grey systems are outside the scope of the MARVEL project. Red arrows represent thermal interfaces, orange arrows represent nuclear interfaces, yellow arrows represent electrical interfaces, teal arrows represent instrumentation and control (I&C) interfaces, and green arrows represent gaseous interfaces.

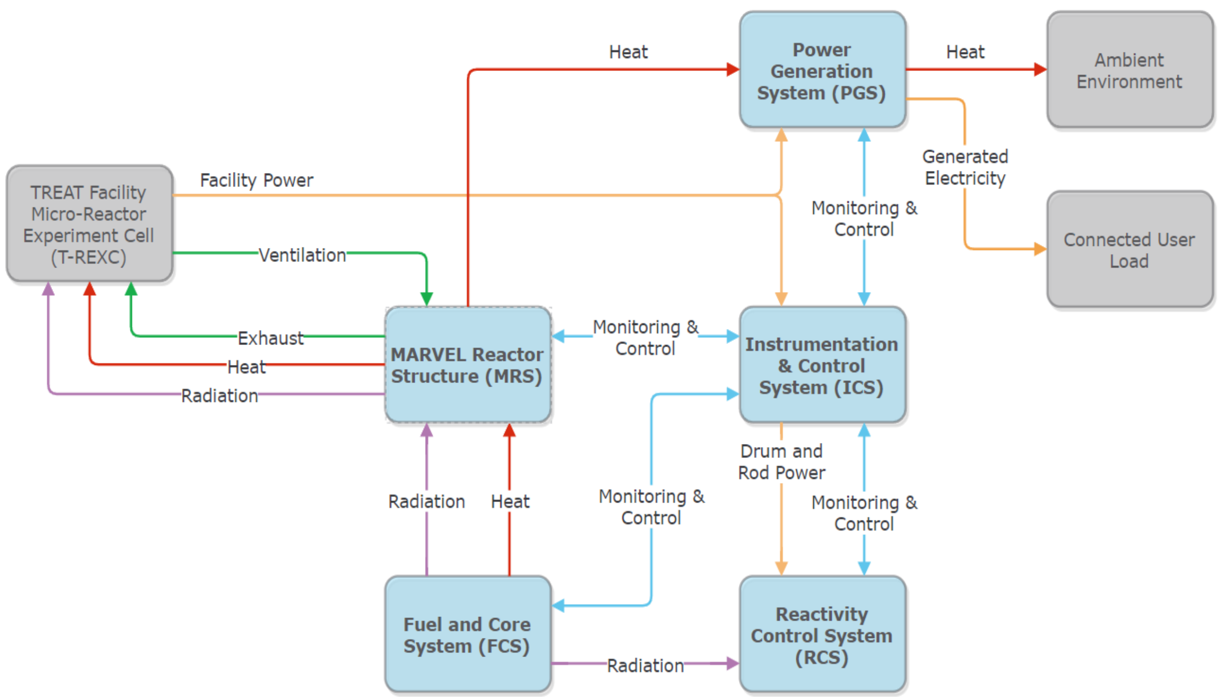

Figure 1. MARVEL systems and interfaces (from FOR-868).

The Level 1 and Level 2 requirements are further refined into Level 3 requirements found in the technical and functional requirements (T&FR) documents for the project.

- TFR-2574, "MARVEL Instrumentation and Control System (ICS)"[4]
- TFR-2575, "MARVEL Power Generation System (PGS)"[5]
- TFR-2576, "MARVEL Reactor Structure (MRS)"[6]
- TFR-2577, "MARVEL Fuel and Core System (FCS)"[7]
- TFR-2578, "MARVEL Reactivity Control System (RCS)"[8]

FOR-868, Appendix B, provides the requirements management philosophy for the project. FOR-868, Appendix C, contains a mapping of how all MARVEL principal design criteria and DOE Order 420.1C design requirements are implemented across the T-REXC and MARVEL projects. The reactor will be built inside the T-REXC, a separate project governed by the requirements of FOR-684, "Transient Reactor Test (TREAT) Facility Micro-Reactor Experiment Cell (T-REXC)."[9]

2.2 Requirements Traceability

Requirements traceability schema is presented in Figure 2. All Level 1 requirements are traced to Level 2 requirements by the "derives/derived by" connection. Level 3 requirements and are traced to both Level 2 requirements (derived by) and to the respective assets/functions (satisfied by). If a requirement is traced to a function, this function in turn is traced to an asset via "performed by/performs" connection. Similarly, validation procedures are also traced to the respective requirements via "validated by/validates" connection.

The requirements management tool used for this project - DOORS provides a single source of truth for all design groups. It enables online collaboration when working of the requirements and baselining and historian capabilities. When needed it automatically produces snapshots of the current state of the requirements documents in Word or PDF formats. System specific requirements (Level 2 and 3) are stored in DOORS as segregated documents, where each requirement presents a single item that can be traced independently of other items. Additionally, each requirement tracks with it attributes describing how it is verified in the final design. These attributes include the verification method (analysis, review, or qualification testing), the verifying person, and a link to the objective evidence that the requirement has been met (via ECAR, drawing, etc.). These verification artifacts are rendered in tabular form in the MARVEL Verification Matrix (VM)-118, "MARVEL Verification Matrix."[10]

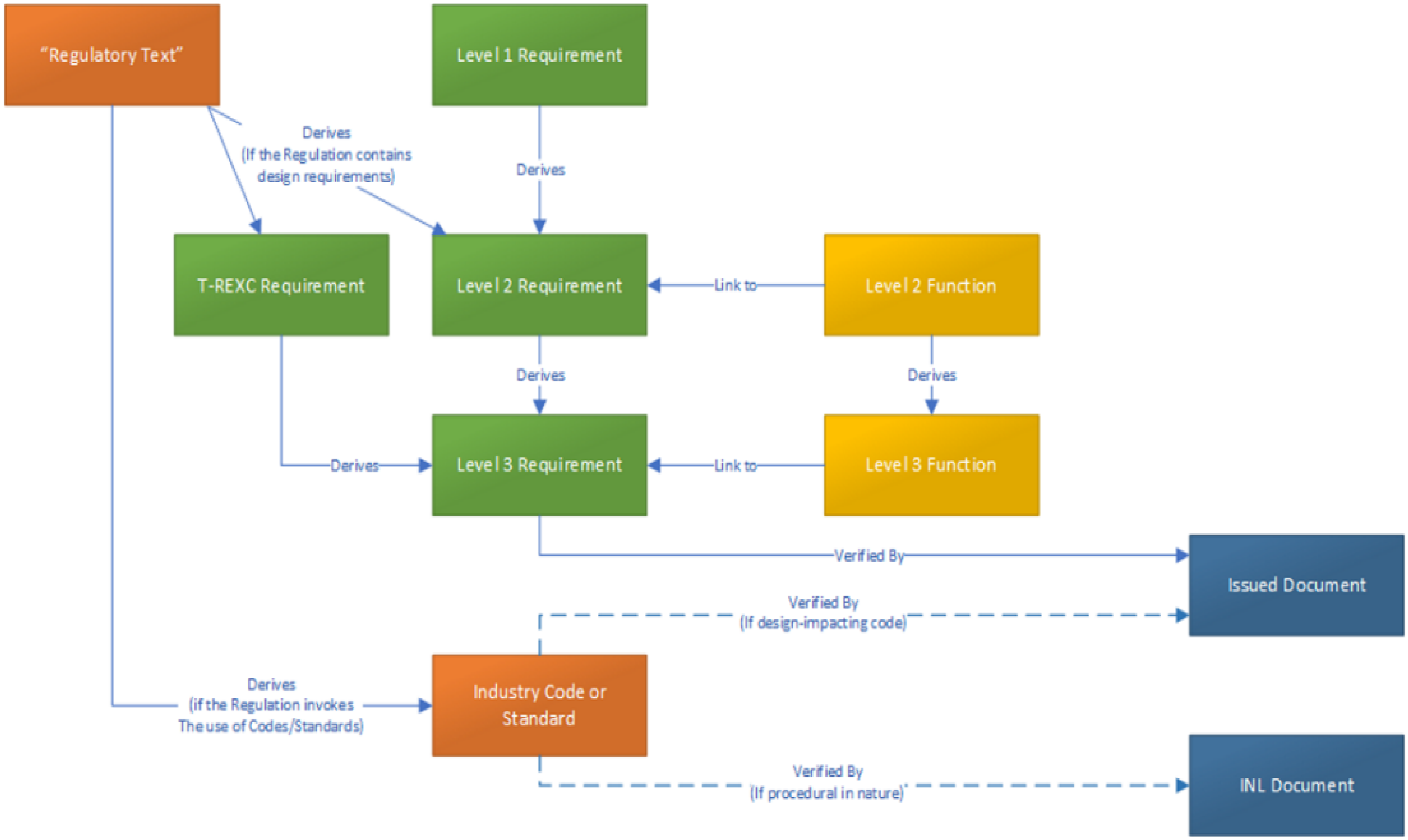

Figure 2. MARVEL requirement's structure.

2.3 Compliance Documents

Code of Record (COR)-0011, "MARVEL Project Code of Record,"[11] identifies the codes, standards, and procedures necessary to design, develop, construct, and startup of the MARVEL project. The documents listed in COR-0011 shall be considered requirements for compliance, unless or until otherwise determined. Any conflicts between this document and the compliance documents should be brought to the attention of the MARVEL design lead for a possible specification change.

2.4 Configuration Management

Configuration management (CM) is a systems engineering process for establishing and maintaining consistency of a product's performance, functional, and physical attributes with its requirements, design, and operational information throughout its life. CM at INL is performed by the engineering change (EC) process in AssetSuite (AS) as described in LWP-10500, "Managing the Configuration of Structures, Systems, and Components,"[12] and PDD-10502, "INL Configuration Management Program."[13]

2.4.1 Digital Ecosystem

The project requirements and associated verification matrices are developed and managed within the INL instance of the IBM DOORS software. Once the requirements are reviewed and agreed upon, they are locked in DOORS to form the design baseline. Any subsequent changes will be managed using the DOORs change-sets feature to create a new baseline. New baseline requirements and verification matrices are exported to PDF files using the appropriate templates, whereupon they are entered into the Electronic Document Management System (EDMS) system via the document change review (DCR) process. The DCR review process ensures that changes do not result in conflicts with other systems, procedures, or processes and that they meet all applicable quality standards.

2.4.2 As-designed Verification

All project drawings and documents will be verified against the criteria established for the requirements in VM-118 prior to or during the DCR review process. Records of this verification will be maintained in EDMS.

2.4.3 Final Verification

Final verification test criteria (i.e. verification of as-built system) and procedures are maintained by DOORS and linked to the requirements by the relations in Figure 2. The test object in DOORS contains procedure, duration, desired outcome, observed outcome, and pass/fail checkbox. DOORS will provide a coverage report summarizing final test results. Validation tests to confirm functionality of the integrated, as-built system are the purview of the commissioning and start-up program.

2.5 Principal Design Criteria for the MARVEL Reactor

DOE O 420.1C, "Facility Safety,"[14] requires that for any new Department of Energy (DOE) nuclear reactor, a set of reactor-specific safety design criteria (DC) must be established. As stated in Section 3.1 of safety analysis report (SAR)-420, "Transient Reactor Test Facility (TREAT) Final Safety Analysis Report,"[15] the Transient Reactor Test (TREAT) facility general design criteria (GDC) were developed by comparison to 10 *Code of Federal Regulations* (CFR) 50 Appendix A, "General Design Criteria for Nuclear Power Plants."[16]

However, significant work has recently been performed in the development of Advanced Reactor Design Criteria (ARDC), specifically for advanced reactor designs, in Nuclear Regulatory Commission (NRC) regulatory guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors."[17]

Although MARVEL is not a fast reactor, since the MARVEL reactor is a NaK cooled microreactor, the sodium fast reactor (SFR)-DC in RG 1.232 Appendix B are judged more applicable to MARVEL than the RG 1.232 Appendix A ARDC, or the 10 CFR 50 Appendix A GDC used in the existing TREAT safety design bases. Therefore, the SFR-DC in RG 1.232 Appendix B are selected for developing the principal design criteria (PDC) for the MARVEL reactor. Although DOE Order 5480.30 "Nuclear Reactor Safety Design Criteria,"[18] has been cancelled, the order also provides additional guidance and justification for a graded approach for MARVEL PDCs.

The design and operating requirements for DOE microreactors such as MARVEL are not necessarily the same as those that apply to NRC licensed SFRs. The process for developing the MARVEL PDC is shown in Figure 3. Of the 64 RG 1.232 SFR-DC, 19 are not applicable because of the special nature of the MARVEL design. Of the remaining 45 SFR-DC, 8 were revised to reflect that NaK is the primary coolant, and 3 were rewritten as new PDC to implement the following fundamental safety functions (FSFs) discussed in ECAR-6440, "MARVEL Hazard Evaluation."[19]

- Criterion 16: Confinement of Radioactive or Hazardous Material
- Criterion 26: Reactivity Control
- Criterion 34: Core Flow/Heat Removal.

Appendix A summarizes the SFR-DC that have been applied to the MARVEL reactor. Appendix A provides a statement regarding which SFR-DC were 1) adopted as a PDC for MARVEL with or without revision, 2) rewritten with a MARVEL design specific PDC, and 3) not adopted since not applicable to the MARVEL design. Appendix A also summarizes the applicable DOE requirements for each MARVEL PDC, and how the PDC in are implemented in the MARVEL design.

Figure 4 shows how the PDC are translated into design requirements in the FOR and TFR documents, which are then verified in VM-118. The PDCs are listed in FOR-868 and traced in a matrix to the Level 2 requirements which implement them for the specific MARVEL design. These are decomposed and mapped to Level 3 requirements that are listed in the TFR documents. Finally, the TFRs are mapped to design outputs that verify the Level 3 requirements (and thereby the PDCs) have been satisfied, as documented in VM-118.

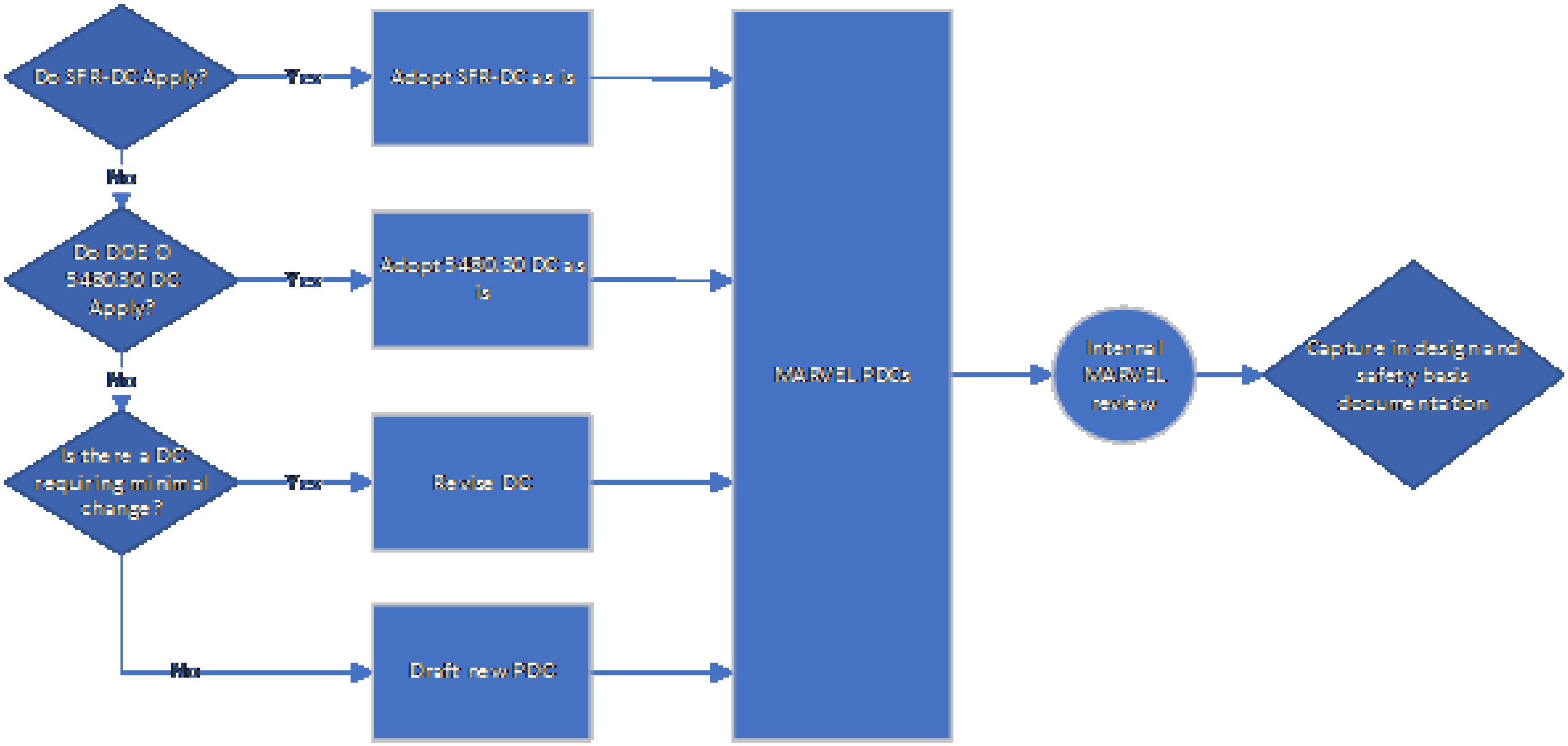

Figure 3. Flow chart of MARVEL PDC development methodology.

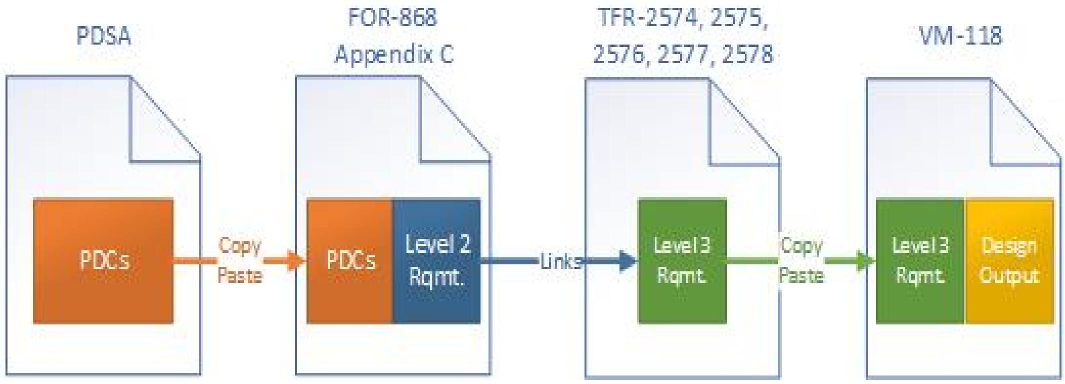

Figure 4. PDC translation into design requirements.

3. MARVEL REACTOR DESIGN DESCRIPTION

3.1 MARVEL Summary Description

The MARVEL microreactor is capable of generating up to 100 kilowatt-thermal (kWth) but will operate at a nominal 85 kWth. The operational life will be two effective full-power years, even though it may operate intermittently within the two year period. A visual of the MARVEL microreactor with support structures is shown in Figure 5.

MARVEL is a beryllium (Be) and graphite reflected, hydrogen (UZrH) moderated, solid fuel, loop-type reactor. The MARVEL reactor will contain 36 fuel rods sourced from TRIGA International. The fuel contains 30% uranium, by weight, 19.75 % enriched (TRIGA low-enriched uranium (LEU) 30/20), and the moderating material. Although similar to TRIGA fuel elements, MARVEL will differ in that each fuel rod will contain five fuel slugs instead of three. The MARVEL reactor core is composed of the 36 high-assay low-enriched uranium (HALEU) fuel rods in six subassemblies, arranged in a triangular pitch lattice around a central voided location for the central insurance absorber (CIA) rod assembly. Surrounding the core is a thick axial neutron reflector composed of beryllium oxide.

The primary coolant system (PCS) is a four-loop hydraulic circuit assembled to transport nuclear fission heat from the nuclear fuel to the intermediate heat exchanger (IHX) using the natural convection flow of the primary coolant. The PCS is a high-temperature, low-pressure boundary that houses the core internals, reactor primary coolant, and argon gas headspace. In addition, the PCS passively maintains decay heat removal capability. The boundary is a metal weldment made from 316H stainless steel for high-temperature reactors designed per American Society of Mechanical Engineers (ASME) Section III Division 5.

Approximately 120 kg of NaK liquid metal at room temperature, serves as the primary coolant. The heated liquid metal NaK rises above the top of the active core, through the upper grid plate and distribution plenum to the IHX. The IHX extracts the heat and cools the NaK. The secondary coolant system is a natural circulation loop of eutectic (e) Gallium (Ga)-Indium (In)-Tin (Sn) with compositions of 66.5Ga/20.5In/13.0Sn to 78Ga/15In/7Sn.

The selection of liquid NaK coolant and U-ZrH fuel with a natural circulation primary system arrangement provides a highly reliable reactor system with a large operational safety margin. This margin ensures that the system is not damaged during normal operations or off-normal events.

The coolant thermophysical properties provide superior heat removal and transport characteristics at low operating pressure with a large temperature margin to boiling. The U-ZrH fuel operates at a relatively low temperature, below the coolant boiling point. The NaK coolant also has high thermal conductivity which facilitates heat transfer from the fuel.

MARVEL utilizes commercial, off-the-shelf Stirling Engines from Qnergy. Each Stirling engine can produce 5-7 kilowatt and comes equipped with supporting ancillary equipment for low-grade heat rejection. Qnergy equipment will absorb available reactor heat and then convert that heat into either using electric power or low-grade heat for rejection into site processes or finally into the ultimate ambient heat sink. An alternate high grade heat extraction system can be optionally installed in the future for extracting high grade heat for process heat applications testing but is not within the scope of this report.

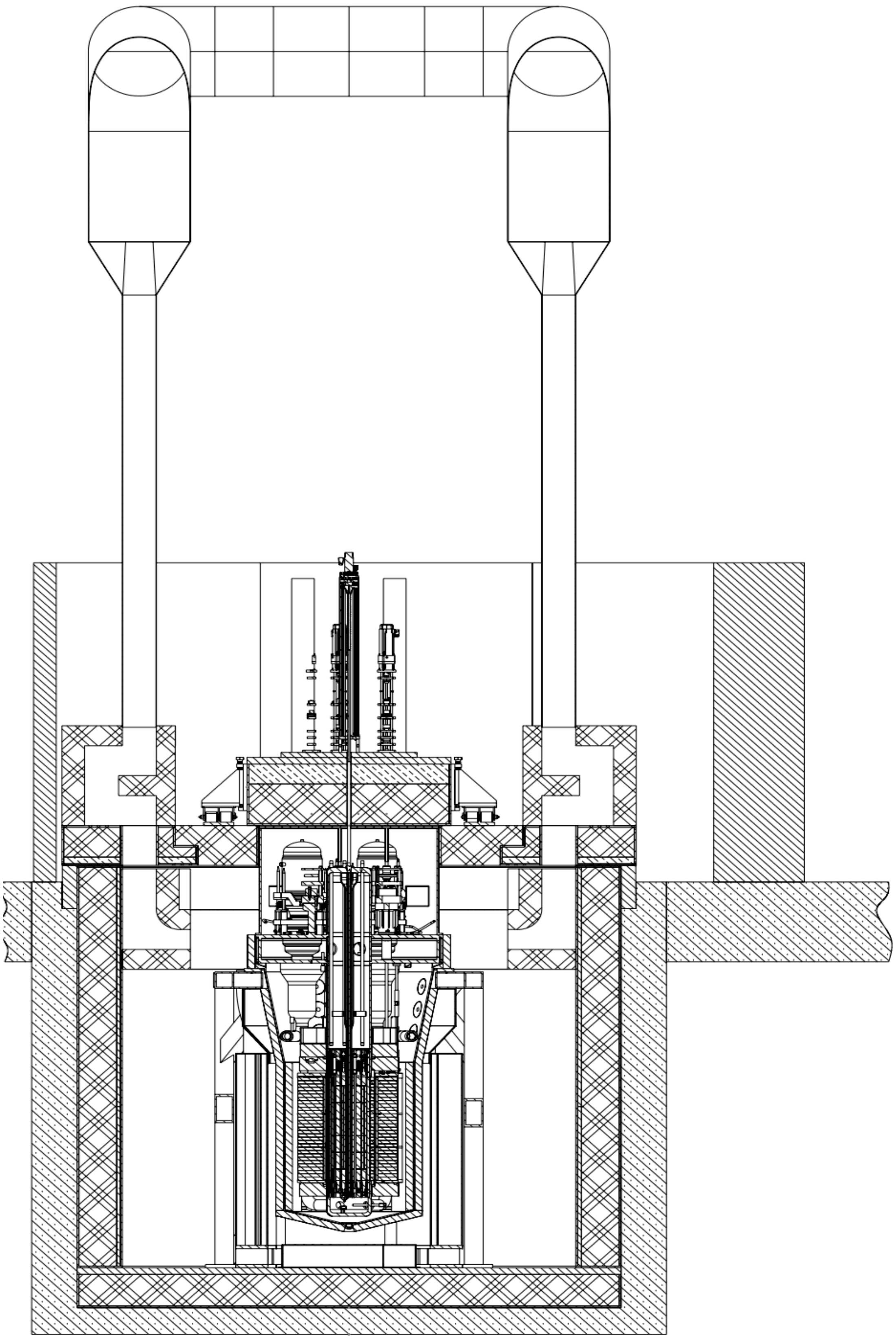

Figure 5. Illustration of 90% final design configuration of the MARVEL microreactor in TREAT pit.

The Stirling engine heat exchangers connect to the reactor vessel and interface with the NaK coolant via the IHX containing the eGa-In-Sn. The Stirling engine coils or high-grade heat exchanger, depending on configuration, extract heat from the primary coolant and reduce the NaK temperature. The cooled NaK then flows downward through four downcomer pipes where the NaK from each downcomers meet in the lower plenum. The NaK then rises back up through the active core under natural circulation forces driven by the heated or fueled section of the active core.

During normal shutdown operations, residual heat is removed via the power conversion heat exchangers. However, the low power density and large thermal mass also allow heat to be removed from the fuel by conduction throughout the system to the boundary of the guard vessel where it is removed by convection, radiation, and conduction to the environment without the use of the power generation system (PGS) heat exchangers. The small amount of decay heat generated by fission products in the reactor core after shutdown is thermally connected via conduction to large thermal masses provided by structures and shielding. This means fuel temperatures can remain below operating limits relying purely on passive conduction, convection, and radiation.

Additionally, instrumentation to ensure reliable plant control and early recognition of abnormal conditions is provided. MARVEL plant control SSCs are designed with considerations associated with ensuring that stable plant states are maintained during plant power changes, and control variables are evaluated to ensure that changes resulting in abnormal operations are minimal.

The reactor barrel acts as the up-flow coolant boundary . The side reflector is a BeO annulus that moderates and reflects neutrons back into the active core. The BeO annulus has two parts, the four rotatable control drums (CDs) and the stationary reflectors located between the CDs. The MARVEL reactor has elements used for reactivity control and shutdown. The first of those elements are the four CDs with boron carbide (B4C). The B4C is located on a third of the surface of the drum with the remaining volume being BeO, to act as neutron reflectors when not rotated in for shutdown. The second is the CIA rod that is in the central location of the core. The control rod is fully withdrawn during operation and used during shutdown.

The MARVEL reactor also has a very high net negative temperature feedback for prompt reactivity control inherently due to the Doppler broadening of resonances. Manual and Passive Reactivity Control and shutdown are achieved with four custom-designed, safety-related (SR) CDs and a defense-in-depth CIA that control reactivity via motor drives in the control cabinet to position the drums and rods within the reactor structure. Passive actuation functions are built into the design for loss of power and inadvertent energizations of the motors.

The core design uses a ternary fuel composition of UZrH. The cylindrical fuel pellets are stacked vertically and clad in stainless steel. The low burnup of the MARVEL design and the characteristics of U-ZrH fuel mean that most radionuclides remain in the fuel matrix over the course of the fuel lifetime.

A cutaway view of the MARVEL microreactor with all major systems is shown in Figure 6. The overall MARVEL reactor key design parameters are summarized in Table 2. Figure 7 shows the MARVEL drawing tree for reference in the following systems descriptions.

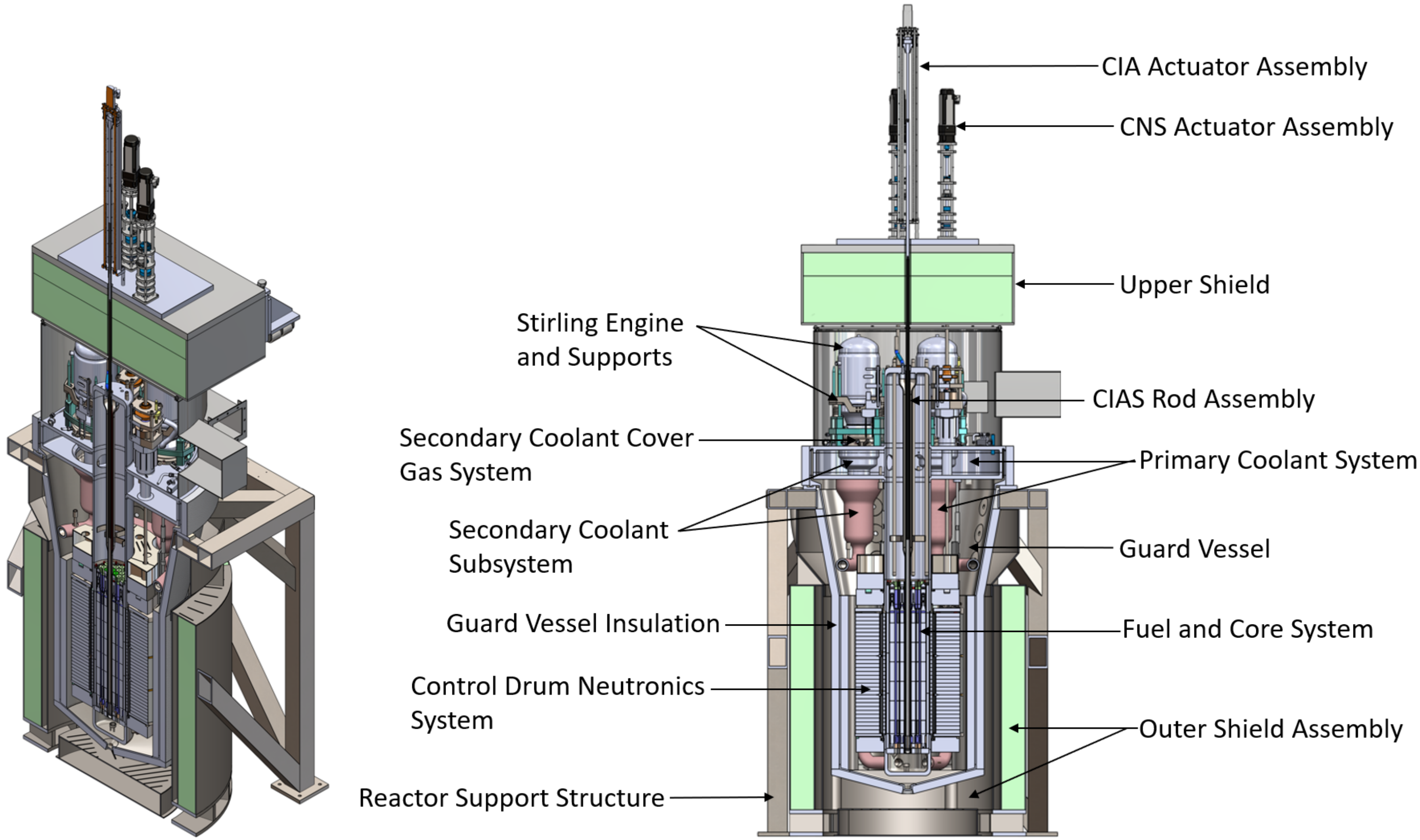

Figure 6. Illustration of MARVEL microreactor cutaway view.

Table 2. MARVEL overall design parameters.

Parameter	Value	Type
Nominal operating power (Thermal)	85	kWth
Electrical output	20	kWe
Fuel type	U-ZrH	
Number of fuel rods	36	
Zr/H ratio	1.6	
Uranium loading	30	wt %
U-235 enrichment	19.75	w %
U-235 loading (per rod)	235.0	g
U-235 loading (core total)	8.461	kg
Active fuel length	63.5	cm
Fuel pellet length	12.7	cm
Fuel pellet outer diameter	3.4823	cm
Pellets per rod	5	
Fuel pellet center annulus diameter	0.635	cm
Zirconium filler rod diameter	0.56134	cm
Axial reflector material	graphite	
Axial reflector diameter	3.279	cm
Top axial reflector length	6.604	cm
Bottom axial reflector length	8.687	cm
Fuel rod pitch	3.7916	cm
Fuel rod pitch geometry	triangular	
Fuel rod outer diameter	3.5916	cm
Cladding thickness	0.508	mm
Cladding material	304 stainless steel (SS)	
Reactor barrel inner diameter	27.6225	cm
Reactor barrel wall thickness (minimum)	0.3175	cm
Interior neutron reflector material	metallic Be (S65)	
Number of interior neutron reflectors	6	
Primary coolant	eutectic NaK	
Average coolant temperature (hot full power)	760	K
Secondary coolant	eGa-In-Tn	

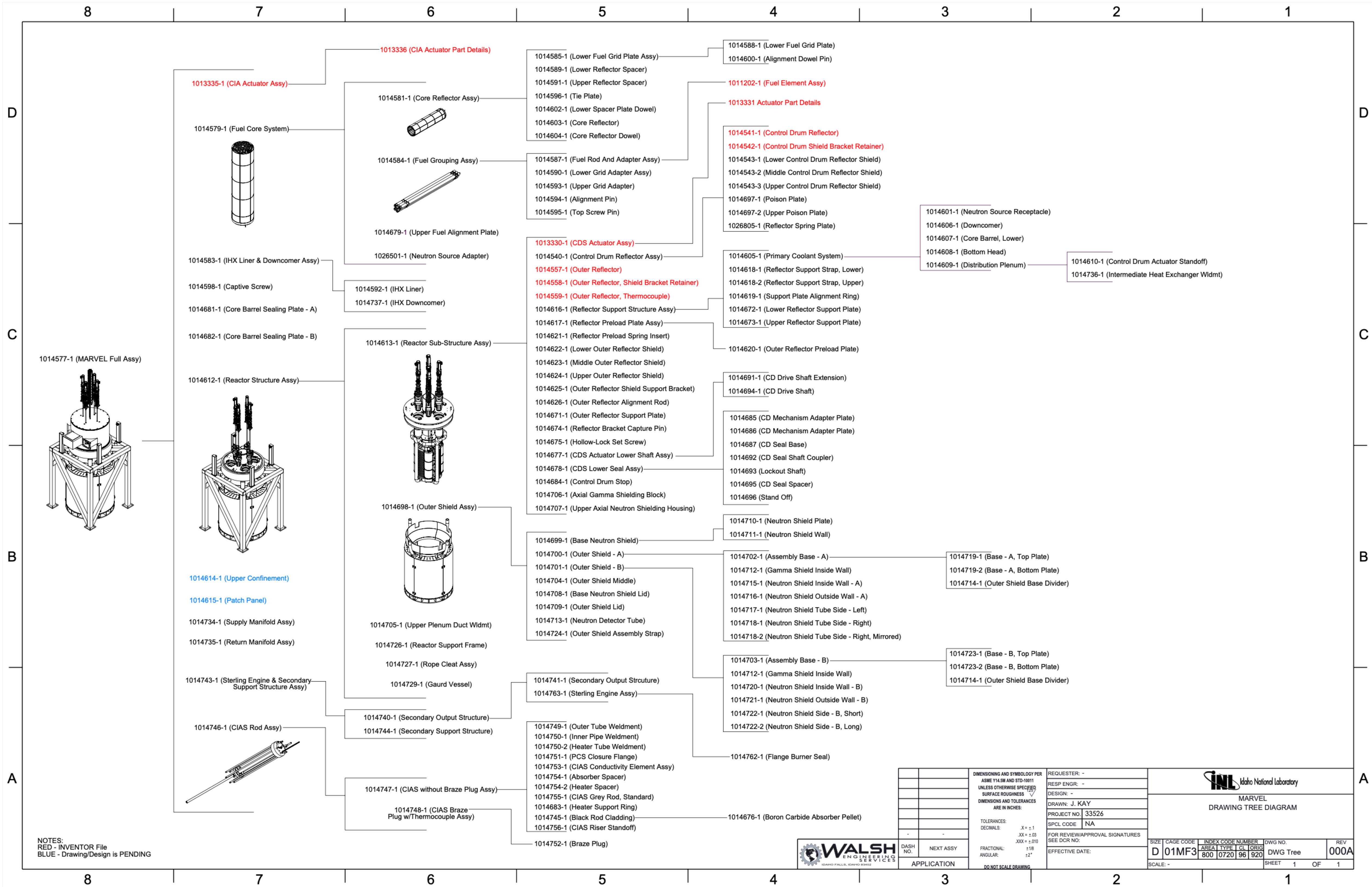

Figure 7. MARVEL mechanical SSCs drawing tree.

3.2 Safety Characteristics

MARVEL takes advantage of well-established uranium zirconium hydride (U-ZrH) fuel, NaK coolant, and structural materials that are stable and compatible. The selection of liquid NaK coolant and U-ZrH fuel with a natural circulation primary system arrangement provides a highly reliable reactor system with a large operational safety margin. This margin ensures that the system is not damaged during normal operations or off-normal events.

The coolant thermophysical properties provide superior heat removal and transport characteristics at low operating pressure with a large temperature margin to boiling. The U-ZrH fuel operates at a relatively low temperature, below the coolant boiling point. The NaK coolant also has high thermal conductivity which facilitates heat transfer from the fuel.

MARVEL produces only 85 kilowatt-thermal (kWth) power [nominal hot full power (HFP)]. Heat is transported from the fuel to the Power Generation System (PGS) via natural circulation of NaK coolant, which carries heat from the fuel to the PGS Stirling engine heat exchangers. During normal shutdown operations, residual heat is removed via the power conversion heat exchangers. However, the low power density and large thermal mass also allow heat to be removed from the fuel by conduction throughout the system to the boundary of the guard vessel where it is removed by convection, radiation, and conduction to the environment without the use of the PGS heat exchangers.

The small amount of decay heat generated by fission products in the reactor core after shutdown is thermally connected via conduction to large thermal masses provided by structures and shielding. This means fuel temperatures can remain below operating limits relying purely on passive conduction, convection, and radiation.

Additionally, instrumentation to ensure reliable plant control and early recognition of abnormal conditions is provided. MARVEL plant control structures, systems, and components (SSCs) are designed with considerations associated with ensuring that stable plant states are maintained during plant power changes, and control variables are evaluated to ensure that changes resulting in abnormal operations are minimal.

MARVEL uses four independent and redundant CDs to shut down the reactor and maintain it in a shutdown condition. In addition to this system, the MARVEL design benefits from favorable reactivity feedbacks that together with the low-pressure NaK coolant and reference metal hydride fuel, provide passive shutdown and passive safety behavior under various reactor upset conditions. MARVEL has a negative reactivity feedback due to thermal expansion of the fuel and structural materials, as well as doppler broadening. This feedback ensures reactor stability during operations and can help shut the reactor down should the reactor rise in temperature.

The CIA rod is an annular rod composed of boron carbide withdrawn vertically from the core. On a scram it uses gravity to insert into the core. Another feature of the CIA rod is the ability to incorporate a hafnium burnable absorber rod (gray rod) to adjust excess reactivity in the core and compensate for fuel burnup during the life of the core. The CIA rod alone can bring the reactor subcritical in all credible accident scenarios at a hot operation condition. However, The CIA rod by itself is not sufficient to hold the reactor shut down indefinitely. It is used as defense in depth (DID) to supplement the use of the CDs.

MARVEL has multiple layers and barriers to prevent the release of radionuclides. The fuel matrix and cladding provide the first and second barriers. The core design uses a ternary fuel composition of U-ZrH. The cylindrical fuel pellets are stacked vertically and clad in stainless steel. The low burnup of the MARVEL design and the characteristics of U-ZrH fuel mean that most radionuclides remain in the fuel matrix over the course of the fuel lifetime.

The PCS NaK coolant acts as the third radionuclide barrier by retaining fission products by plate-out, chemical solubility, or adsorption mechanisms. The PCS boundary (PCB) design which includes the reactor (fourth barrier) and any associated piping, including the downcomers, ensures activated NaK and any leaked fission or activation products remain DID to the release of radionuclides to the environment. The guard vessel forms a partial fifth barrier to the PCB in areas where liquid NaK is located.

The MARVEL design is capable of accommodating various design basis accident (DBA) and beyond DBA (BDBA) basis accident initiators (See Section 7.3) without producing conditions that might lead to a severe accident and release of radioactive or hazardous materials. The inherent and passive features of the system are responsible for bringing the system to a stable state at safe temperatures. The passive performance mechanisms for ensuring reactivity control and cooling provide performance with generally stronger feedbacks as temperatures increase. These design features help to control the level of severity of facility upsets.

Additionally, the various levels of confinement barriers (fuel matrix, cladding, coolant, reactor barrel, guard vessel) provide thresholds that serve to control the release of radioactive material if facility conditions are severe enough to result in fuel failures and releases. Finally, significant adverse consequences from hypothesized releases of radioactive or hazardous materials are limited by the MARVEL limited core size and fission product inventory.

The safety-in-design strategy is implemented by conservative design for the FSFs of reactivity control, DHR, and confinement of radioactive materials. Success in meeting the objectives of the overall safety-in-design strategy is shown by virtue of the fact that all DBAs analyzed are successfully mitigated by the SR-SSCs performing the FSFs.

The ultimate means of protection of public and worker safety from the consequences of postulated DBA loss-of-cooling and transient overpower events without scram (unprotected) will be the negative inherent reactivity feedback (IRF) resulting from reactor system temperature increases. To ensure that the design incorporates this inherently safe response capability during postulated DBA’s (combining accident initiators with no control drum (CD) actuation) passive design requirements will be imposed on the design in the MARVEL Technical Specifications (TSs).

3.3 Fuel and Core System

3.3.1 Design Description

This section describes the fuel and core system (FCS) design. Figure 8 as taken from TFR-2577, shows the physical boundaries of the subsystems within the FCS described in this Section. Boxes surrounding the system boundary represent interfacing subsystems not within the scope of the FCS. Red lines represent thermal interfaces, purple lines represent nuclear interfaces, teal lines represent instrumentation and control (I&C) interfaces, and black lines represent important mechanical or structural interfaces.

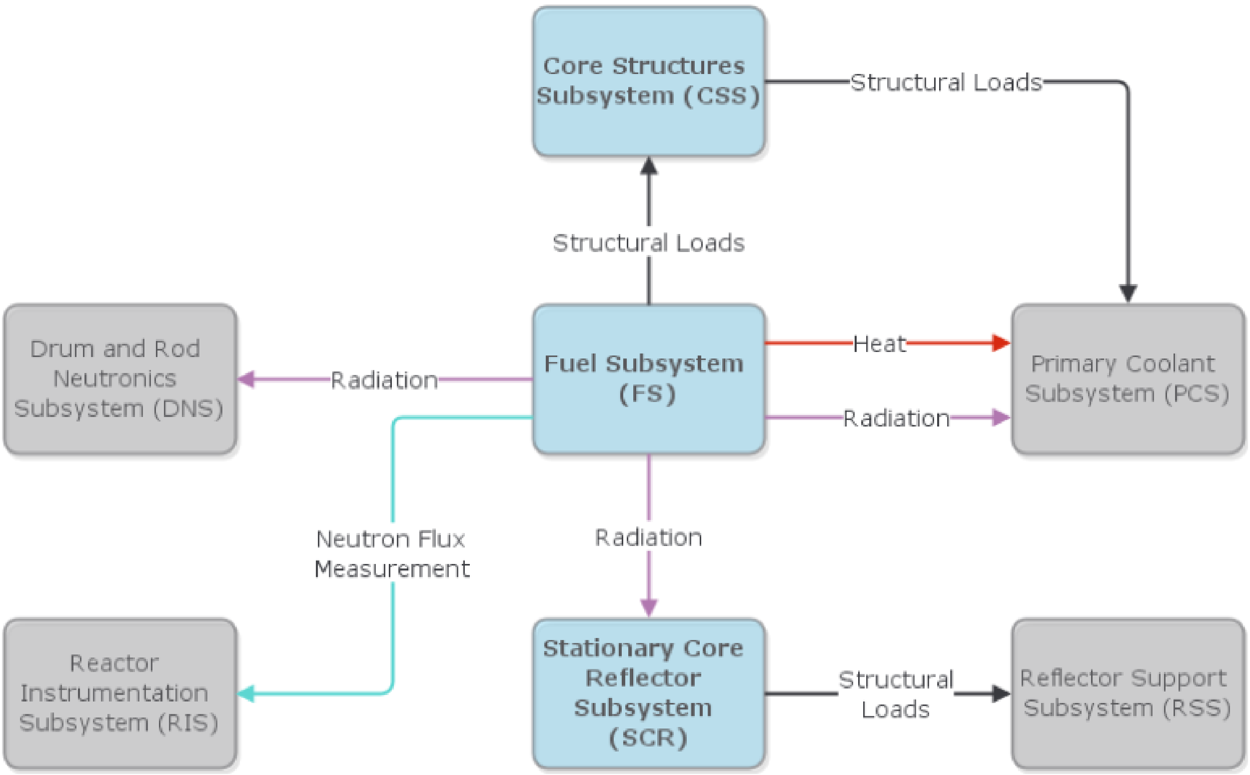

Figure 8. General system diagram of FCS with interfaces.

The Fuel Subsystem (FS) is designed to produce a controlled self-sustaining fission chain reaction generating a nominal 85 kWth. The fuel contains high-assay low-enriched uranium (HALEU) enriched to 19.75% and Uranium-Zirconium Hydride (UZrH) as a moderating material. The thermal energy generated by nuclear fission is transferred from the fuel matrix through the cladding to PCS. The fuel pin cladding acts as the primary pressure boundary and fission product confinement boundary. The PCS is described in Section 3.4.

Control of the nuclear chain reaction is accomplished with four control drums during operation. The shutdown of the reactor is accomplished with the control drums and ensured with a CIA rod. These Reactivity Control System (RCS) components are described in Section 3.5.

The MARVEL microreactor is a high-leakage system. This means that many of the neutrons generated in the core escape the system boundaries due to its small size and the fact that moderation is contained only within the fuel. The fission chain reaction is made possible by the axial and radial neutron reflectors by reducing loss of neutrons from the core due to leakage. The axial neutron reflectors are part of the FS fuel pin. Smaller Beryllium radial inserts are provided as part of the Core Structures Subsystem (CSS) within the core region. Outside of the primary coolant boundary, additional larger radial Beryllium

Oxide (BeO) neutron reflectors are provided as part of the Stationary Core Reflector Subsystem (SCR). These reflectors are supported by the Reflector Support Subsystem (RSS). The neutron reflectors also provide some moderation of the neutrons, which assists to increase fission in the core.

The assembly of 36 fuel elements into the reactor core is supported inside the reactor vessel by the CSS components. These components hold the fuel elements with the correct spacing to ensure that coolant can flow in all subchannels around the fuel to maintain a coolable core geometry.

The selected fuel material, UZrH, and the geometry of the fuel-supporting CSS components provide an important safety function known as IRF. This means that, by properties of natural physical phenomena, the reactor has multiple sources of self-limiting feedback. As the reactor increases in power and temperature it takes more reactivity to sustain the nuclear chain reaction. This ensures that a positive power feedback loop leading to a runaway power increase cannot physically occur.

The FCS interfaces with the PCS and reflector support structure (RSS) of the MRS through heat transfer and physical contact. The MRS including the reactor support structure and PCS SSCs is discussed in detail in Section 3.4.

The FCS also interfaces with the RCS of the ICS through neutron radiation, reflection, and absorption. Due to the passive operation of the FCS with no moving parts, there are no significant system reliability features other than the original design of the components in the subsystems. There are factors of safety built in to ensure proper function of the subsystems in off-normal or accident scenarios. There are no significant system control features in the FCS alone. The RCS is discussed in detail in Section 3.5. The ICS is discussed in Section 3.6.

3.3.2 Design Bases

The FCS SSC safety classifications are provided in Appendix B. Key safety functions of the FCS SSCs are to:

- Provide negative reactivity insertion as a function of temperature increase such that the resulting reactor power is reduced to passive heat rejection levels before fuel and PCB temperature limits are challenged and core damage occurs.
- Provide structural, mechanical, and geographic spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible.
- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and accident conditions.
- Retain radionuclides within the fuel matrix.
- The cladding around the fuel provides a barrier for gaseous fission products.
- Fuel and cladding structure design to remain within temperature limits to maintain core coolable geometry.

3.3.2.1 Design Criteria. In addition to the overall principal design criteria (PDC) 1-5 in Appendix A, the FCS SSCs shall be designed to meet the following PDC:

- The FCS shall be designed with appropriate margins to assure that specified fuel, clad, and PCB temperature limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences [PDC-10].
- The FCS 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 [PDC-11].
- The FCS shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed [PDC-12].

- The FCS shall provide for the control of the release of radioactivity to the environment and to ensure that the functional confinement barrier design conditions are not exceeded for as long as postulated accident conditions require [PDC-16].
- The FCS shall be designed to provide negative reactivity feedback as the temperature of the reactor increases such that the any accidental positive reactivity insertion is passively counteracted and the reactor is brought to new stable state before fuel, clad, and PCB temperature limits are challenged and core damage occurs [PDC-26].
- The FCS shall be designed to remove all heat from fission and radioactive decay, such that fuel and Barrel temperature limits and the design conditions of the PCB are not exceeded, and safe shutdown is achieved and maintained during normal operation, anticipated events, and postulated accident conditions [PDC-34].
- The FCS shall facilitate decay heat removal in all operating scenarios, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded [PDC 44].
- The FCS shall be designed to be subcritical during handling operations outside the reactor [PDC-61].

3.3.2.2 Functional and Operating Requirements. The requirements and bases for the FCS are discussed in detail in FOR-868 and TFR-2577. Key system functional requirements are as follows:

- The FCS shall be designed to generate thermal power via sustained nuclear fission.
- The FCS shall be designed to transfer all fuel matrix heat into the primary coolant, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.
- The FCS shall prevent the leakage of neutrons from the core region necessary to support sustained nuclear fission.

Key system operational requirements are as follows:

- The FCS shall be designed to withstand the credited environmental conditions (e.g., temperature, pressure, radiation, etc.) in which it is installed for at least 2 calendar years of operation, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.
- FCS SSCs important to safety shall be designed to withstand the effects of seismic events without the loss of the capability to perform their safety functions.
- The FCS shall contain sufficient excess reactivity for at least two calendar years of operation.
- The FCS shall be designed to provide sufficient negative reactivity feedback as the temperature of the reactor increases such that it is self-protecting and does not require action from the reactivity control systems to limit reactivity excursions.
- The FCS shall provide barriers that prevent or mitigate the release of radioactive materials from the reactor to the public and the environment during normal operations, anticipated events, and postulated accident conditions.
- The FCS shall provide structural support for internal fuel and core SSCs.

3.3.3 Description of Systems, Subsystems, and Major Components

3.3.3.1 Fuel Subsystem (FS). The basic unit of the FS is the completed fuel rod. There are 36 fuel rods in the FS. The major components of the MARVEL fuel system are listed in Table 3.

MARVEL fuel elements will contain the following (Figure 9): five (5) uranium-zirconium hydride (U-ZrH) each housing an internal zirconium rod, two (2) cylindrical graphite neutron reflectors (one on top of the fuel meat stack, and one on the bottom of the fuel meat stack), one (1) molybdenum disks to separate the bottom graphite reflector from the fuel, and top and bottom cladding endcaps made of 304 stainless steel (Figure 10). The fuel meats will be of similar dimension and composition to TRIGA fuel meats, thereby maintaining the prompt negative temperature coefficient associated with TRIGA fuel. The nominal values and tolerances of each of the fuel element components are summarized in Table 4. Note that the tolerances for the fuel meat outer diameter and cladding inner diameter indicate that a gas gap exists between the fuel meat and the cladding in their as-fabricated room-temperature conditions.

As described in detail in INL/RPT-22-6855, "MARVEL Reactor Fuel Performance Report,"[20] the HALEU will be provided by Y-12. The fuel meat and internal rod (Figure 11) will be fabricated from reactor grade zirconium meeting the requirements of ASTM B349 and ASTM B351 (Grade R60001), respectively. The molybdenum disc will be Type 361 molybdenum meeting the requirements of ASTM 386. The space between the as-assembled fuel/graphite and cladding, henceforth referred to as the gas gap, accommodates fuel and internal gas expansion during the fuel's operational cycle (as does the void space above the top graphite reflector). The gas gap will be composed of atmospheric pressure room temperature air when the fuel pin is hermetically sealed during assembly. Forthcoming analysis will show that increasing the number of fuel meats per element from 3 or 4 (TRIGA fuel) to 5 (MARVEL fuel) does not negatively impact fuel performance.

Table 3. MARVEL FS components and functions.

Subsystem	Components	Parameters	Functions
Fuel system	Cladding	Type 304 Stainless steel	• Contains fuel • Contains hydrogen • Contain fission products and hydrogen • Transfer heat to primary coolant
	Fuel	$U\text{-}ZrH_{1.6}$	• Generate thermal energy through fission • Moderation through abundance of H atoms • Retain fission products
	Axial reflector	Graphite	Reflect neutrons
	Fuel to clad bond	Air/mechanical bond	Transfer heat to cladding

Table 4. MARVEL fuel rod design parameters and specifications.

Parameter	Nominal Value [a]	Tolerance
Fuel Meat H/Zr Ratio	1.60	+0.05/-0.03
Fuel Meat Uranium Content	30 wt%	-- [b]
Enrichment	19.75 wt%	+0.20/-0.20 [c]
Fuel Meat Erbium Content	0	N/A
Fuel Meat Inner Diameter	5.72 mm (0.225 in.)	+0.25 mm/-0.25 mm (+0.010 in./-0.010 in.)
Fuel Meat Outer Diameter	34.79 mm (1.370 in.)	+0.0 mm/-0.50 mm (+0.0 in./-0.020 in.)
Fuel Meat Height [d]	127.00 mm (5.000 in.)	+0.0 mm/-1.50 mm (+0.0 in./-0.059 in.)
Central Zr Rod Diameter	5.61 mm (0.221 in.)	+0.10 mm/-0.10 mm (+0.004 in./-0.004 in.)
Central Zr Rod Height [d]	127.00 mm (5.000 in.)	+0.00 mm/-0.25 mm (+0.000 in./-0.010 in.)
Top Graphite Diameter	32.74 mm (1.289 in.)	+0.05 mm/-0.05 mm (+0.002 in./-0.002 in.)
Top Graphite Height	66.04 mm (2.600 in.)	+0.20 mm/-0.20 mm (+0.008 in./-0.008 in.)
Molybdenum Disc Diameter	34.65 mm (1.364 in.)	+0.05 mm/-0.05 mm (+0.002 in./-0.002 in.)
Molybdenum Disc Height	0.79 mm (0.031 in.)	+0.10 mm/-0.10 mm (+0.004 in./-0.004 in.)
Bottom Graphite Diameter	32.74 mm (1.289 in.)	+0.05 mm/-0.05 mm (+0.002 in./-0.002 in.)
Bottom Graphite Height	86.87 mm (3.420 in.)	+0.20 mm/-0.20 mm (+0.008 in./-0.008 in.)
Cladding Outer Diameter	35.92 mm (1.414 in.)	--
Cladding Thickness	0.51 mm (0.020 in.)	+0.04 mm/-0.03 mm (+0.0016 in./-0.0012 in.)
Cladding Height	818.44 mm (32.222 in.)	+0.76 mm/-0.76 mm (+0.030 in./-0.030 in.)
Plenum Void Gap Height [e]	12.17 mm (0.479 in.)	--

a. The values shown here are associated with the components' as-fabricated, unirradiated, room-temperature conditions
b. Information is defined by the manufacturer and not shared here.
c. From Ref 20.
d. 5 per MARVEL fuel element
e. The top and bottom fittings occupy some of the space within the ends of the fuel element.

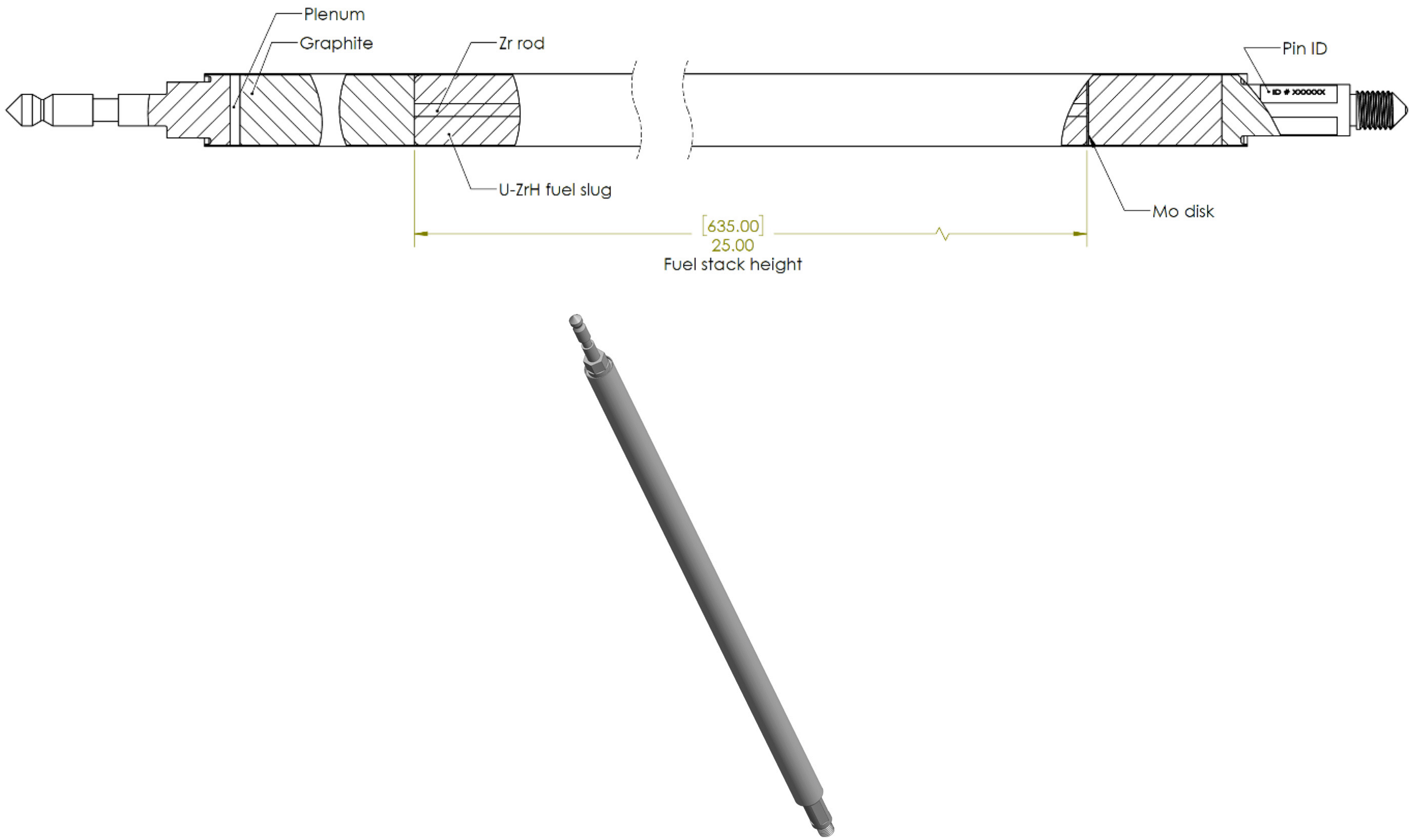

Figure 9. (Top) Schematic and (bottom) isometric illustration of a MARVEL fuel element.

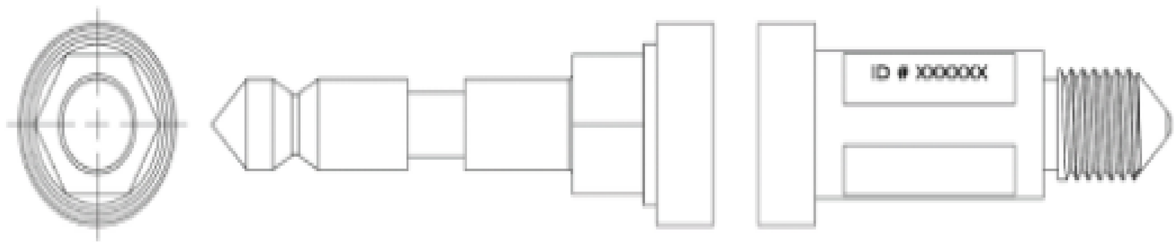

Figure 10. 304 stainless steel end plugs of a MARVEL fuel pin.

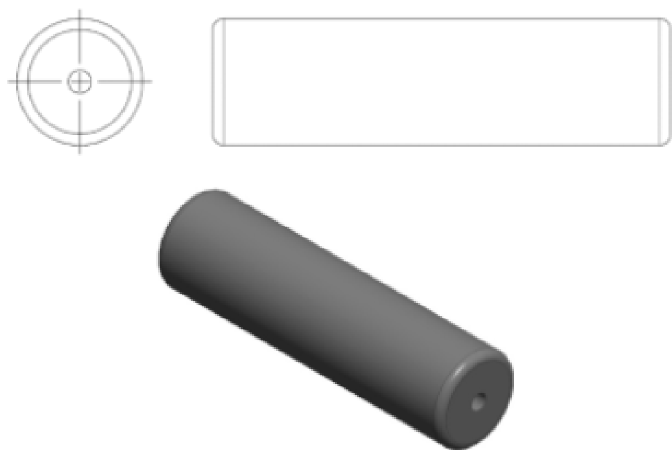

Figure 11. (Top) Schematic and (bottom) isometric illustration of an annular MARVEL U-ZrH fuel meat.

3.3.3.2 Core Structures Subsystem (CSS). The CSS include the components that support the fuel inside the primary coolant system (bottom grid plate, top grid plate(s), alignment plates/rings, fuel subassembly components, etc.).

See Figure 12 through Figure 17 and Table 5 for a detailed description of the CSS. The fuel rod assemblies will be held in place and aligned using an upper and lower grid adapter assembly (Type 316H stainless steel). The lower grid adapter takes the structural load of the entire FCS.

The CSS operates passively by the nature of its material properties and location in relationship to the fuel and the primary coolant boundary. There is no need for active operation. The CSS provides a load-bearing path to support the fuel into the MRS. This is performed through the selection of high temperature steel parts. Additionally, the CSS has a neutron reflection and shielding function which it provides by the material properties of the metallic beryllium reflector plates.

The CSS provides negative inherent reactivity feedback by nature of the material properties of the SS316H used in the bottom grid plate and top alignment plates. The geometry of these components also plays a role in providing negative IRF. By nature of the relatively high thermal expansion, these components can provide significant negative IRF as the coolant temperature increases, which adds to the safety of the reactor design.

The CSS has physical bounds radially on the inside surface of the reactor vessel in the core region and the outermost surface of the CIA rod housing in the central location in the reactor core. Axially, the CSS is bounded by the bottom of the neutron source attached below the lower fuel grid plate and the upper surface of the upper fuel grid plate. The CSS is entirely within the PCB and does not include any components external to the reactor vessel.

The overall thickness of the lower grid plate is approximately 1.346 inches, and the outer diameter is approximately 10.80 inches. The thickness was selected to adjust the height of the fuel region (fuel meats and graphite regions) of the fuel rods to center them with the center of the beryllium oxide reflectors. The geometry utilized is optimized for maximum primary coolant flow through the lower grid plate. The fuel assemblies are restricted at the top and bottom via the grid plates to reduce curvature of the fuel elements over time. The top grid plate is designed to allow axial expansion of the fuel rods by a floating design.

Table 5. CSS component descriptions.

Description	Function	Material	Total Qty.
Fuel Pin Collar	The transfer the load of the fuel pin and capture component number 2.	SS316H	36
Insert	Adapts a fuel pin to be lifted with the top cap of the fuel pins and component numbers 1, 3, 8, 9, and 10.	SS316H	72
Fuel Pin Sleeve	The transfer the load of the fuel pin and capture the pin inserts to aid in lifting the fuel pins.	SS316H	36
Lower Grid Plate Assembly	Provides structural support for the fuel pins and other CSS components. Ties each of the groups of six fuel pins to each other as an array to maintain distance from each other.	SS316H	1
Thread to Pin Adapter	Allows a group of six fuel pins to be threaded together prior to be lowering into the core barrel. Provides appropriate spacing between each of it's six fuel pins.	SS316H	6
Small Alignment Pins	Aligns component 16 with component 4.	SS316H	12
Large Alignment Pins	Locates the grid plate to ensure the placement of the clocking of the fuel is as designed.	SS316H	3
Locating Pin	Locks the fuel pin to the group of six fuel pins. Additionally, it provides a means to tie the group of six pins to the other groups of six pins.	SS316H	18
Top Screw Pin	Locks the fuel pin to the group of six fuel pins. Additionally, it provides a means of attaching by thread to a fabricated crane lifting attachment.	SS316H	18
Top Adapter	Provides appropriate spacing between each of it's six fuel pins.	SS316H	6
Neutron Source	Provides neutron feedback to the neutron detectors while fuel loading and give additional reactivity to start the MARVEL nuclear core.	?	1
Upper Spacer	316H Stainless Steel, Matches the cross-sectional geometry of the core reflectors.	SS316H	6
Tie-Plate	Ties the Upper Spacers to each other	SS316H	1
Upper Grid Plate	Ties each of the groups of six fuel pins to each other as an array to maintain distance from each other.	SS316H	1
Top Seal	Seals the outer void between the core barrel and the Upper Grid Plate to ensure no parasitic flow occurs between them. The top seal is made from 316H Stainless Steel and acts as a "Belleville washer" to increase in diameter as screws are tightened to through to the upper metallic beryllium spacer. This allows for the top seal to be a smaller diameter for installation and a larger diameter to prevent parasitic flow.	SS316H	1
Lower Spacer	Provides support for the Core Reflectors and aids in raising the core reflectors to the height of the graphite region of the fuel pins.	SS316H	6
Core Reflector	Provides neutron reflection. For more information about this component see appropriate sections of this document.	Be (S200)	36

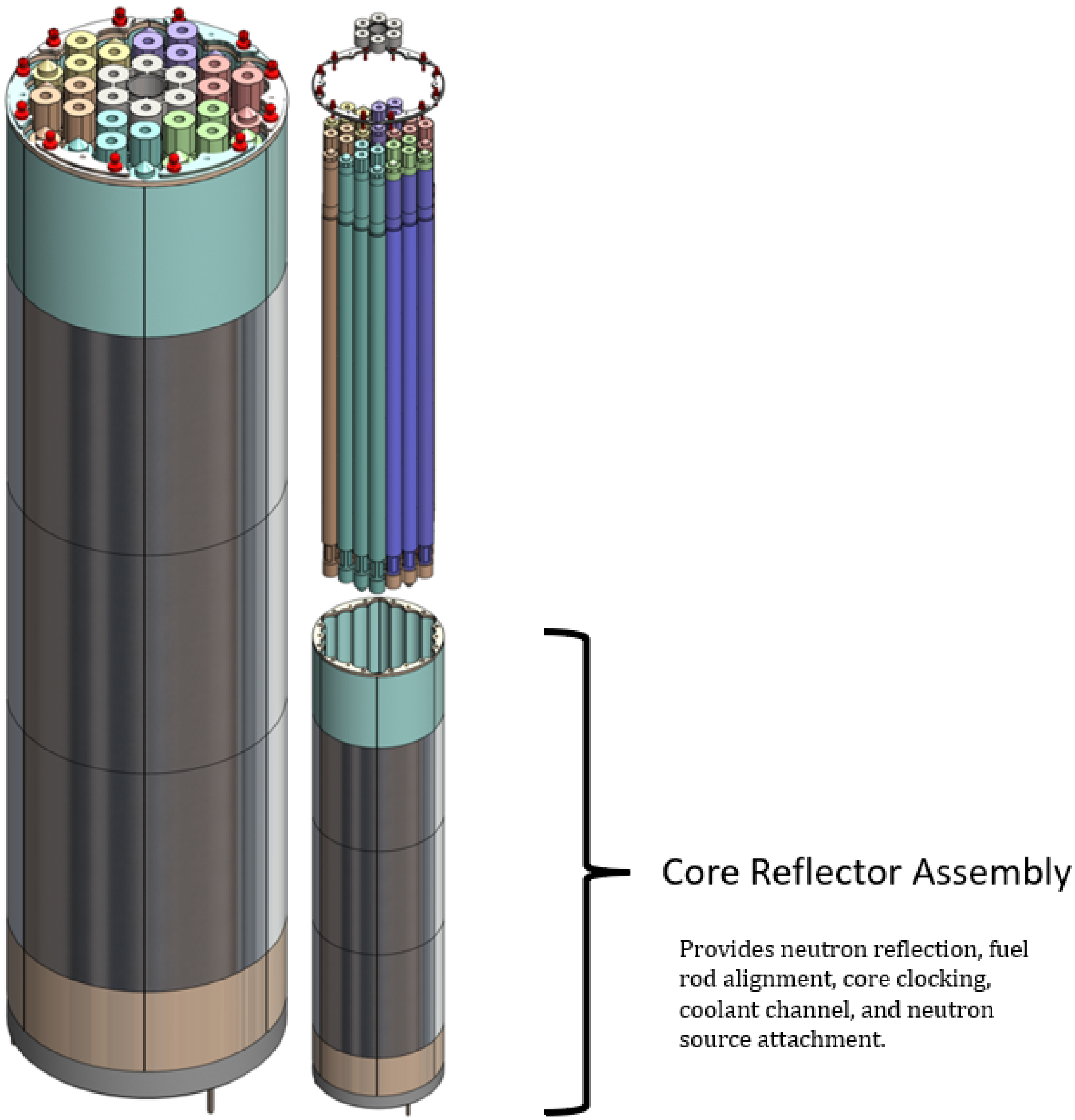

Figure 12. CSS components (1 of 6).

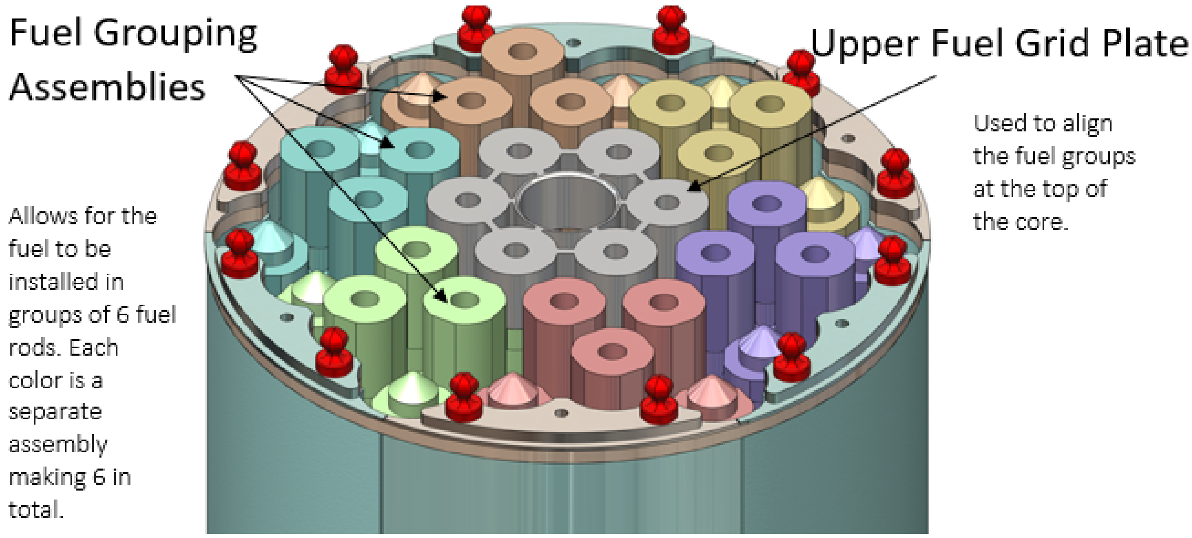

Figure 13. CSS Components (2 of 6).

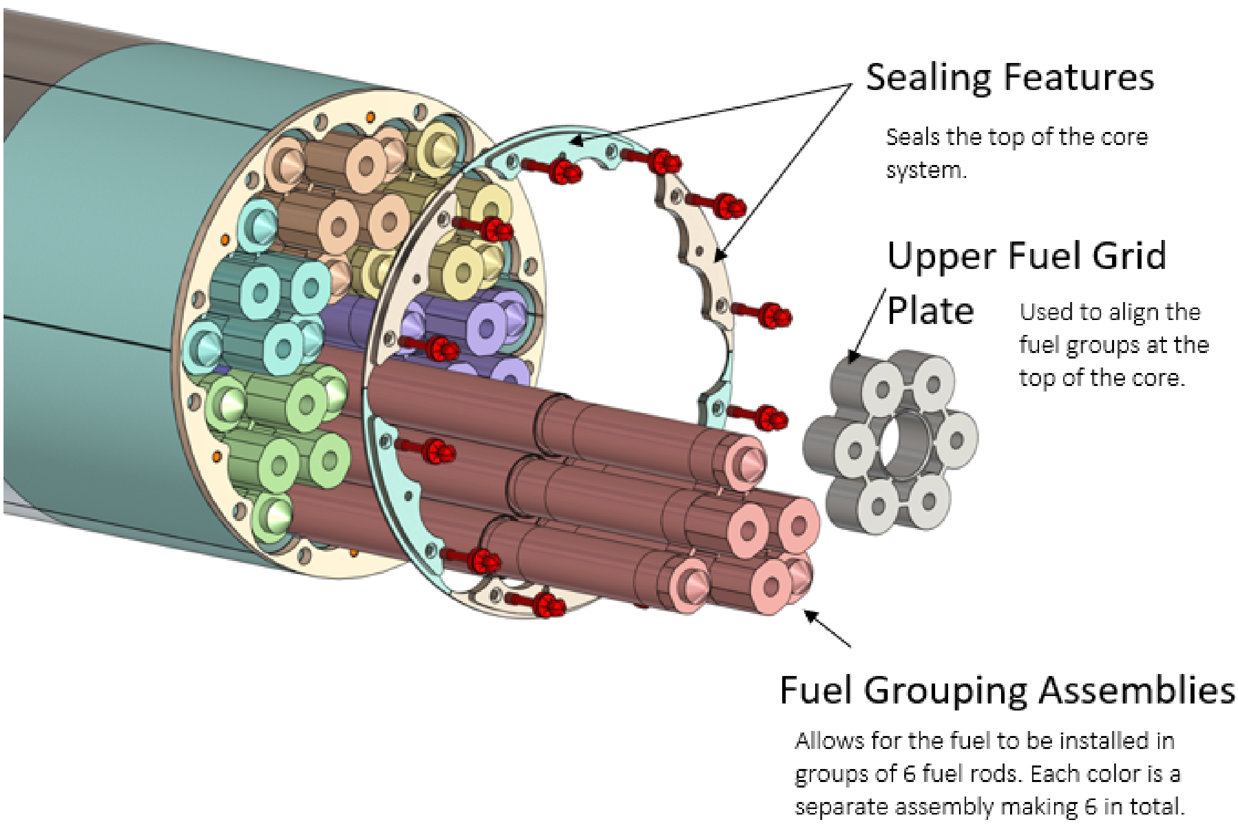

Figure 13. CSS components (3 of 6).

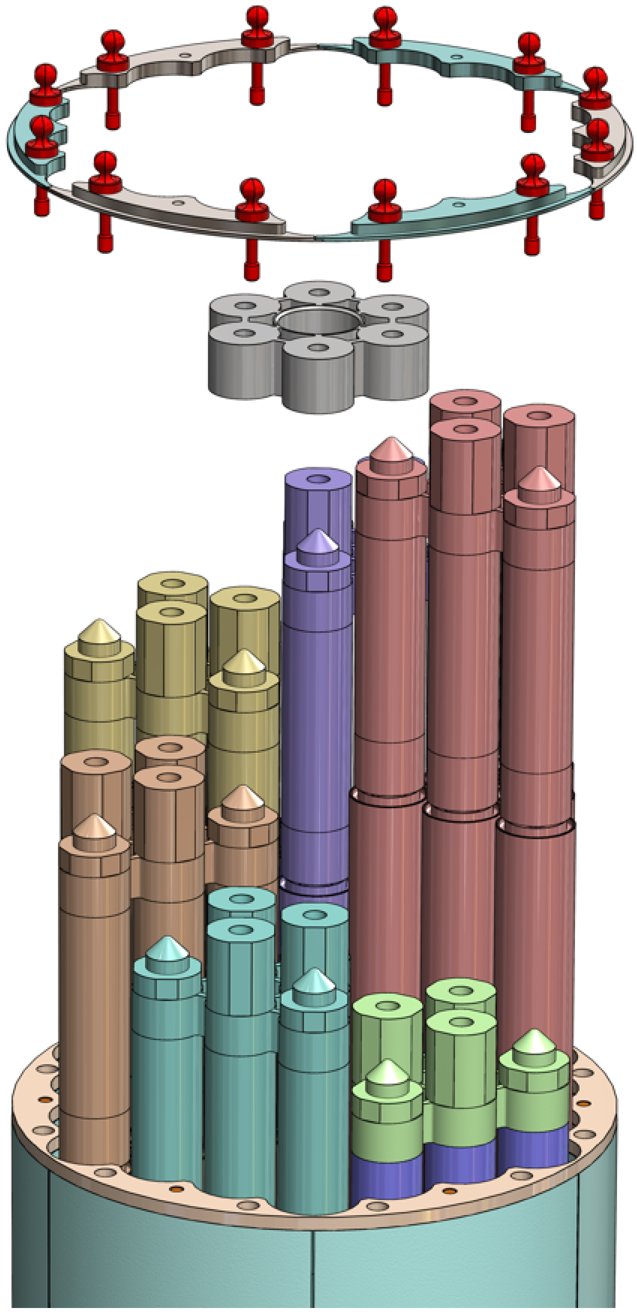

Figure 14. CSS components (4 of 6).

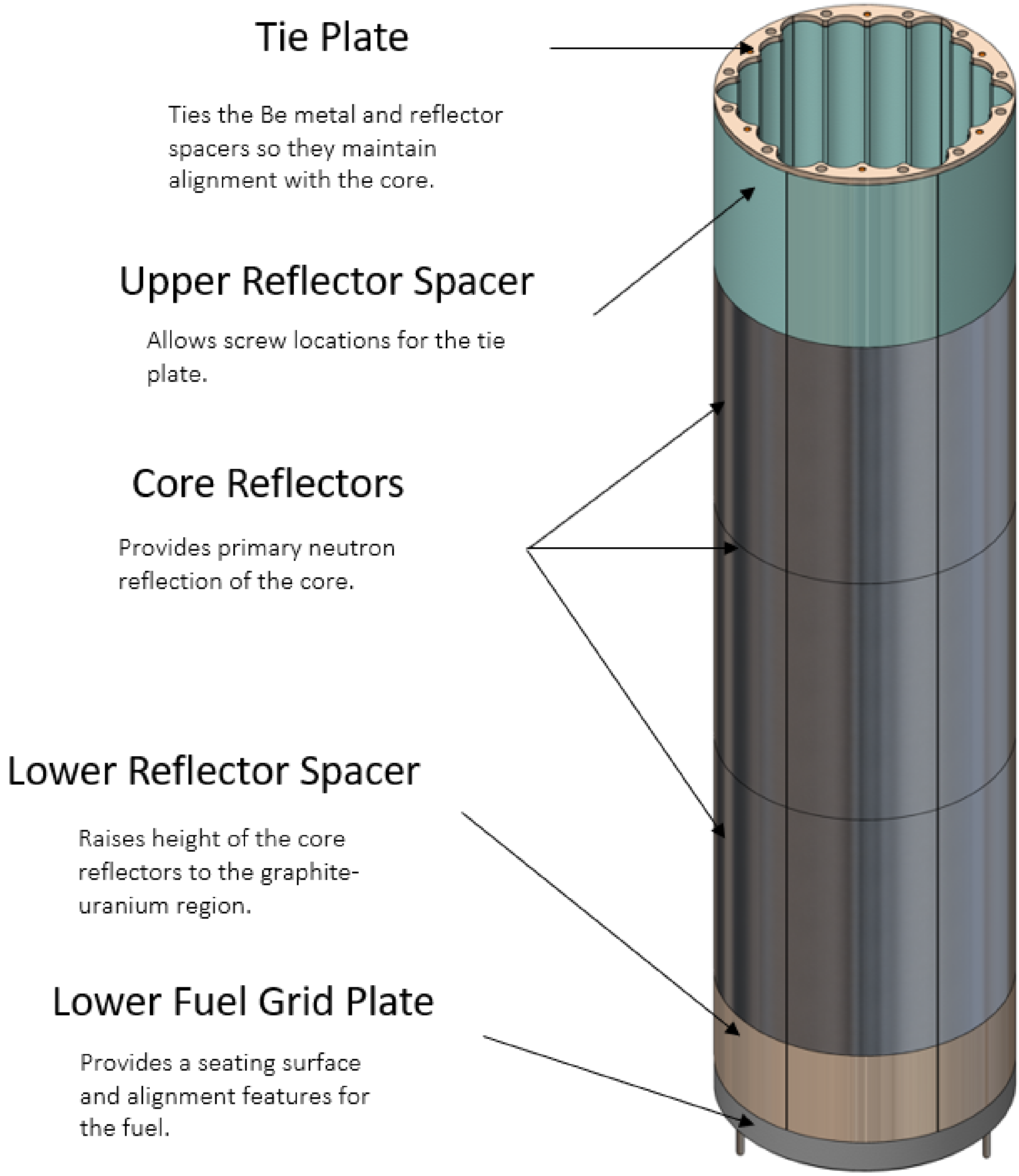

Figure 15. CSS components (5 of 6).

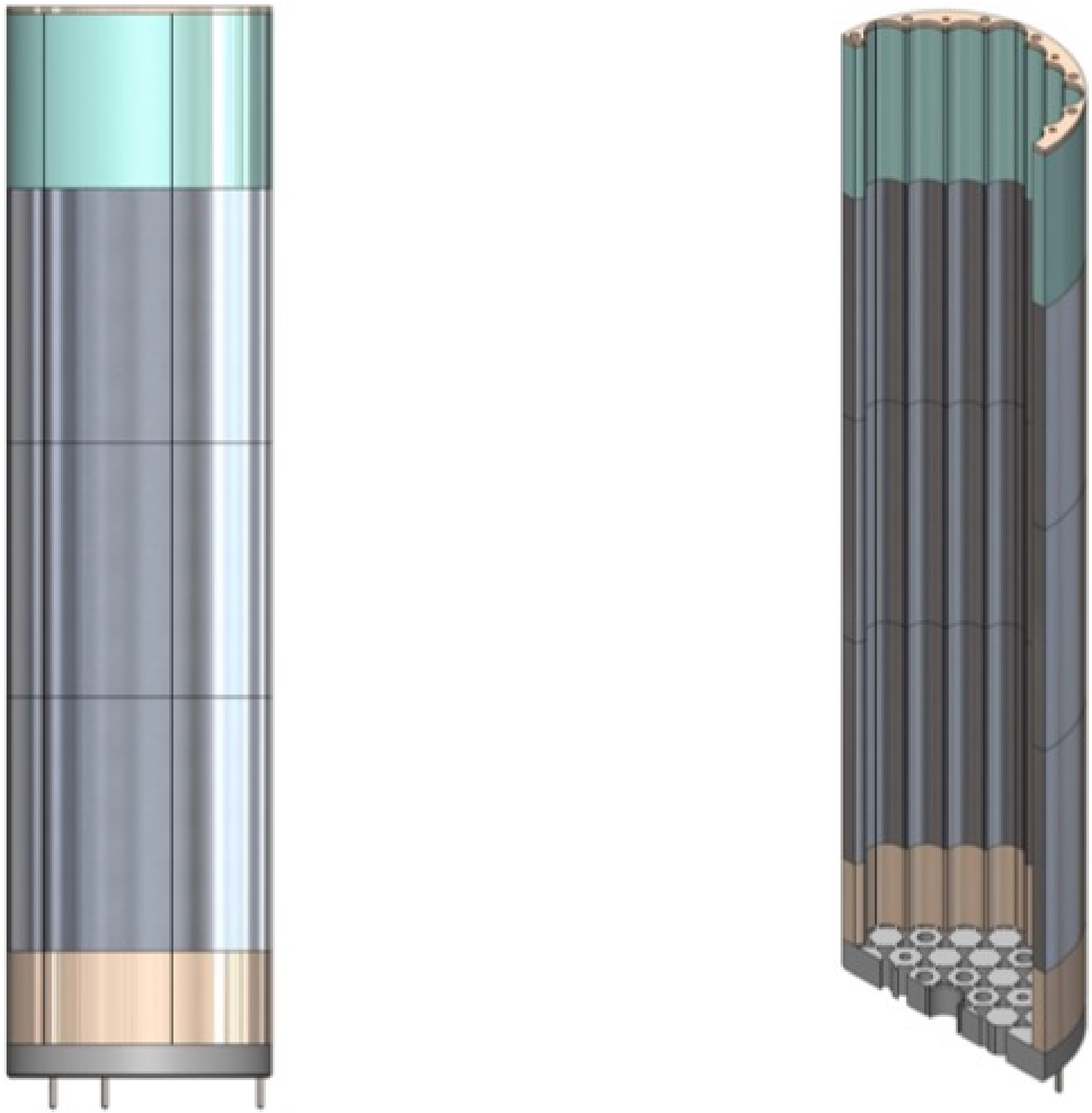

Figure 16. CSS components (6 of 6).

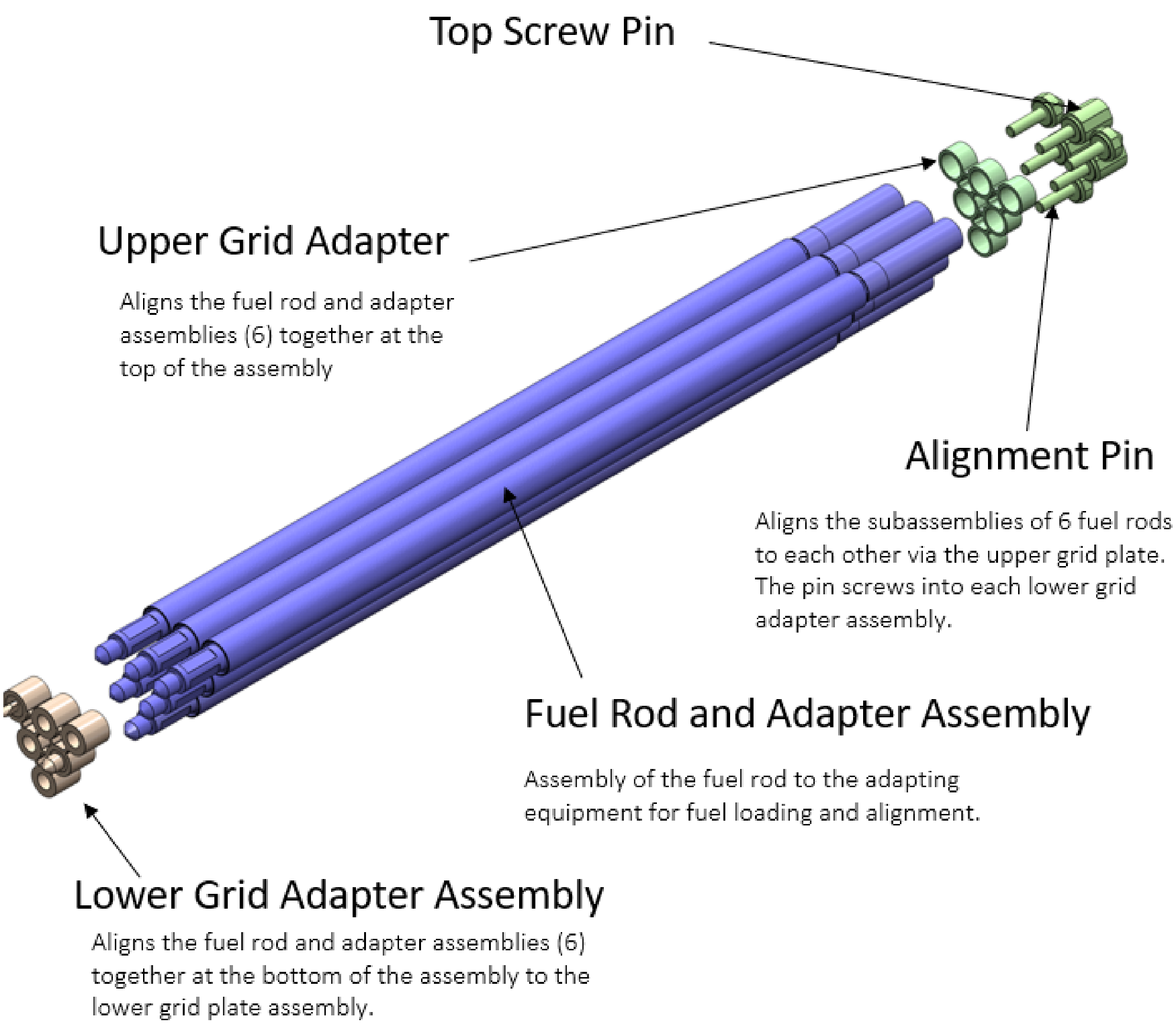

Figure 17. Fuel grouping assembly -exploded view.

3.3.3.3 Stationary Core Reflector Subsystem. The SCR (Figure 18) is a simple system that consists of a single material, BeO. All of the components are one-inch-thick plates that are stacked together into four stacks and spaced equiangularly around the reactor barrel. There are three versions of the BeO plates in design to accommodate thermocouples and the debris shield brackets to hold the stationary reflector debris shields in the MRS.

The SCR consists of 4 stacks of BeO plates located immediately outside the reactor barrel 90 degrees apart from each other creating a cylindrical gap where the control drums will reside. A progression of the assembly sequence for the SCR is shown in Figure 19. The three different color plates are nearly identical with only minor modifications to accommodate thermocouples or structural components such as brackets. The location of the SCR in relationship to the fuel, control drums and reflector support plate is shown in Figure 20. The SCR interfaces with the MRS, ICS, RCS, and the FS through physical contact or neutron radiation.

The SCR operates passively by the nature of its material properties and its location in relationship to the fuel. The primary function of the SCR is to reflect neutrons back into the core to decrease loss of neutrons from leakage. By the action of reflecting neutrons back into the core, the SCR also provides some neutron shielding to other reactor components. The secondary function of the SCR is to provide a heat transport path out of the core to the decay heat removal system in the event of an emergency. This also is accomplished by the material properties, namely the high thermal conductivity of BeO.

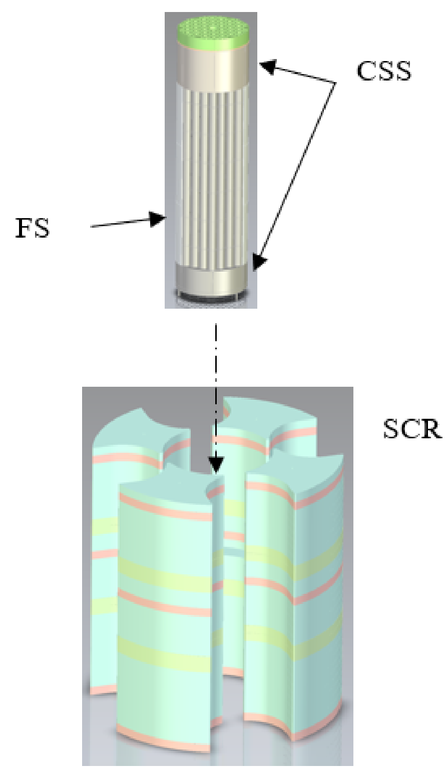

Figure 18. Visual of SCR subsystem.

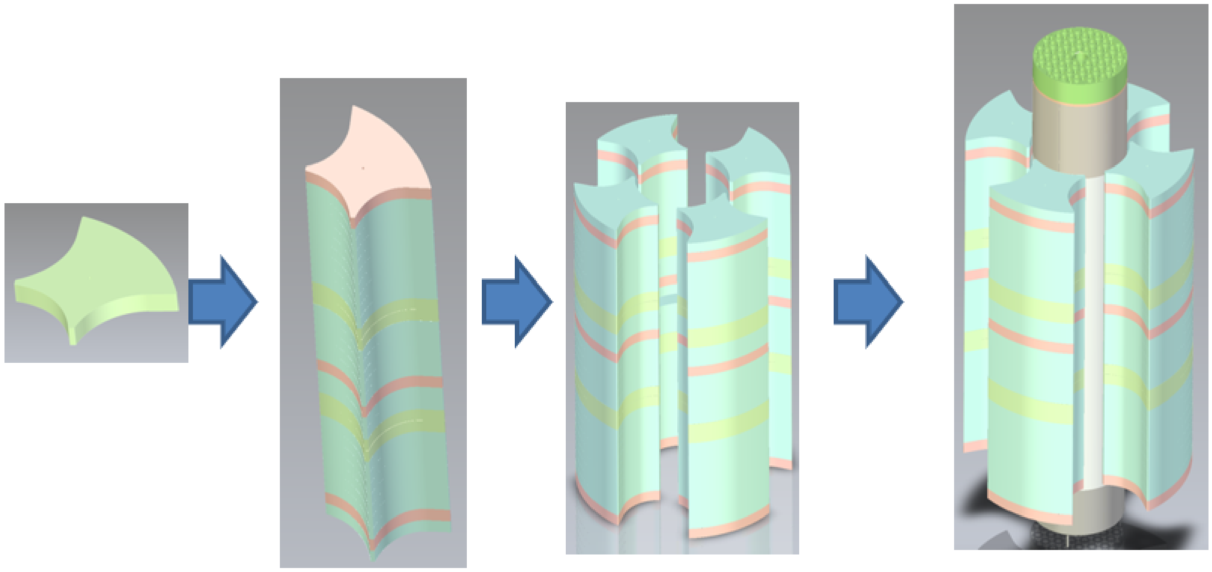

Figure 19. Layout of BeO plates in SCR up through location of SCR in the FCS.

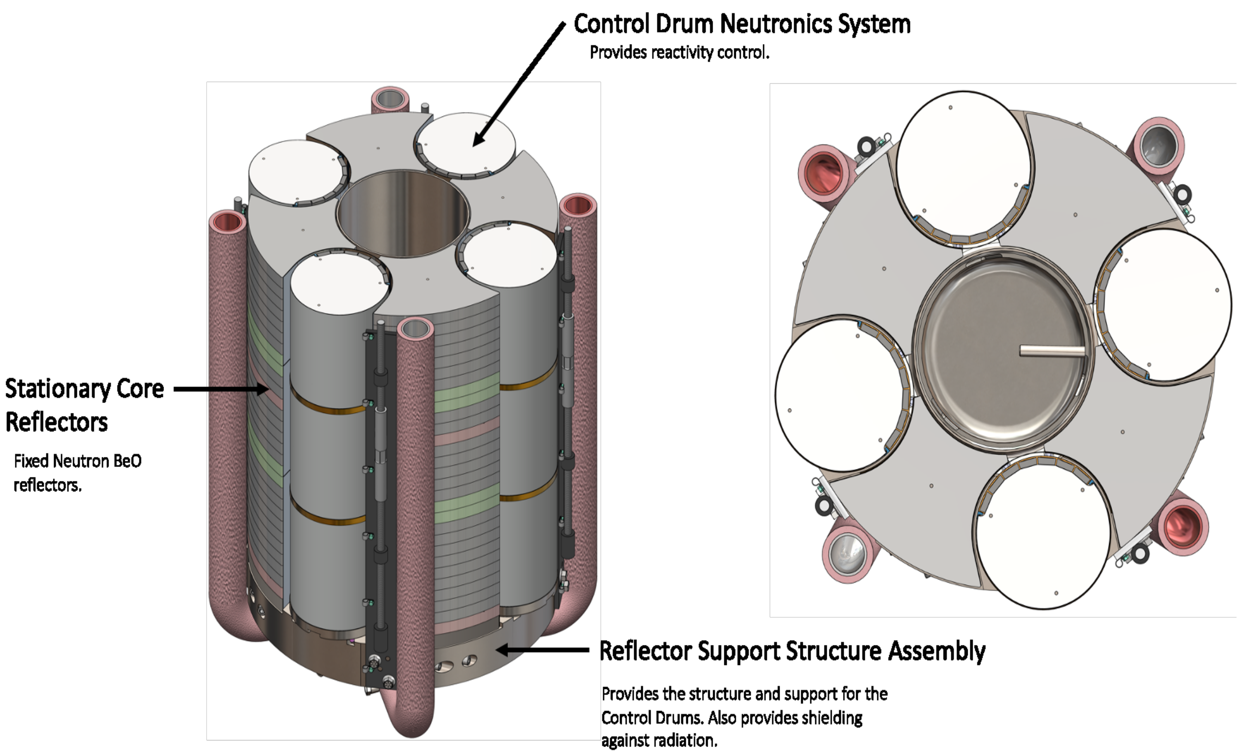

Figure 20. SCR location in relationship to the control drums.

3.3.4 Design Evaluation

3.3.4.1 Fuel Evaluation. INL/RPT-22-6855 provides the detailed analysis of U-ZrH fuel performance calculations and simulations under MARVEL irradiation conditions. The report provides a comprehensive survey of the known thermophysical properties, performance, and quantitative relationships associated with the MARVEL reactor fuel rod and to use this information to determine its mechanical integrity during the most extreme accident scenario predicted for the reactor. The analysis conclusions are briefly summarized here.

As stated previously, MARVEL will utilize the well-known steel-clad U-ZrH fuel element purchased from TRIGA International. Qualifying or licensing this fuel system is not an objective of the MARVEL design effort. This fuel system is already licensed by the United States Nuclear Regulatory Commission (NRC) in training, research, isotopes, General Atomics" (TRIGA) reactors and will be used as a basis for MARVEL fuel elements (rods). INL/RPT-22-6855 provides a general historical description of the uses and fabrication of U-ZrH fuel in TRIGA reactors and in the SNAP-10A Reactor.

INL/RPT-22-6855 discusses in detail the standard cast-and-hydride fabrication process of TRIGA fuel and the resulting microstructures and thermophysical and mechanical properties of the MARVEL fuel. INL/RPT-22-6855 also discusses in detail cladding composition and microstructure, and the thermophysical and mechanical properties of the MARVEL fuel cladding.

Even though the NaK coolant could only contact the fuel meat if catastrophic failure and rupture of the cladding occurred, compatibility of U-ZrH with NaK coolant is still important and was first established in the 1950s as part of the SNAP program. During these experiments, the compatibility of the fuel with liquid NaK was determined using two different techniques: (1) heating a U-ZrH specimen in a NaK-filled capsule of 304 SS for 120 hours at a constant high temperature, and (2) intermittently washing NaK of a given temperature over the hydrided specimen which was maintained at a temperature 150°C to 370°C lower than the NaK. Corrosion of the hydrided samples were determined from (1) dimensional changes in the specimens and (2) inspection of the surfaces for changes in color and appearance of pre-machined grooves. Both experiments revealed essentially no observable changes in δ-phase zirconium hydride specimens exposed to NaK below 1000°F (~ 538°C). Similarly, metallic uranium does not experience appreciable corrosive attack from liquid sodium up to 500°C. Above this temperature, weight loss measurements performed on uranium in flowing NaK reveal dissolution rates on the order of a few mils per month. However, the 304SS cladding seemed compatible with Na under 550°C.

There are several noteworthy features of the MARVEL fuel's performance in the analysis in INL/RPT-22-6855. Unlike LWR cladding, radiation-enhanced creep is very small because MARVEL coolant is not highly pressurized, and the small amount of radiation-enhanced creep in the cladding is radially outward rather than inward because the pressure inside the cladding is higher than the pressure outside. This means that radiation-enhanced creep of the cladding causes radial expansion rather than contraction during the fuel's operational cycle. Also, the peak burnup at end of life (EOL) for MARVEL fuel is extremely small (about 2.5 $MWd \cdot kgU^{-1}$) compared to standard LWR fuel (where burnup values up to 50 $MWd \cdot kgU^{-1}$ are typical). Consequently, the contribution to the internal fuel pin pressure from fission gas release is small for MARVEL fuel.

The analysis in INL/RPT-22-6855 also indicates that the MARVEL fuel rod maintains its structural integrity during the most extreme accident scenario predicted for the MARVEL reactor during which the maximum hoop stress generated in the cladding reaches only 8.21 MPa, far less than the predicted yield strength of 304 SS under accident scenario conditions (about 189 MPa). The analysis suggests that the fuel temperature limit of about 950°C is accurate in order to avoid internal gas over pressurization and cladding damage, which is far beyond the maximum temperature predicted in MARVEL fuel of about 680°C.

The hoop stress on the cladding under the most extreme prolonged beyond design basis accident (BDBA) which the most extreme temperatures are expected to result from a prompt insertion of 0.4$ with no control drum actuation, is significantly less than the yield stress of 304 SS under these conditions. The anticipated peak fuel meat temperature during the BDBA is 952 K (about 679°C). This temperature is low enough that the fuel meat void- and radiation-induced swelling does not generate fuel-cladding mechanical interactions at the low level of burnup of MARVEL fuel. This analysis suggests that the fuel temperature limit of about 1223 K (950°C) is accurate in order to avoid internal gas over pressurization (which is far beyond the maximum temperature predicted in MARVEL fuel).

During operation, the radial expansion of the cladding exceeds the radial expansion of the fuel meat, prohibiting fuel-cladding mechanical interactions (FCMI) even if the fuel meat and cladding were in intimate contact in their room temperature as-fabricated state. Under the most conservative assumptions, the resulting FCMI hoop stress is only about 5.6 MPa which is less than the internal gas pressure hoop stress experienced during the high temperature BDBA scenario.

Uranium and zirconium both form eutectics with iron, nickel, and chromium, which are the primary constituents of the 304 SS cladding. Eutectic melting in 304-clad U-ZrH elements has not been observed to occur below about 1050°C, and the effects of fuel-cladding chemical interactions (FCCI) under anticipated MARVEL normal and transient conditions are negligible. [References: [1] D. D. Keiser, *et al.,* "High temperature Chemical Compatibility of As-Fabricated TRIGA Fuel and Type 304 Stainless Steel Cladding", INL/EXT-12-27153, 2012, [2] D. Keiser. Jr., J.-F. Jue, F. Rice, E. Woolstenhulme, Post irradiation examination of a uranium-zirconium hydride TRIGA fuel element, Front Energy Res 11 (2023) 12. [3] Keiser Jr, Dennis D., Rice, Francine J., Sell, David A., Jue, Jan-Fong, Forsmann, Bryan L., and Woolstenhulme, Eric C.. An Investigation of Liquefaction in Irradiated TRIGA Fuel Exposed to Relatively High Temperatures. United States: N. p., 2020. Web. doi:10.2172/1737565.]

Based on the known behavior of the MARVEL fuel element, its anticipated conditions during the worst case BDBA, and the analysis thereof, the MARVEL fuel element's structural integrity remains stable and predicted before, during, and after the transient.

3.3.4.2 Neutronics Evaluation. ECAR-6099, "Neutronics Analyses for the MARVEL Preliminary Documented Safety Analysis,"[21] provides details of the neutronic analysis methodologies and results performed for the MARVEL core design. The calculations were performed using the Monte Carlo N-Particle (MCNP) transport code, version 6.2 on Idaho National Laboratory's Sawtooth supercomputer.

MCNP 6.2 was used in the design process and selection of the MARVEL core design (fuel, neutron reflectors, control elements, etc.) As stated earlier, MARVEL utilizes U-ZrH fuel with 30 wt. percent uranium with a nominal U-235 enrichment of 19.75% and nominal hydrogen to zirconium ratio of 1.6. There are metallic beryllium and BeO neutron reflectors used in the core design. There are four control drums consisting of BeO and B4C neutron poison plates. Plots of the MCNP model geometry of the radial and axial slice of the reactor core are shown in Figure 21 and Figure 22 respectively.

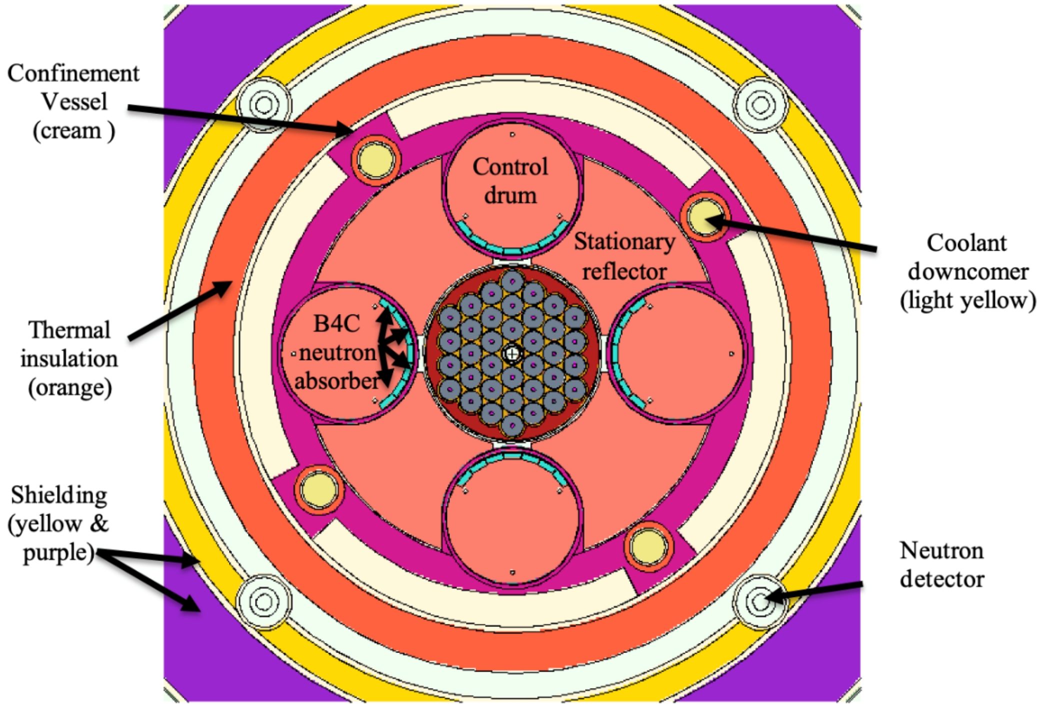

Figure 21. Radial view of MCNP model of MARVEL core.

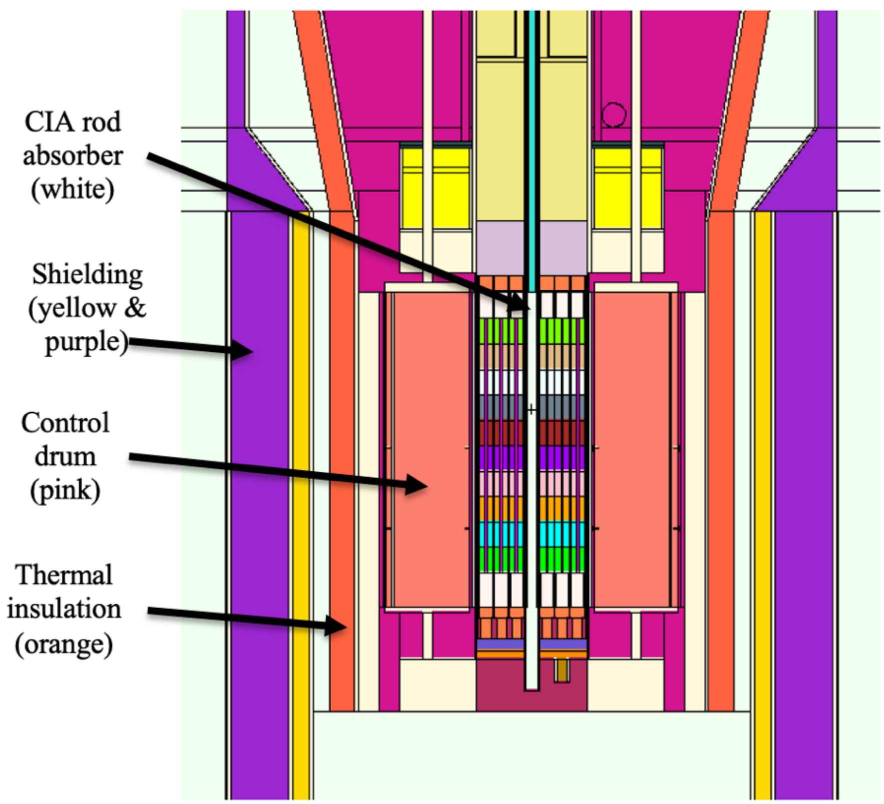

Figure 22. Axial view of MCNP model of MARVEL core.

The fuel rod power peaking factors under normal and transient conditions have been analyzed for the 2D radial, 2D axial, and 3D peaking. The maximum 2D radial an axial peaking factors are 1. 046 and 1.29, respectively. The maximum 3D peaking factor is calculated to be 1.64 on the outside of a corner rod adjacent to the neutron reflectors. The power distribution in the core during normal operation is slightly bottom peaked yielding an axial offset of -1.1%. The maximum 2D peaking factor during an reactivity insertion accident with one control drum withdrawn to the hardstop is 1.055. This data feeds ECAR-6332, "RELAP5-3D Thermal-Hydraulic Analysis of Marvel Microreactor - Final Design,"[22]

An MCNP analysis of the reactivity effects of temperature on fuel, coolant, and neutron reflectors was performed to characterize the reactivity coefficients and to ensure that the net temperature reactivity coefficient is negative for the core system. A negative net temperature reactivity coefficient ensures that the reactor will enter into a positive power feedback loop that would make the reactor unstable and at high risk for core melting in an accident scenario. The U-ZrH fuel has a large negative temperature reactivity coefficient of -5.22 pcm/K. The BeO, Be, and NaK all have relatively small positive temperature reactivity coefficients of 1.26, 0.3, 0.16 pcm/K respectively. The net temperature reactivity coefficient is -3.84 pcm/K, which ensures stable reactor operations. This data feeds ECAR-6332.

An analysis of the depletion of the core and resulting loss of reactivity was performed to ensure the design could meet the power and lifetime requirements. There is a loss of 954 pcm over 2 effective full power years at 85 kw_{th}. The beginning of life excess reactivity of about 3300 pcm easily allows for the reactivity loss due to depletion to be absorbed with additional excess reactivity at the end of life to ensure operation through the design life.

The reactivity worth of the control drums as function of angle is calculated to ensure sufficient shutdown margin and to characterize control drum effects on the core. All four control drums have a combined reactivity worth of 13727 pcm, or $18.30. This provides a shutdown margin of $10.10 subcritical at hot conditions and $6.54 subcritical in cold conditions with the highest value control drum stuck out for both cases. This calculation shows that sufficient shutdown margin is guaranteed in all design basis scenarios. Additionally, the reactivity worth of the central insurance absorber (CIA) rod as a function of withdrawn length is calculated. The hot full power reactivity worth of the CIA is $3.05. This data feeds ECAR-6332.

The reactor kinetics parameters are characterized. This data characterizes the time-dependent behavior of the reactor. The beta-effective is calculated to 0.00746 ± 0.00004. The neutron generation time is 17.99 ±0.03 microseconds. This analysis provides data for ECAR-6332.

The irradiation damage doses to the steel fuel cladding and the steel reactor vessel in terms of displacements per atom (DPA) are analyzed. The peak cladding dose is 0.098 DPA over the reactor lifetime. The reactor vessel peak 3D damage dose is 0.063 DPA with an average of 0.0519 DPA over 2 effective full power years. The relative materials property changes due to irradiation were analyzed for the 0.2% yield strength, the ultimate tensile strength minus the yield strength, the uniform elongation, and the total elongation minus the uniform elongation. The irradiation effects give a 12.5% increase, 1.9% decrease, 2.5% decrease and 1.7% increase of the aforementioned properties, respectively.

The irradiation effects on the BeO neutron reflector is analyzed to ensure material degradation is not significant and will not affect the safety function of the control drums. A limit for fast fluence for the BeO of 2.0x1020 nvt was identified from literature for the MARVEL system. This is a conservative limit. The analysis yields a peak lifetime fast fluence of 6.25x1019 nvt. This result is within the limit and yields a calculated volumetric expansion of the BeO of 0.19% with no significant crack formation, which is acceptable. The neutron and photon heating of the BeO is also analyzed. The 3D heating data is provided to ECAR-7228, "MARVEL Control Drum Actuator Stress Analysis,"[23] for consideration in a control drum bowing analysis. The total combined neutron and photon heating of a single control drum is 295 watts. The total neutron and photon heating in the BeO is 2.40 kW with 1.22 kW in the stationary BeO reflectors and 1.18 kW in all the control drum BeO.

3.3.4.3 CSS Evaluation. Since the fuel rods and other items in the FCS rest on the lower grid plate, the scope of this evaluation is to demonstrate by static analysis the structural integrity for normal operating conditions and worst-case accident scenarios applied to the lower grid plate then compare the results of the structural analyses to ASME BPVC Section III Division 5, Subsection HG.

The analyses contained in ECAR-6585, "Core Support Structure Analysis ,"[24] includes a static structural analysis of the lower grid and seismic analysis. The results of the seismic analysis are discussed in Chapter 3, Section 3.4.2.3. The results of the static analyses are summarized in this section. Since the MARVEL Reactor is more than 800°F and directly supports the nuclear fuel pins, per ASME BPVC Section III Division 5, this system falls under the Class SM Metallic Core Support and under Elevated Temperature Service. All allowable stresses are determined and directed from this section of the code.

A temperature of 1066°F (630°C) was utilized for performing the FEA. Limits from ASME Service Class A-D utilized 1040°F (560°C) for normal operating temperature and 1066°F (630°C) for max temperature during worst case accident transients. The total weight of the nuclear fuel pins, beryllium reflectors, reflector spacers, and upper grid plate assembly is approximately 602 lbs. (612 lbs. minus 10.4 lbs. for lower grid plate, rounded up). The structural static load on the Lower Grid Plate in the analyses was increased to 650 lbf for uncertainty in fabrication and miscellaneous weight increases.

The MARVEL reactor will be in operation for approximately two years and be critical only on the weekends (Friday through Sunday). ASME BPVC takes under consideration the time the core support is under a load. For the FEA and ASME compliance contained herein, it has been assumed for conservatism that the lower grid plate will be at operating temperature for two full years (approximately 1.75×10^4 hrs.) at full power (no ramp up and ramp down time) and not three sevenths of two years (for operating only on weekends).

A summary of the ASME BPVC analysis for the CSS is shown in Table 6. All demand-to-capacities, per ASME BPVC Section III Division 5 Subsection HG, are less than one (1) and are satisfactory. Therefore, the lower grid plate will safely support the fuel as during normal operations. A full description of the analysis and results can be found in ECAR-6585.

Table 6. Grid Plate demand to capacity results summary.

Service Loadings	Demand-to-Capacity Description	Demand (ksi)	Capacity (ksi)	Demand-to-Capacity	Reference*
N/A	Maximum Allowable Membrane Stress Intensity	0.87	9.03	0.10	HGB-3222 (a) (1)
N/A	Allowable Combined Membrane and Bending Stress Intensity	3.06	13.55	0.23	HGB-3222 (b) (2)
A/B	Maximum Primary Membrane Stress Intensity for Level A & B Service Loadings	0.87	13.73	0.06	HGB-3223 (a) (3)
A/B	Allowable Combined Primary Membrane and Bending Stress Intensities 1	3.06	30.20	0.10	HGB-3223 (c) (4)
A/B	Allowable Combined Primary Membrane and Bending Stress Intensities 2	2.33	16.20	0.14	HGB-3223 (c) (5)
C	Allowable Primary Membrane Stress Intensities 1	0.87	18.12	0.05	HGB-3224 (a)
C	Allowable Primary Membrane Stress Intensities 2	0.87	16.24	0.05	HGB-3224 (a)
C	Use-Fraction 1 for Level C Service Loading	0.02	1.00	0.02	HGB-3224 (b)
C	Allowable Combined Primary Membrane and Bending Stress Intensities 1 for Level C Service Loadings	3.06	27.20	0.11	HGB-3224 (c)(9)
C	Allowable Combined Primary Membrane and Bending Stress Intensities 2 for Level C Service Loadings	2.33	16.20	0.14	HGB-3224 (c)(10)
C	Use-Fraction 2 for Level C Service Loading (unitless)	0.02	1.00	0.02	HGB-3224 (d)
*References from ASME BPVC Section III Division 5, Subsection HB.					

3.3.4.4 SCR Evaluation. The SCR functions are primarily neutronic in nature with secondary thermal requirements. The SCR acts to reflect neutrons back into the core to decrease loss of neutrons from leakage. By the action of reflecting neutrons back into the core, the SCR also provides some neutron shielding to other reactor components. The secondary function of the SCR is to provide a heat transport path out of the core to the decay heat removal system in the event of an emergency.

The evaluation of the reflectors in performing the neutron reflection functions are provided by virtue of the adequacy of the core neutronics design in ECAR-6099 and in ECAR-6332.

The reflectors will be purchased from a commercial vendor. A commercial grade dedication (CGD) plan in accordance with NQA-1 has been created for the reflectors (CGI-1224). Additionally, a procurement specification has also been created, SPC-3178.

3.3.4.5 Evaluation of Compliance with Design Criteria. Consistent with PDC-1, FCS SSCs designated as SR and NSR-AR are treated with an appropriate graded approach. Generally recognized codes and standards are used and documented MARVEL design, construction and operation complies with applicable quality requirements as outlined in PDD-13000.

Consistent with PDC-2, FCS SSCs designated as SR and NSR-AR are designed to withstand the effects of SDS-2 seismic events without the loss of the capability to perform their safety functions.

Consistent with PDC-4, FCS SSCs designated as SR and NSR-AR are designed to withstand the environmental conditions associated in operating in a NaK environment during normal plant operation as well as during postulated events as a result of equipment failures.

There are no specific FCS design features required for meeting the SSC sharing requirements in PDC-5.

Consistent with PDC-10, FCS SSCs are designed to ensure heat is effectively transferred to the primary coolant.

Consistent with PDC-11, the FS is 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.

There are no specific design features required for the FCS for meeting the power oscillation requirements in PDC-12.

Consistent with PDC-16, the FCS fuel matrix and cladding provide multiple barriers against the release of fission products. The adequacy of meeting this Criterion is demonstrated in Section 4.4 and Addendum Chapter 15 in providing an effective functional confinement against the release of fission products during normal operations and accident conditions.

Consistent with PDC-26, the FCS provides negative reactivity feedback as the temperature of the reactor increases assure that shutdown margin is maintained.

Consistent with PDC-34, the FCS provides for passive core flow/heat removal from the reactor core at a rate such that fuel and vessel temperature limits and the design conditions of the PCB are not exceeded.

Consistent with PDC-62, the FCS provides for inadvertent criticality prevention during fuel storage and handling.

3.4 MARVEL Reactor Structure

3.4.1 Design Description

This section describes the MRS design. Figure 23, as taken from TFR-2576, shows the physical boundaries of the subsystems within the MRS. Functions highlighted in green are those that are SR per ECAR-6440 and listed in Appendix B. Functions highlighted in yellow are those that are Non-Safety-Related with Augmented Requirements (NSR-AR). Other Non-Safety Related (NSR) functions are in light blue. The classification of these functions is derived in ECAR-6440 and listed in Appendix B. The figure also shows MRS interfaces with the FCS and T-REXC SSCs.

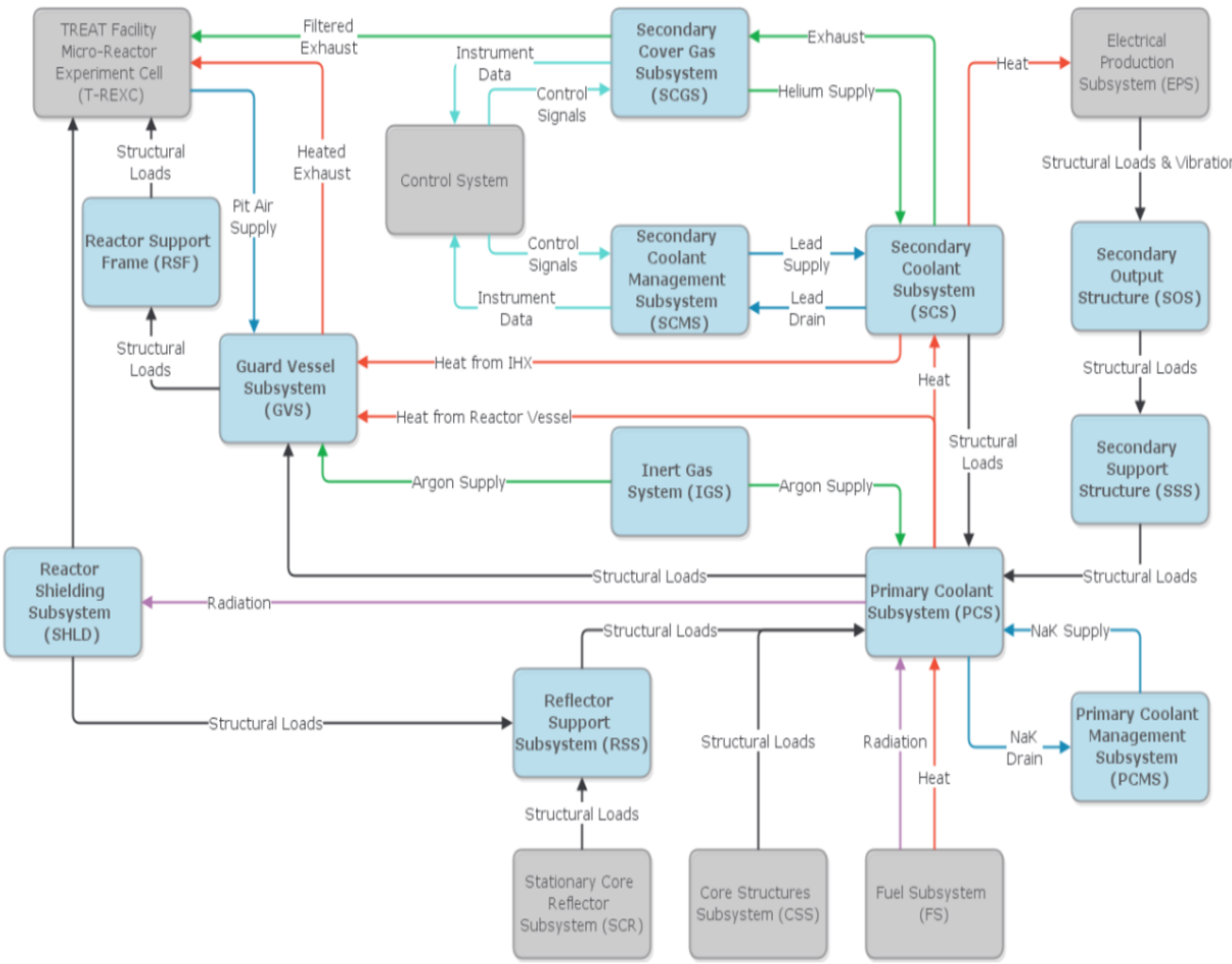

Figure 23. MRS subsystems and structures.

The MRS is a structural system that provides the necessary support for all other subsystems. The MRS is composed of a variety of subsystems: power generation and conversion, coolant, gas, shielding, housing, and structural support. The MRS provides and/or facilitates cooling, radiation shielding, and system physical protection. The MRS also contains an inert gas system to mitigate the exposure of the fuel pins in the event of a leak. Figure 24 shows the physical boundaries of the subsystems within the MRS described in this section.

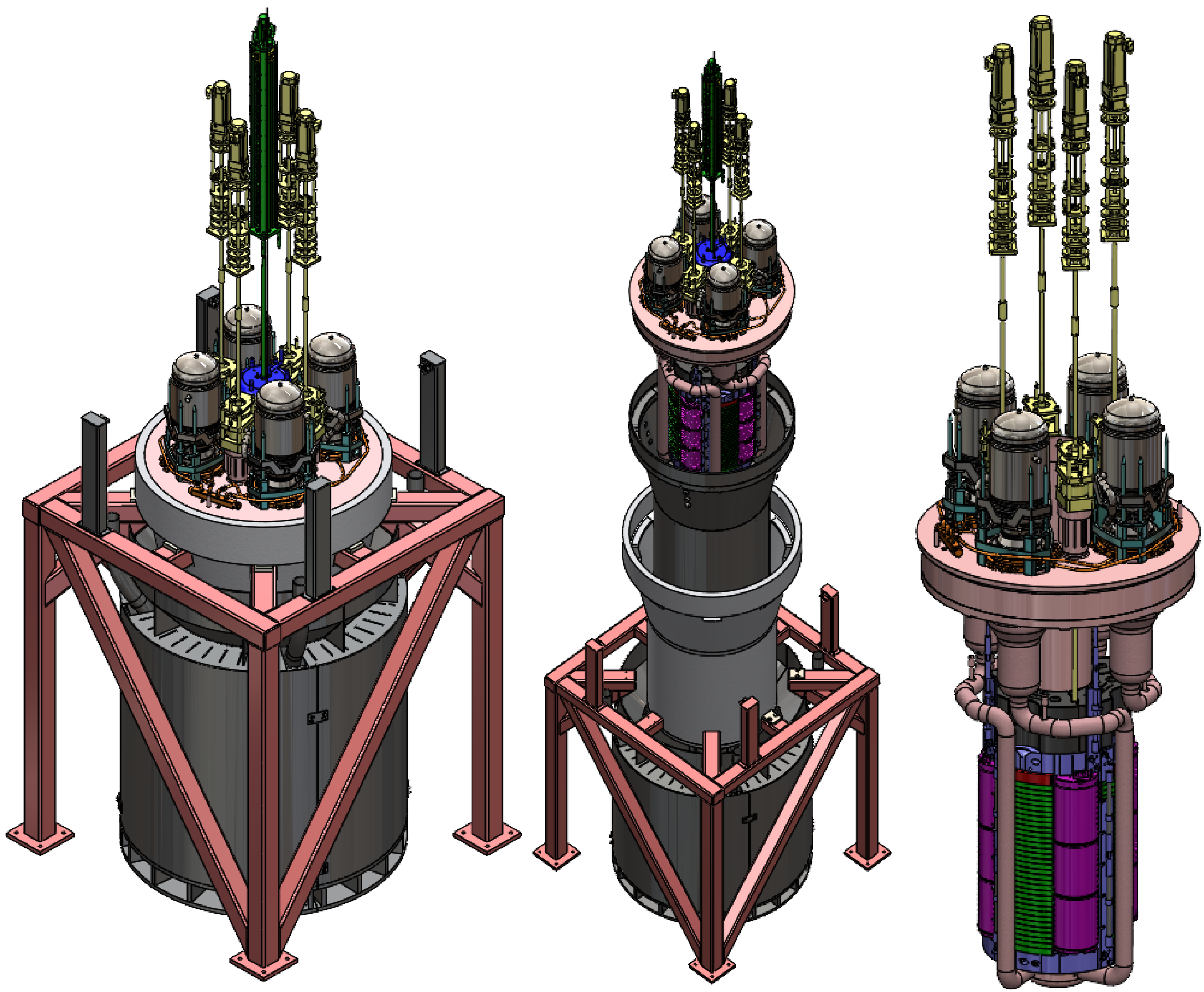

Figure 24. Reactor structure assembly.

3.4.1.1 Primary Coolant Subsystem (PCS) Overview. The PCS is a high temperature, low pressure boundary that leverages eutectic NaK alloy as a coolant to remove heat from the Fuel Subsystem (FS). During normal operation this heat is transferred to the secondary coolant for power generation and during postulated accident events the PCS conducts decay heat to the guard vessel for convection to T-REXC. This path of heat exchange also occurs during normal operation.

The NaK primary coolant acts as a radionuclide barrier by retaining fission products by plate-out, chemical solubility, and/or adsorption mechanisms. Fission heat is removed from the core by the primary coolant, which is driven by natural convection. The hot coolant rises out of the core into the distribution plenum. The secondary coolant system (SCS) in the distribution plenum removes heat from the primary coolant, and buoyancy forces drive the cooled NaK downward into one of the four insulated PCS downcomer pipes. The piping returns the coolant to the lower plenum where it will be again pulled into the core, completing the loop.

Structurally, the PCS is seated on the stand at the distribution plenum and supports the weight of most MRS subsystems except the guard vessel. The hanging arrangement sets the neutral plane of thermal expansion to accommodate unrestrained expansion at temperature. The PCS includes an argon gas headspace above the primary coolant level supplied by the IGS. Coolant makeup and draining functions are provided by the PCMS.

3.4.1.2 Secondary Coolant Subsystem (SCS) Overview. The SCS removes fission heat from the PCS, transferring the heat through an IHX to the engines through a natural convection loop for power generation. The secondary coolant fills the IHX to an opening at the top where the engine is suspended partially in the coolant. The design follows a pipe-in-pipe arrangement, where the secondary coolant in the outer annulus pulls heat from the primary coolant, is driven up by buoyancy forces to the engine, and sinks through the center downcomer pipe once the heat is removed by the engine. GaInSn filling and draining is provided by the interfacing SCMS. The SCS is filled with an inert argon gas blanket to limit interaction of oxygen from the air with the secondary coolant.

3.4.1.3 Primary Coolant Management Subsystem (PCMS) Overview. The PCMS provides the equipment necessary to purify, fill, and drain the primary coolant.

3.4.1.4 Secondary Coolant Management Subsystem (SCMS) Overview. The SCMS provides the equipment necessary to purify, fill, and drain the secondary coolant and will be procured as a completed system from an engineering vendor, and thus its design is outside the scope of this report.

3.4.1.5 Inert Gas Subsystem (IGS) Overview. The IGS supplies ultra-high purity (UHP) argon to the primary coolant boundary and the guard vessel to prevent NaK reactions with air.

3.4.1.6 Secondary Cover Gas Subsystem (SCGS) Overview. The SCGS provides a helium cover gas to the IHX for chemistry control of the secondary coolant, vents the helium cover gas from the IHX headspace, and filters the exhaust stream to contain any volatile particulates. This offgas is transferred to the T-REXC ventilation system for exhaust.

3.4.1.7 Guard Vessel Subsystem (GVS) Overview. The GVS serves as the secondary containment boundary for the primary coolant and envelops the PCS. The GVS is composed of two primary structures: the guard vessel (GV) and the upper confinement structure (UCS). Heat during normal operations, anticipated events, and postulated accident scenarios is conducted and convected from the PCS into the secondary confinement space enclosed by the GV and eventually passively transferred to the T-REXC space. The GV supports the reactor by transferring the weight of the MRS subsystems to the RSF. It also contains features that allows for cables and wiring to be passed through the pressurized inert gas boundary.

In the event of a loss of coolant accident (LOCA), the guard vessel prevents the core from being uncovered by providing a controlled, inert environment for the coolant to flow into. If primary coolant leaks into the guard vessel, the fluid level in the guard vessel will rise as liquid level in the primary subsystem falls, until both subsystems equilibrate.

Finally, the guard vessel has sloped, dished heads to ensure any NaK leaked drains down into the sump at the bottom of the vessel. This helps to make leak detection easier and allows D&D to use a central low-point to drain the NaK after a postulated LOCA.

3.4.1.8 Reactor Support Frame (RSF) Overview. The RSF transfers the load from the GVS (and therefore all the MRS subsystems beside the lower radial shield) to the T-REXC structure.

3.4.1.9 Reactor Shielding Subsystem (SHLD) Overview. The SHLD subsystem consists of both axial and radial shielding components. Axial shields are located on top of the upper RSS plate to reduce gamma and neutron radiation from the core into the upper zone of the MRS. Radial shields surround the lower portion of the GVS to reduce gamma and neutron activation of the T-REXC structures.

3.4.1.10 Reflector Support Subsystem (RSS) Overview. The RSS supports and locates the Stationary Core Reflector Subsystem (SCR) and provide vertical support for the control drums. The RSS is suspended from the PCS distribution block by bolted structural straps which utilize a turnbuckle to allow for adjustment and leveling of the support plates. The stainless-steel support plates are each formed from four quarter circle segments which are bolted together around the lower core barrel.

3.4.1.11 Secondary Support Structure (SSS) Overview. Four SSS assemblies are connected to the top the PCS distribution block, each supporting a SOS structure.

3.4.1.12 Secondary Output Structure (SOS) Overview. The SOS suspends the Stirling engines in the pool of secondary coolant. The SOS is also responsible for isolating the vibration caused by the operating engines to minimize impacts on the rest of the MRS. Each SOS assembly is designed to be assembled outside the pit and then lowered via lifting eye onto the SSS.

3.4.2 Design Bases

The MRS SSC safety classifications are provided in Appendix B. Key safety functions of the MRS SSCs are to:

- Provide structural, mechanical, and geographic spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible.
- Provide design provisions to ensure major core flow blockages are not credible.
- Maintain core coolable geometry in a seismic event.
- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and accident conditions.
- Confinement barrier to ensure primary NaK and any leaked fission or activation products remain within PCB and oxygen remains outside.
- Reduce probability of large NaK leaks due to pipe design under normal and transient operating conditions.

3.4.2.1 Design Criteria. In addition to the overall principal design criteria (PDC) 1-5 in Appendix A, the MRS SSCs shall be designed to meet the following PDC:

- The MRS shall facilitate the transfer of heat from the core region with appropriate margin to assure that specified acceptable fuel design limits are not exceeded [PDC-10].
- Inaccessible MRS SSCs shall be constructed of materials designed to withstand the credited environmental conditions (e.g., temperature, pressure, radiation, etc.) in which they are installed for at least 2 calendar years of operation without maintenance [PDC-14].
- The MRS shall facilitate the transfer of heat from the core region with appropriate margin to assure that specified acceptable fuel design limits are not exceeded. [PDC-15].

- The MRS shall provide barriers that prevent or mitigate the release of radioactive materials from the reactor to the public and the environment during normal operations, anticipated events, and postulated accident conditions [PDC-16].
- Inaccessible MRS SSCs shall be constructed of materials designed to withstand the credited environmental conditions (e.g., temperature, pressure, radiation, etc.) in which they are installed for at least 2 calendar years of operation without maintenance [PDC-31].
- The MRS shall facilitate decay heat removal in all operating scenarios, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded [PDC-34].
- The MRS shall facilitate decay heat removal in all operating scenarios, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded [PDC-44].
- Normally accessible MRS SSCs important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features [PDC-45].
- Normally accessible MRS SSCs important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features [PDC-46].
- The MRS shall include means to control the compliant release of radioactive materials in gaseous form such that the air emission limits do not exceed those established by T-REXC [PDC-60].
- The PCS is designed with sufficient margin to assure that (1) the design conditions of the intermediate coolant boundary are not exceeded during normal operations, including anticipated occupational occurrences, and (2) the integrity of the primary coolant boundary is maintained during postulated accidents [PDC-70].
- The MRS shall provide pure cover gas to locations separated from the primary coolant by a single passive barrier [PDC-70].
- The PCS is designed to meet the intent that the primary coolant is separated from chemically incompatible fluid by two redundant, passive barriers with one active and one passive barriers [PDC-78]. (See Appendix A)
- The PCS is designed to ensure that the primary coolant NaK design limits are not exceeded as a result of cover gas loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the primary coolant boundary [PDC-79].

3.4.2.2 Functional and Operating Requirements. The requirements and bases for the MRS are discussed in detail in FOR-868 and TFR-2576. Key system functional requirements are as follows:

- The PCS shall be capable of removing at least {93.5} kWth of heat from the core region during normal operations, anticipated events, and postulated accidents.
- The PCS shall remove heat from the core region such that the fuel cladding temperature remains beneath {650}°C during normal operation and anticipated events.
- The PCS shall remove heat from the core region such that the fuel cladding temperature remains beneath {704}°C throughout the duration of postulated accidents and beneath {788}°C for short-duration extremely unlikely postulated accidents.
- The SCS shall be capable of extracting at least {90} kWth of heat from the primary coolant during normal operating conditions.
- The guard vessel shall be capable of removing at least {1} kWth of heat from the PCS in all modes of operation.
- The GVS shall be capable of passively transferring at least {1} kWth of heat to the TREAT building atmosphere in all modes of operation.

- The IGS shall be capable of connecting to and transferring argon gas to the primary coolant pressure boundary, and guard vessel pressure boundary.
- The MRS shall be capable of withstanding design basis seismic events without damage to safety SSCs.

  Key system operational requirements are as follows:

- The PCS shall be designed to maintain heat loss to the guard vessel annulus beneath 10% of reactor full power.
- The PCMS shall provide a means to drain or remove the primary coolant from the reactor vessel.
- The SHLD shall prevent T-REXC concrete activation such that the dose rate is less than 0.5 mrem/hr at 30cm from the structure 90 days after shutdown of the reactor, and provide shielding to limit radiation exposure of instrumentation in the upper confinement space to less than {1,336} Rad/hr.

3.4.3 Description of Systems, Subsystems, and Major Components

3.4.3.1 Primary Coolant System (PCS). The PCS (Figure 25) is a high temperature, low pressure boundary that leverages NaK alloy as a coolant to remove heat from the Fuel Subsystem (FS). During normal operation this heat is transferred to the secondary coolant for power generation and during postulated accident events the PCS conducts decay heat to the guard vessel for convection to T-REXC. This path of heat exchange also occurs during normal operation.

The NaK primary coolant acts as a radionuclide barrier by retaining fission products by plate-out, chemical solubility, and/or adsorption mechanisms. Fission heat is removed from the core by the primary coolant, which is driven by natural convection (Figure 26). The hot coolant rises out of the core into the distribution plenum. The SCS in the distribution plenum removes heat from the primary coolant, and buoyancy forces drive the cooled NaK downward into one of the four insulated PCS downcomer pipes. The piping returns the coolant to the lower plenum where it will be again pulled into the core, completing the loop.

Structurally, the PCS is encased on the GVS, which is seated on the stand at the distribution plenum and supports the weight of most MRS subsystems. The hanging arrangement sets the neutral plane of thermal expansion to accommodate unrestrained expansion at various temperature. The PCS includes an argon gas headspace above the primary coolant level supplied by the IGS. Coolant makeup and draining functions are provided by the PCMS.

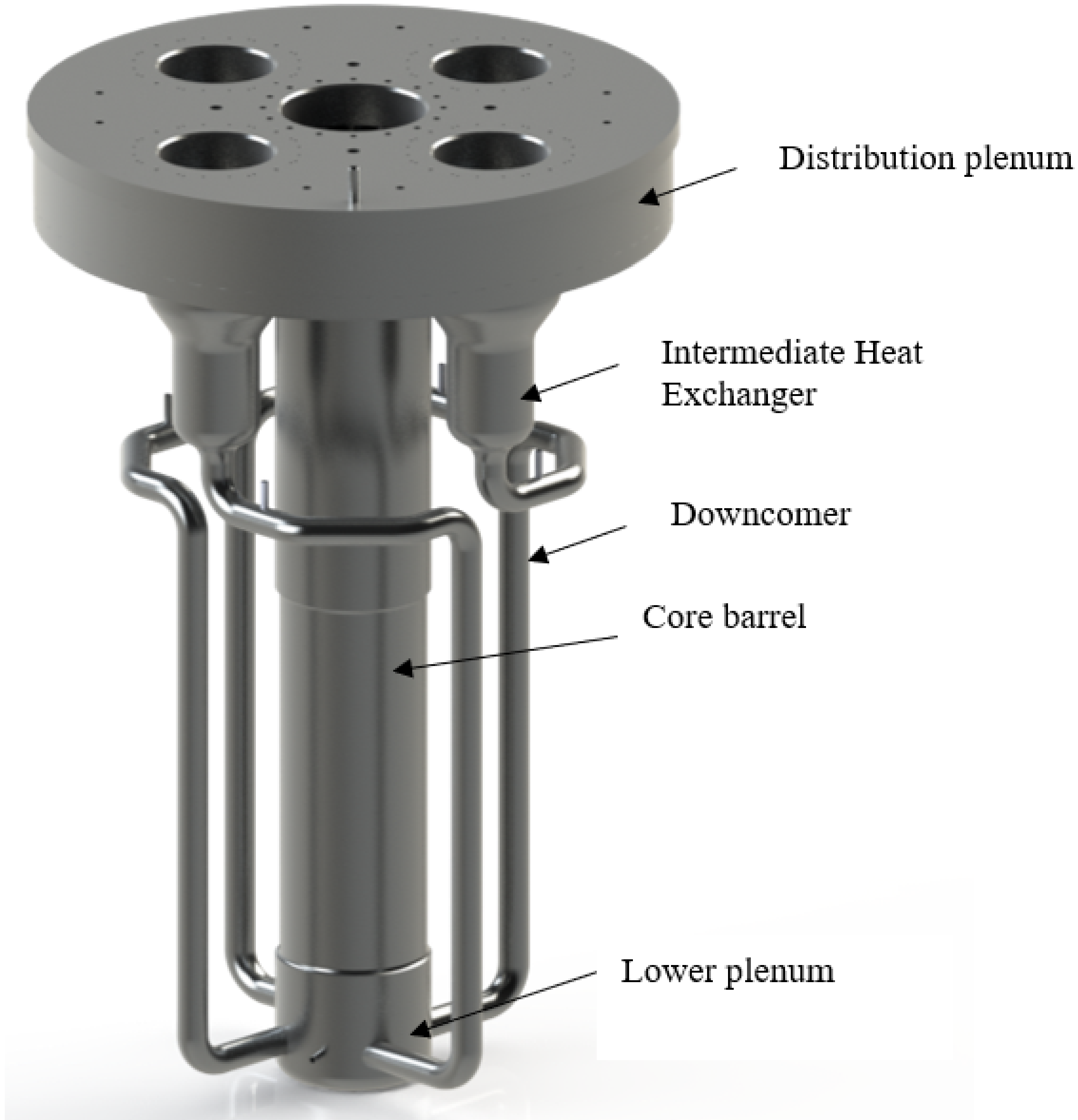

Figure 25. Illustration of the primary coolant system.

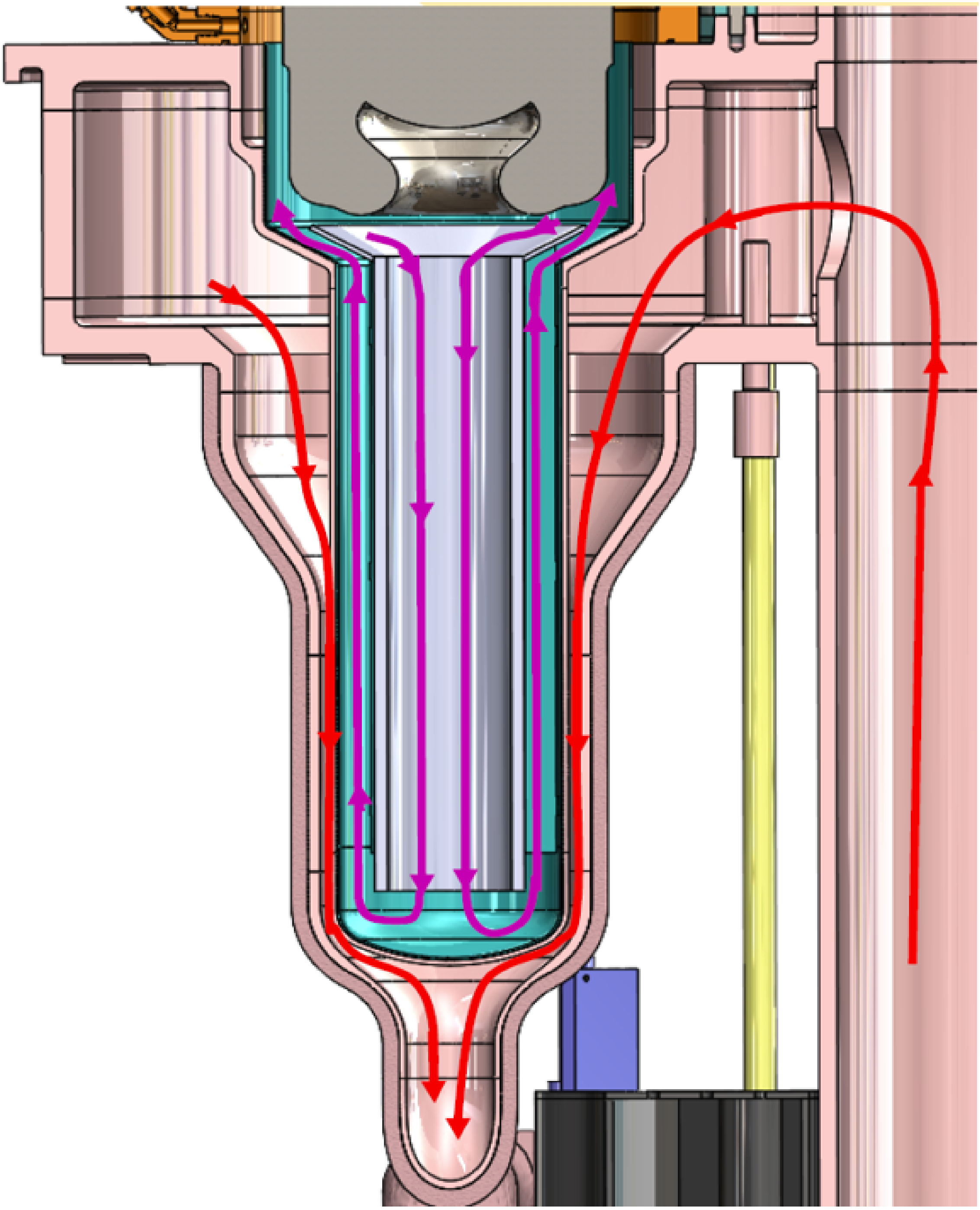

Figure 26. MARVEL PCS and SCS flow.

3.4.3.2 Guard Vessel Subsystem (GVS). The GVS shares a boundary with the IGS. They work together to prevent the core from becoming exposed during a leak.

The GVS (Figure 27) serves as the secondary containment boundary for the primary coolant and contains the primary coolant system. In addition, the guard vessel supports the reactor by transferring the loads from the reactor support frame to the primary coolant system. So, any weight held by the primary coolant system, which accounts for most of the reactor weight, must be supported by the guard vessel as well.

The guard vessel contains features that allows for cables and wiring to be passed through the pressurized inert gas boundary.

In the event of a Loss Of Coolant Accident (LOCA) the guard vessel prevents the core from being uncovered by providing a controlled, inert environment for the coolant to flow into. In order to do this, it is a pressure vessel able to contain an argon environment, as well as being form-fitting enough to restrict the amount of primary coolant required to fill past a certain height. Physically, it must have connections for purging atmosphere, recharging with argon, and monitoring the pressure inside the vessel. In addition, any wires and instrumentation required to monitor data within the guard vessel, including thermocouples and leak detectors, must pass through penetrations in the guard vessel and be sealed.

If primary coolant leaks into the guard vessel, the fluid level in the guard vessel will rise as liquid level in the primary system falls, until both systems equilibrate. The equilibrium process relates to the Inert Gas System, since the pressure in the IGS and pressure in the GVS both interact to determine how much the NaK level changes before stabilizing. Both systems must balance each other to ensure core coverage after a LOCA.

Finally, the guard vessel has sloped, dished heads to ensure any NaK leaked drains down into the sump at the bottom of the vessel. This helps to make leak detection easier and allows decontamination and decommissioning (D&D) to use a central low-point to drain the NaK after a postulated LOCA.

The guard vessel is analyzed and fabricated per ASME Section 3 Division 5, class B vessel code.

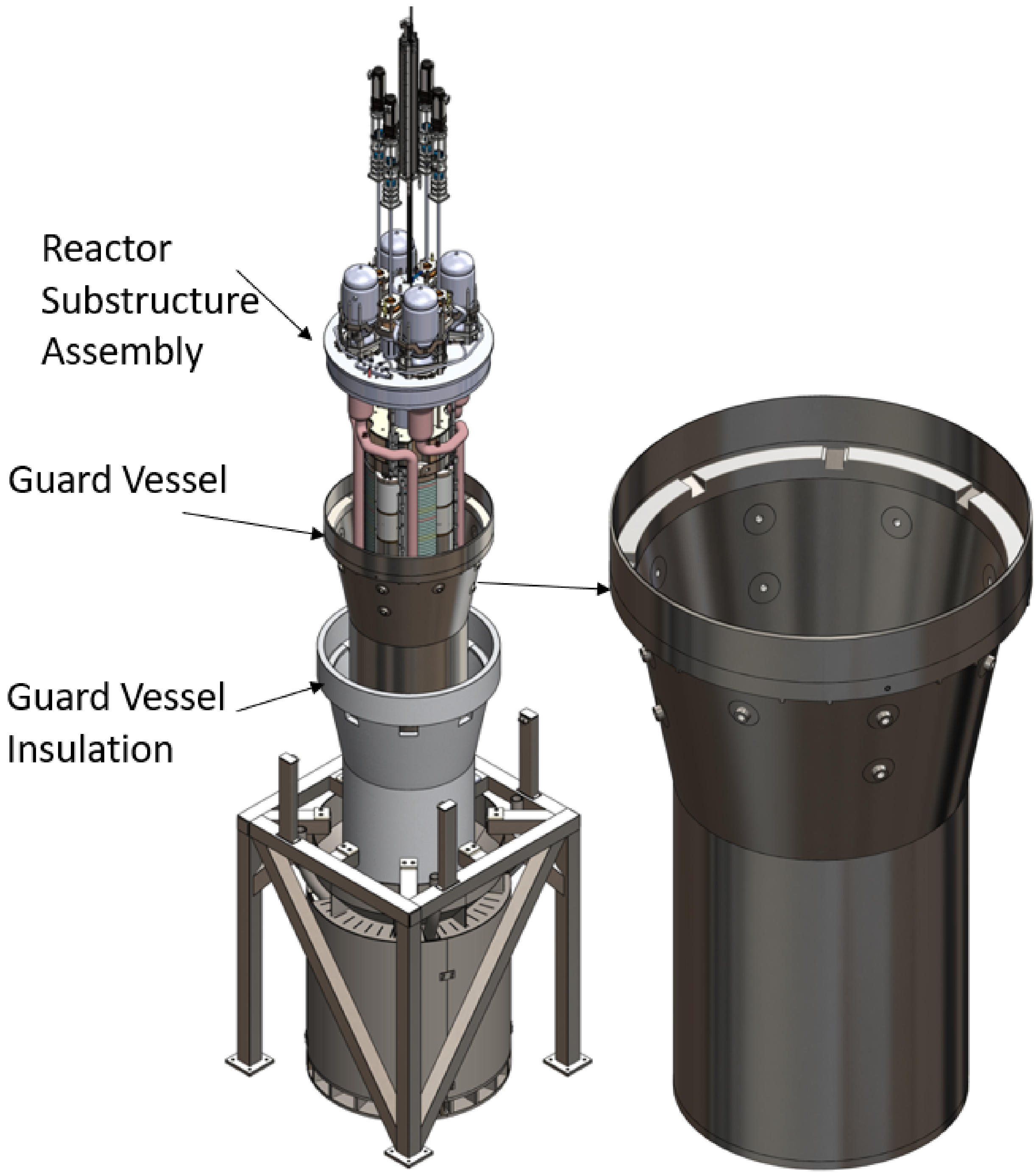

Figure 27. Illustration of GVS.

3.4.3.3 Secondary Coolant Subsystem (SCS) Overview. The SCS (Figure 28) removes fission heat from the PCS, transferring the heat through an IHX to the engines through a natural convection loop for power generation.

The secondary coolant is molten eGa-In-Sn at atmospheric pressure and fills the IHX to an opening at the top where the engine is suspended partially in the coolant. The design follows a pipe-in-pipe arrangement, where the eGa-In-Sn in the outer annulus pulls heat from the primary coolant, is driven up by buoyancy forces to the engine, and sinks through the center downcomer pipe once the heat is removed by the engine. Secondary coolant filling and draining is provided by the interfacing SCMS. The SCS is filled with an inert argon gas blanket as well to prevent reaction of any NaK which might leak across the IHX boundary from the PCS.

To prevent restrained thermal expansion between the hot PCS boundary and the cold central downcomer, the downcomer sits freely in a holder that restrains rotation and uplift. Vertical baffles prevent eddies in the flow behavior, while a conical diverter directs the eGa-In-Sn along the hot sidewall and prevents short circuits to the cooled eGa-In-Sn after passing through the SOS heat exchanger.

The SCS is filled with an inert argon blanket gas as well to prevent reaction of any NaK which might leak from the primary coolant system. To ensure that the SCS can accommodate the NaK leakage from a large loss of coolant accident without allowing the reactor core to be uncovered, the pressure of the argon gas in the SCS must be kept within 10 psig (speculative) of the pressure in the primary coolant system. To allow the SCS pressure to be maintained within this band, and active control system is provided which will monitor the pressures in the primary and SCS gas spaces. This system will adjust SCS pressure as required using an electronically controlled back-pressure regulator and an electronic supply flow controller.

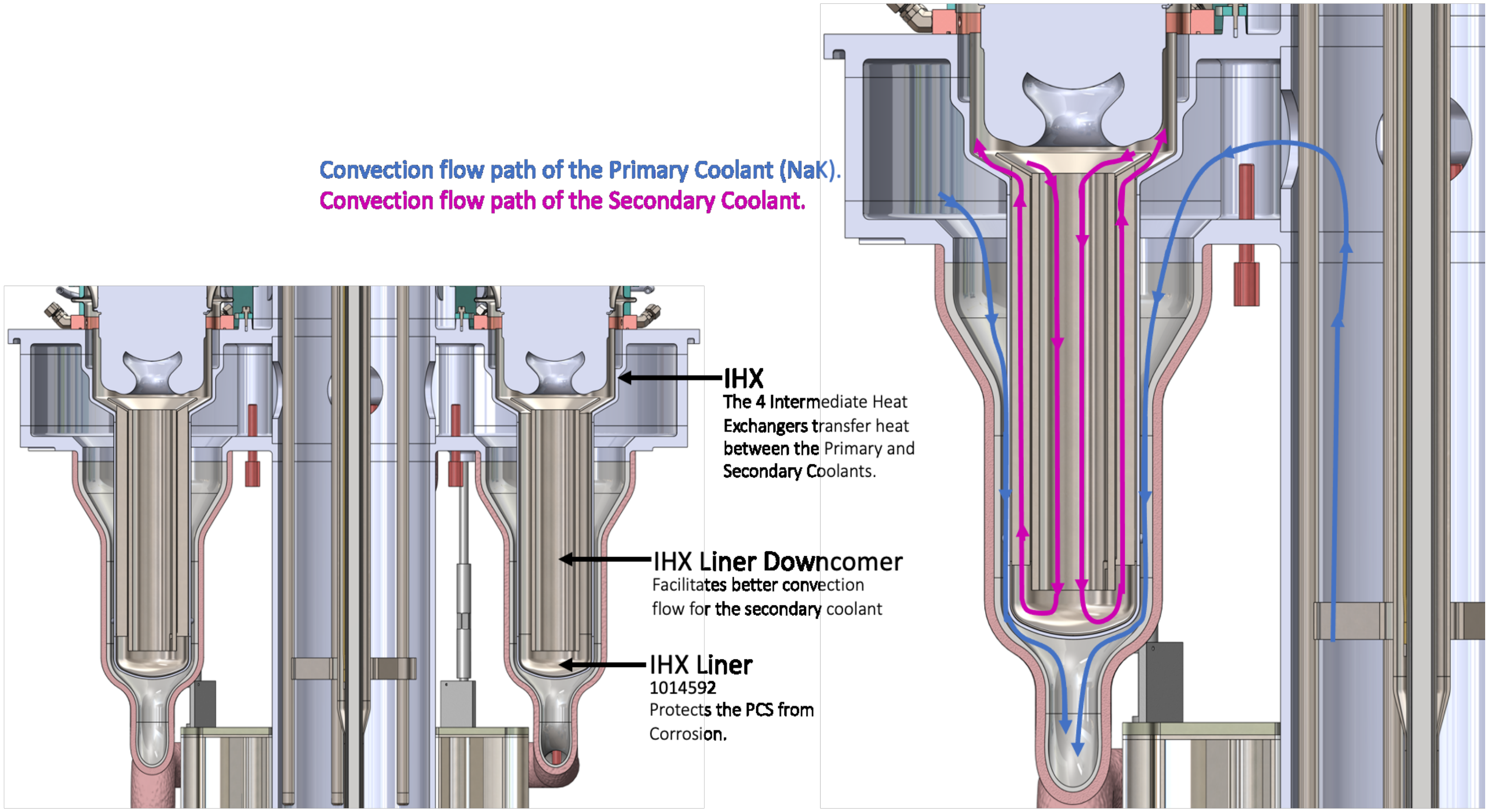

Figure 28. Design illustration of secondary coolant system.

3.4.3.4 Reflector Support Subsystem (RSS). The RSS (Figure 29) supports and locates the Stationary Core Reflector Subsystem (SCR) and provide vertical support for the control drums. The RSS is suspended from the PCS distribution block by bolted structural straps which utilize a turnbuckle to allow for adjustment and leveling of the support plates. The stainless-steel support plates are each formed from four quarter circle segments which are bolted together around the lower core barrel.

The stainless-steel support plates are each formed from four quarter circle segments which are bolted together around the lower core barrel. Bearing plates of non-galling Nitronic 60 alloy are fitted to the inner diameter of the support plates to prevent adhesive wear between the plates and the core barrel as the barrel expands and contracts due to changes in the temperature of the reactor.

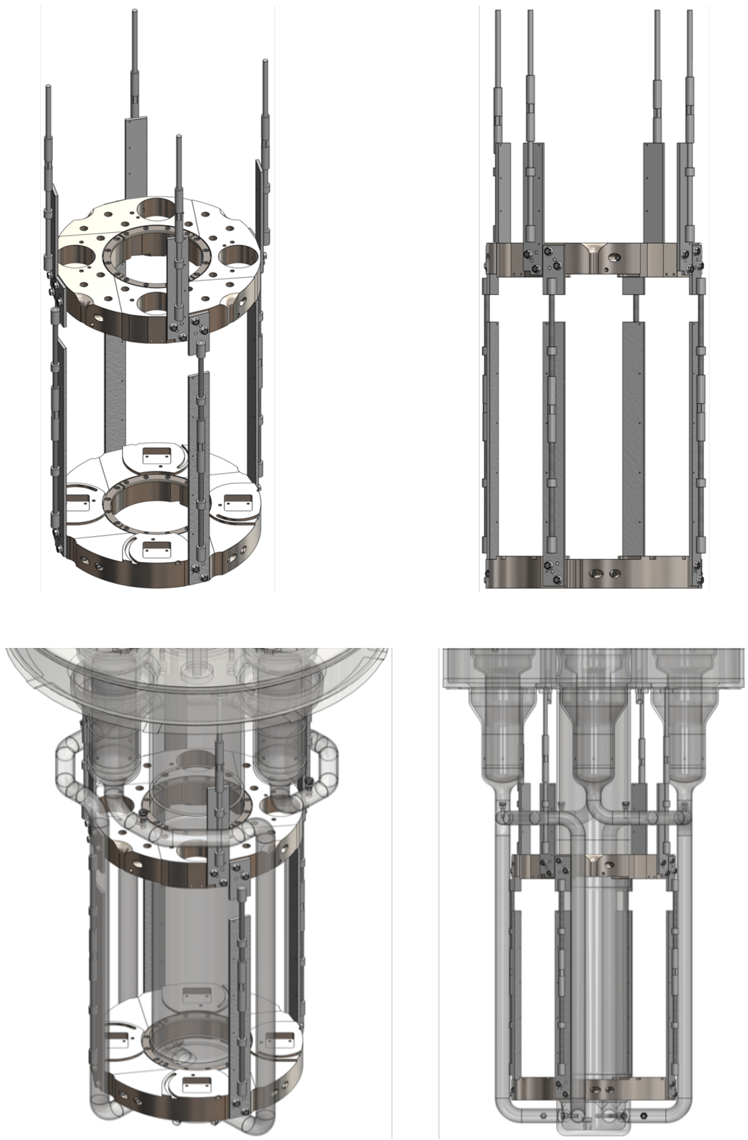

Figure 29. Design illustration of the RSS.

3.4.3.5 Reactor Support Frame (RSF) Overview. The RSF (Figure 30) transfers the load from the GVS (and therefore all the MRS subsystems beside the lower radial shield) to the T-REXC structure.

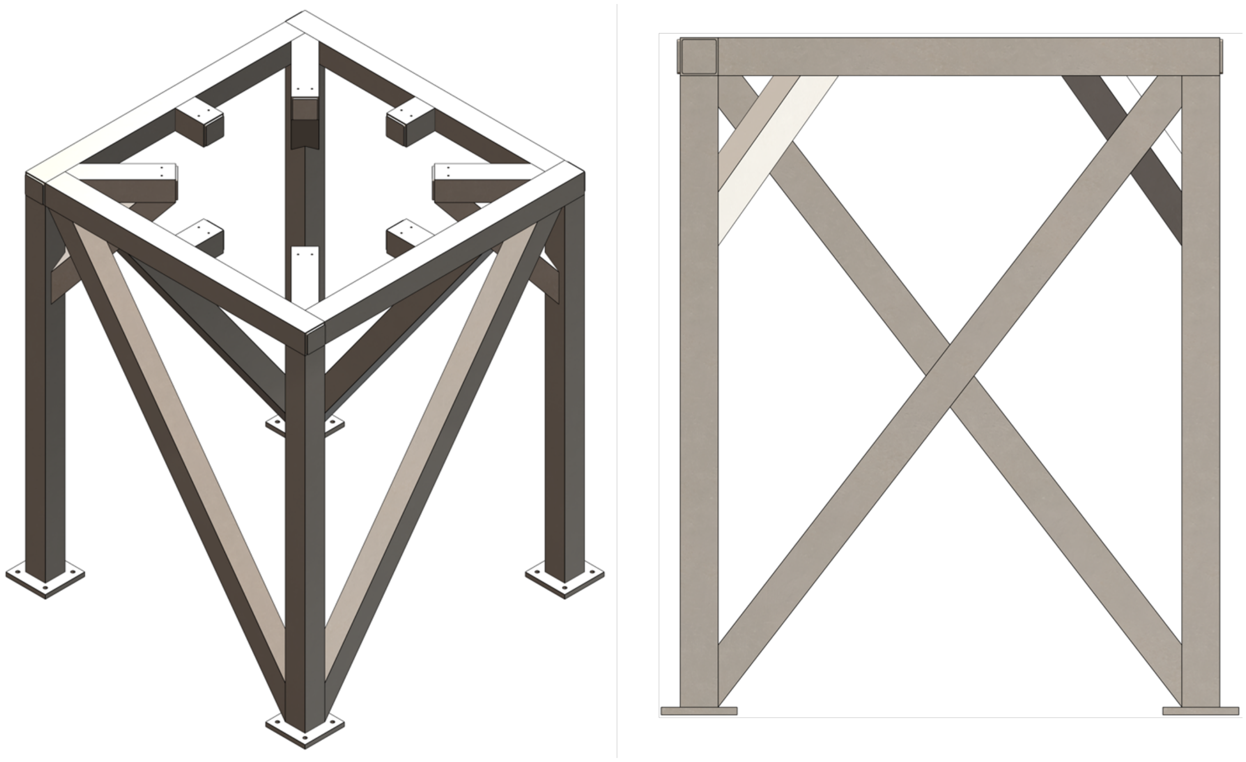

Figure 30. Design illustration of the RSF.

3.4.3.6 Reactor Shielding Subsystem (SHLD) Overview. The SHLD subsystem (Figure 31) consists of both axial and radial shielding components. Axial shields are located on top of the upper RSS plate to reduce gamma and neutron radiation from the core into the upper zone of the MRS. Radial shields surround the lower portion of the GVS to reduce gamma and neutron activation of the T-REXC structures.

Additional Radiation Shields are surrounding the guard vessel and are comprised of a thick stainless-steel plate for gamma radiation shielding, and a container filled with WEP for neutron radiation shielding. Below the reactor support structure is a container filled with WEP for neutron shielding. These components all work together to effectively shield radiation in all directions.

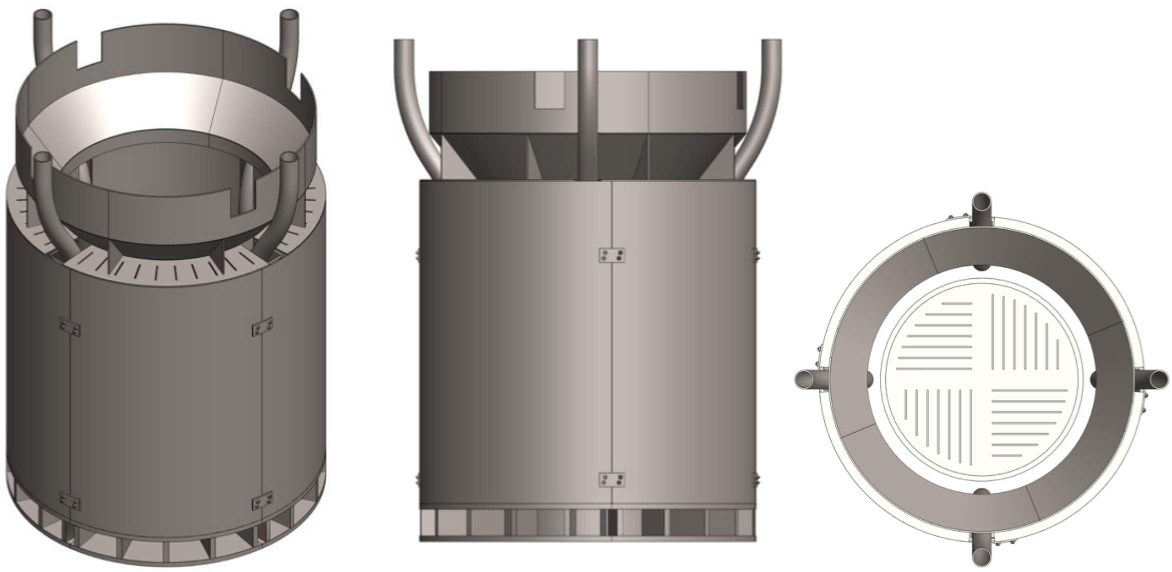

Figure 31. Design illustration of the SHLD.

3.4.3.7 Secondary Support Structure (SSS) Overview. The SSS (Figure 32) is a steel C-channel-based frame which is bolted to the top of the distribution block and suspends four SOSs in the pool of secondary coolant in the IHX. The secondary support structure is designed to be fully assembled with the four SOSs outside the pit and then lowered via three lifting eyes onto the Primary Coolant System, which will already be in the pit. Four SSS assemblies are connected to the top the PCS distribution block, each supporting a SOS structure.

3.4.3.8 Secondary Output Structure (SOS) Overview. The SOS (Figure 32) suspends the Stirling engines in the pool of secondary coolant. The SOS is also responsible for isolating the vibration caused by the operating engines to minimize impacts on the rest of the MRS. Each SOS assembly is designed to be assembled outside the pit and then lowered via lifting eye onto the SSS.

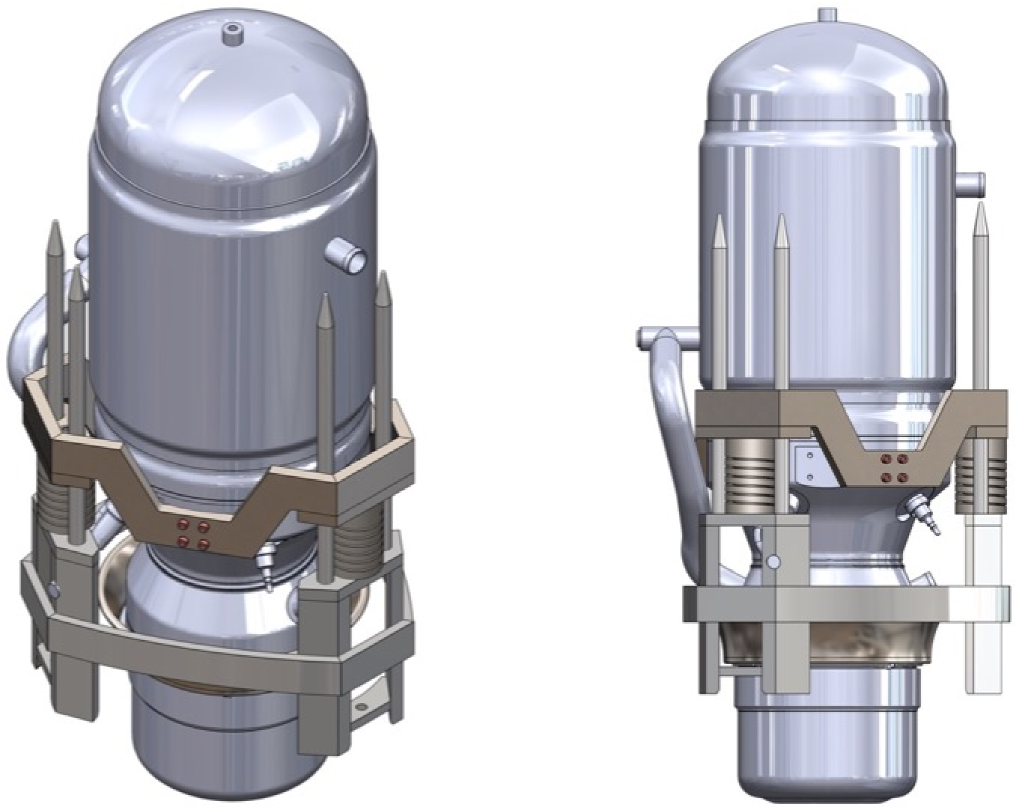

Figure 32. Illustration of the SSS and SOS.

The SOS has been designed to be able to replace the Stirling engine in case of failure. and are corrosion resistant. Four instances of the SOS are connected via bolts in the bottom of the vibration isolators to the Secondary Support Structure (SSS). A lifting eye will be included on top of each PCK80 to allow a single SOS to be lifted out of the pit so that a Stirling can be replaced if needed without removing the entire SSS.

3.4.3.9 Secondary Coolant Management Subsystem (SCMS) Overview. The SCMS (Figure 33) provides the equipment necessary to purify, fill, and drain the secondary coolant.

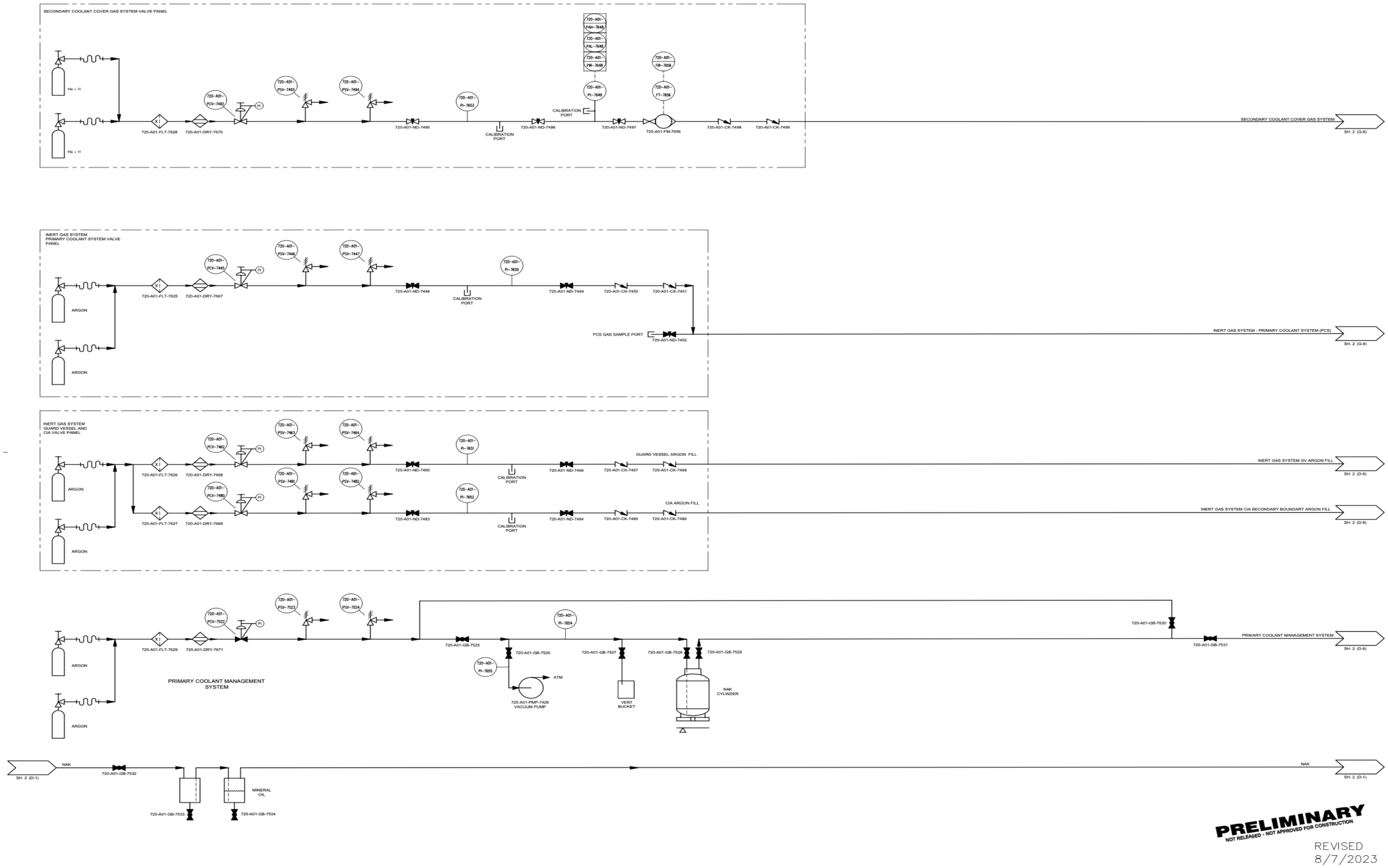

Figure 33. P&ID of PCMS, SCMS and IGS.

3.4.3.10 Primary Coolant Management System (PCMS). The PCMS (see Figure 33) provides the equipment necessary to purify, fill, and drain the primary coolant. A procedure and or process will be developed for the filling of the primary coolant inside the PCS. NaK is used as the primary coolant and must be inserted into the PCS in a specific manner.

3.4.3.11 Inert Gas Subsystem (IGS). The IGS Figure 33) supplies ultra-high purity (UHP) argon to the primary coolant boundary, guard vessel, and the upper confinement space to prevent NaK reactions with air.

The MARVEL reactor uses NaK as its primary coolant fluid, and is maintained under an inert blanket of argon gas from the IGS. Additionally, headspace pressures in the PCS and GVS act in tandem with hydrostatic pressures to constrain the NaK liquid levels within the two vessels in the event of a leak. Pressure in the PCS is to always be higher than GVS to prevent bubbles being introduced into PCS.

3.4.3.12 Secondary Cover Gas Subsystem (SCGS) Overview. The SCGS (Figure 34) provides a helium cover gas to the IHX for chemistry control of the eGa-In-Sn coolant, vents the helium cover gas from the IHX headspace, and filters the exhaust stream to contain the evolved radionuclides. This offgas is transferred to the T-REXC ventilation system for downstream processing.

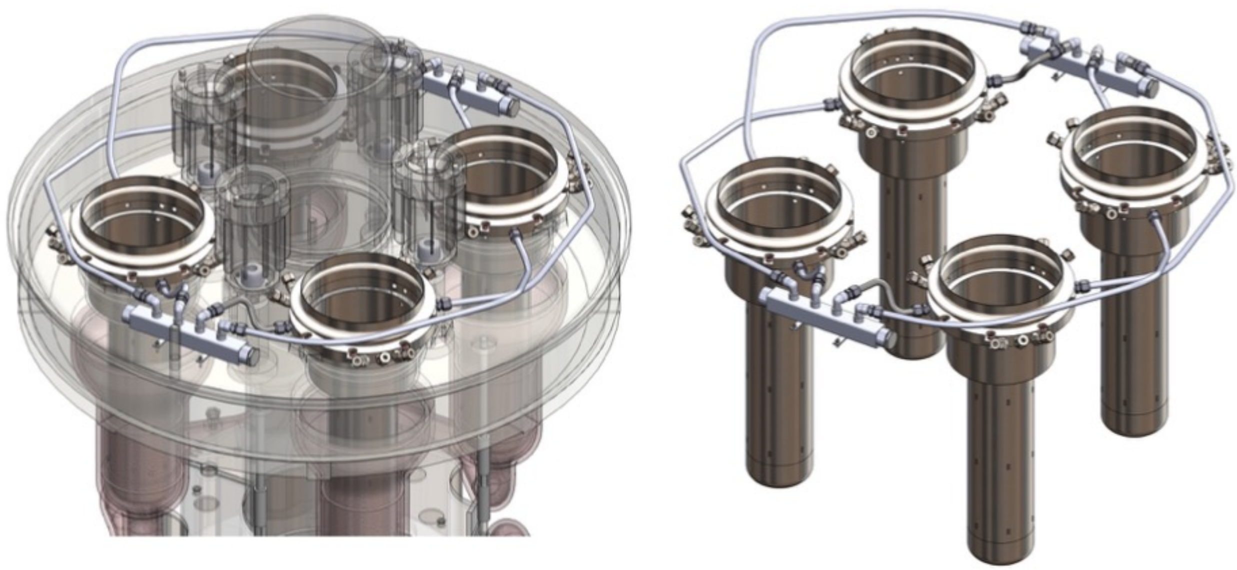

Figure 34. Illustration of the SCGS.

3.5 Reactivity Control System

3.5.1 Design Description

This section describes the Reactivity Control System (RCS) design. Figure 35 shows the overall system architecture of the RCS and how each of the sub-systems interface. Boxes surrounding the system boundary represent interfacing subsystems not within the scope of the RCS. Yellow lines represent electrical interfaces, teal lines represent instrumentation and control (I&C) interfaces, black lines represent important mechanical or structural interfaces, and purple lines represent nuclear interfaces.

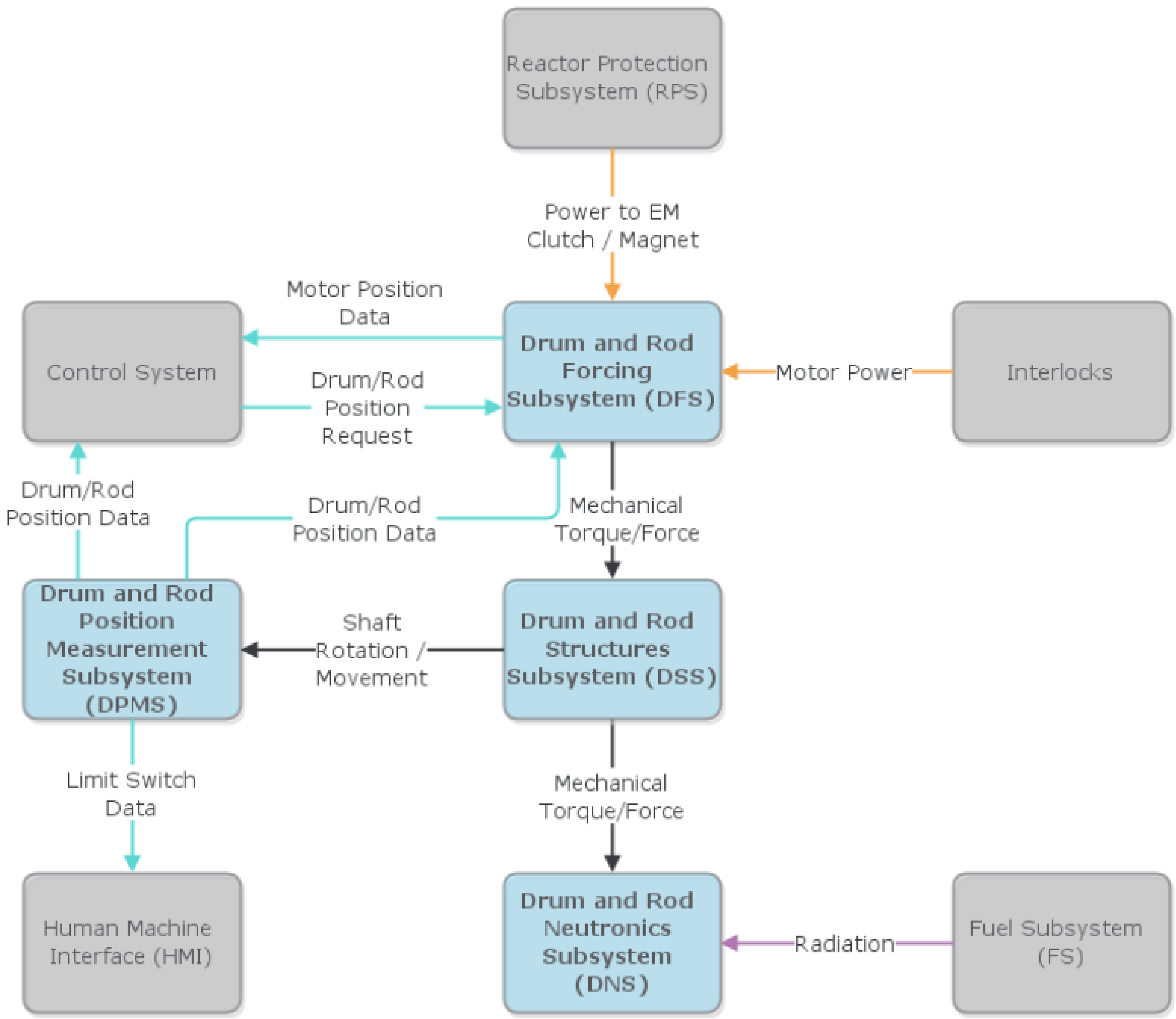

Figure 35. Reactivity control system architecture and interfaces.

The RCS (Figure 36) provides reactivity control during normal operation, included controlled shutdown,and also provides SCRAM shutdown of the reactor in response to abnormal conditions or postulated events. This system consists of four rotatable CDs (Figure 37) evenly distributed about the core periferial and a translatable CIA rod in the center of the core which are actuated by actuators located at the reactor top as driven by the RCS cabinet.

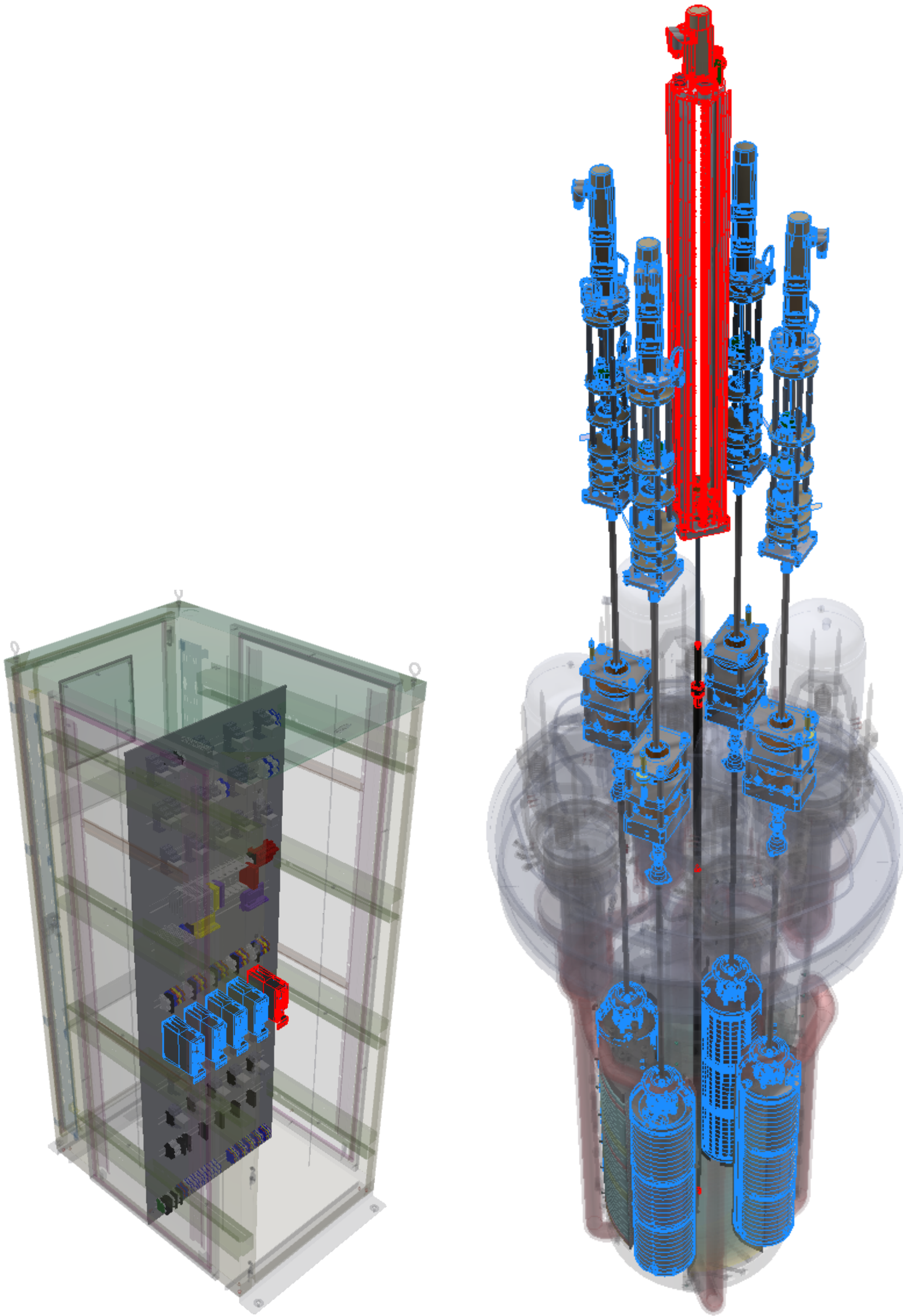

Figure 36. Illustration of Reactivity Control System (left: applicable motor controllers highlighted in the RCS/RPS/Interlock cabinet, right: four CD systems and central CIA system) installed on top of reactor.

The RCS subsystems are described in detail below but are summarized here. The lower portion of the CD and CIA systems constitutes the Drum and Rod Neutronics Subsystem (DNS) and directly influences reactivity via the composition and position of the BeO reflector and B4C absorber materials and its associated interaction with the Fuel Subsystem (FS).

The upper portion of the CD and CIA systems are the actuation structures used to position of the DNS and thus directly control the associated reactivity. Within the actuation structures, the Drum Forcing Subsystem (DFS) consists of servomotors and motor controllers that receive a position request from the control system and motor power via the interlock circuit and then translates the structured electrical energy into mechanical torque on the CD structure or mechanical force on the CIA rod. The DFS reports back the motor position from the resolver to the control system. An electromagnetic clutch is also provided within the DFS that is energized by the interfacing Reactor Protection System (RPS). Upon detection of an accident condition, the RPS deenergizes the CD clutch and CIA electromagnet and the potential energy stored in torsional springs is released to rotate the CDs back to their shutdown position and the gravitational potential energy available to the elevated CIA rod allows it to fall back down to its shutdown position.

The Drum Structures Subsystem (DSS) provides the structural link between the DFS and the neutronics portion of the CD or CIA rod residing about or within the core region. The drum or rod cage, shaft, and shaft assembly can support the imposed forces/torques while providing a reliable reference frame. The DSS also contains a hard stop designed to limit the excess reactivity by limiting the maximum angle which the drum can be turned, or the maximum height which the CIA rod can be withdrawn (Note: the same hard stop is used as the physical shutdown reference as well).

The Drum Position Measurement Subsystem (DPMS) consists of limit switches and potentiometers that more directly report the position of each control drum or the CIA rod to the DFS motor controllers and the control system.

The RCS SSCs interface with the MRS (Section 3.4), and instrumentation and controls (Section 3.6).

3.5.2 Design Bases

The RCS SSC safety classifications are provided in Appendix B. The RCS SSCs perform the following key safety functions:

- Receive input signal and initiate a reactor shutdown by passive negative reactivity insertion of the CDs.
- Initiate a reactor shutdown by passive insertion of the CDs upon loss of off-site power (LOOP).
- Sense a seismic event and provide reactor protection system (RPS) actuation signal to shutdown reactor by passive insertion of the CDs.
- Shut down the reactor and maintain it in a safe shutdown condition by manual operator scram.
- Release CD following signal from RPS manual scram and seismic early warning trip.
- Provide passive insertion of negative reactivity to shut down the reactor and maintain in shutdown condition.
- Ensure unobstructed insertion path and reactor shutdown through structural performance of CDs, guide structures, and core under operating and transient conditions.
- Prevent simultaneous uncontrolled withdrawal of more than one CD as a result of equipment or operator error.

- Limit CD movement to ensure that available excess reactivity insertion does not challenge fuel and temperature limits when inserted instantaneously.
- Provide system performance related to geometric and physics changes in order to provide negative reactivity insertion as a function of temperature increase such that the resulting reactor power is reduced to passive heat rejection levels before fuel and barrel temperature limits are challenged and core damage occurs.

3.5.2.1 Design Criteria. In addition to the overall principal design criteria (PDC) 1-5 in Appendix A, the RCS SSCs shall be designed to meet the following PDC:

- The RCS shall control the rate of reactivity changes resulting from planned, normal power changes to ensure design limits for the fission product barriers are not exceeded [PDC-12].
- Active RCS SSCs important to safety shall be designed to meet the single failure criterion defined in IEEE 379 [PDC-21].
- RCS SSCs important to safety shall be designed to withstand the effects of seismic events without the loss of the capability to perform their safety functions [PDC-22].
- RCS SSCs important to safety shall be designed to fail into a safe state [PDC-23].
- The RCS shall control the rate of reactivity changes resulting from planned, normal power changes to ensure design limits for the fission product barriers are not exceeded [PDC-26].
- The RCS shall provide sufficient negative reactivity to shut down the reactor and maintain it in a safe condition following a normal shutdown or SCRAM [PDC-26].
- The RCS shall control the rate of reactivity changes resulting from planned, normal power changes to ensure design limits for the fission product barriers are not exceeded [PDC-28].
- The RCS shall facilitate decay heat removal in all operating scenarios, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded [PDC-34].

3.5.2.2 Functional and Operating Requirements. The requirements and bases for the RCS are discussed in detail in FOR-868 and TFR-2578.

Key system functional requirements are as follows:

- The RCS shall be capable of moving reactivity control elements to the operator-requested position.
- The RCS shall control the rate of reactivity changes resulting from planned, normal power changes to ensure design limits for the fission product barriers are not exceeded.
- The RCS shall be capable of providing reactivity control element position indication.
- The RCS shall provide sufficient negative reactivity to shut down the reactor and maintain it in a safe condition following a normal shutdown or SCRAM.
- The RCS shall facilitate decay heat removal in all operating scenarios, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.
- Key system operational requirements are as follows:
- Inaccessible RCS SSCs shall be constructed of materials designed to withstand the credited environmental conditions (e.g., temperature, pressure, radiation, etc.) in which they are installed for at least 2 calendar years of operation without maintenance.
- RCS SSCs important to safety shall be designed to withstand the effects of seismic events without the loss of the capability to perform their safety functions.
- RCS SSCs important to safety shall be designed to fail into a safe state.

- Active RCS SSCs important to safety shall be designed to meet the single failure criterion defined in IEEE 379.
- The RCS shall include the features and worth to bring MARVEL to the state of sustained nuclear reaction.
- RCS components shall be designed to facilitate decommissioning.
- SR RCS SSCs must be designed to perform all safety functions with no failure mechanism that could lead to common cause failures under postulated service conditions in accordance with IEEE 323-2003 (R2008).
- The RCS shall provide structural support for internal reactivity control element SSCs.

3.5.3 Description of Systems, Subsystems, and Major Components

Although the CD and CIA systems have distinctly different operations (e.g. rotation versus translation) they are comprised of similar components which relate to the requirements in similar ways as shown in Figure 37.

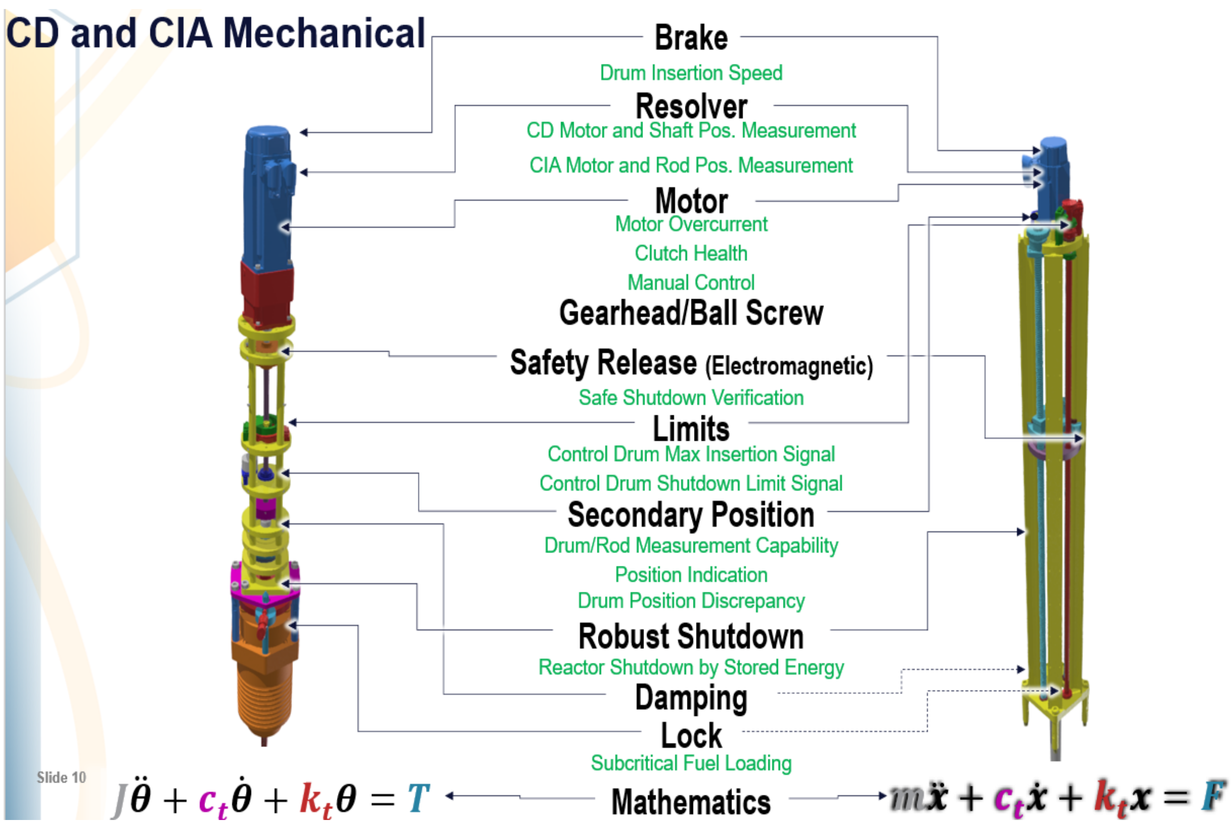

Figure 37. Primary CD and CIA components and their relationship to the requirements.

3.5.3.1 RCS Control Cabinet. The CD and CIA motor controllers are in the RCS/RPS/Interlock cabinet which is further discussed in the I&C section. The controller power is supplied via a 24V power supply and the motor phases are supplied 120V power from the interlock circuit. Communication to the I&C system is achieved via an ethernet connection. The CD and CIA motor power, resolver, limit switches, and potentiometer signals are fed to the motor controllers via in-cabinet wiring, terminals, and intermediate connectors as necessary.

3.5.3.2 CD Position Measurement Components as part of DPMS. The CD actuator position measurement components (Figure 38) provide information about the control drum positional status. The subsystem is to be designed to acquire system data which ensures that conventional motion control can be achieved while accommodating the additional constraints imposed by a nuclear reactor environment. The design implements multiple position and signal processing instruments including the limit switches, the motor resolver as influenced by the two 100:1 gearheads between it and the CD, and the 2nd position indicator (potentiometer) as influenced by its 5:1 gearing.

3.5.3.3 CD Forcing Components as Part of DFS. The CD actuator forcing components shown in Figure 39 control the CD motion by applying torque to the drum. This forcing is designed to either assist or resist the resulting motion and configured/sized such that they accommodate all operational/accident modes. These components include the motor, two 100:1 gearheads in series, the electromagnetic clutch, the two torsional springs, and the damper.

3.5.3.4 CD Actuator Cage Components as part DSS. The CD actuator cage structure (Figure 40) provides the necessary structure to support the imposed forces while providing a reliable reference frame. The more radiation and temperature sensitive 2nd position indicator, limit switches, electromagnetic clutch, and motor components are placed near the top as it provides better wire routing options and better mitigates environmental effects. The structure mitigates the radiation effects via distancing the components from the reactor and each platform provides shielding for all the components above it. The cage can also be disconnected between its lower platform and either the seal platform or the shielding surface which enables the entire actuating assembly to be easily removed. The hard stop is directly attached to the cage but is discussed below.

3.5.3.5 CD Actuator Shaft Assembly as Part of DSS. The CD actuator shaft components (Figure 41) provide the link between the forcing components, position measurement components, and the lower shaft directly attached to the drum. The shaft has been designed to properly couple/interact with the various forcing components and instruments while still accommodating the environmental constraints.

3.5.3.6 Control Drum Seal, Standoff, and Platform with Lock as Part of DSS. The CD seal, standoff, and platform with lock is a configuration central in the CD system which links the actuator, structure, and CD below the seal assembly (Figure 42). The standoff is part of the reactor barrel and is welded to the upper plate of the reactor plenum. It provides mechanical features to mount the CD system. The inner bore provides space for the CD drum shaft assembly.

The Control Drum Seal and Flange is a commercial metal-ceramic face seal is used to seal the rotating CD drive shaft where it passes into the guard vessel. This seal will minimize leakage of argon gas from the guard vessel during operation and ensure that the required gas pressure is maintained. The flange provides the interface between the standoff and seal and allows access to the area below the seal for assembly of the coupling between the control drum shaft and the seal shaft assembly. The flange also provides the mounting features required to mount the CD actuator system to the reactor.

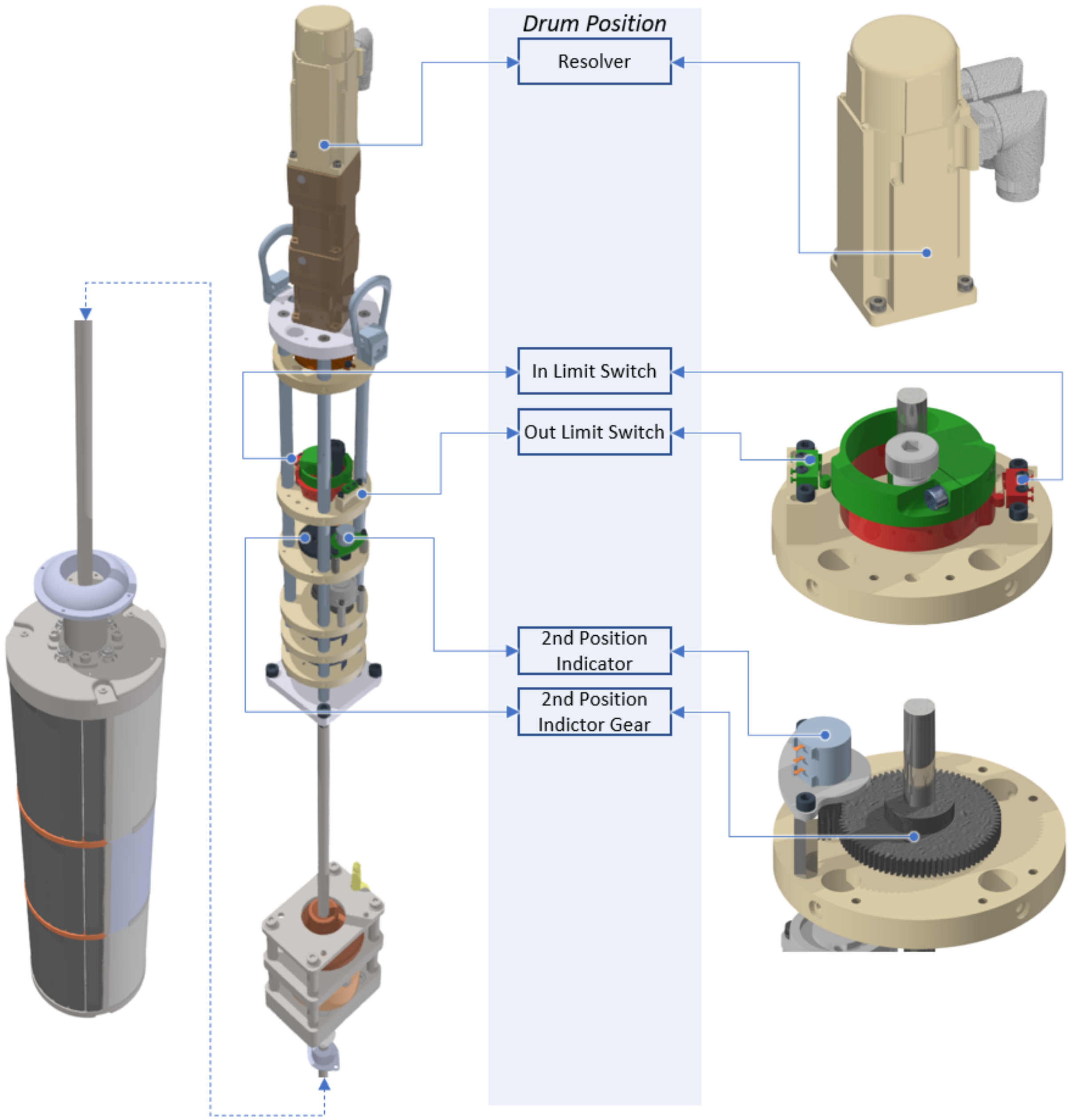

Figure 38: Illustration of CD actuator position measurement components contributing to the DPMS.

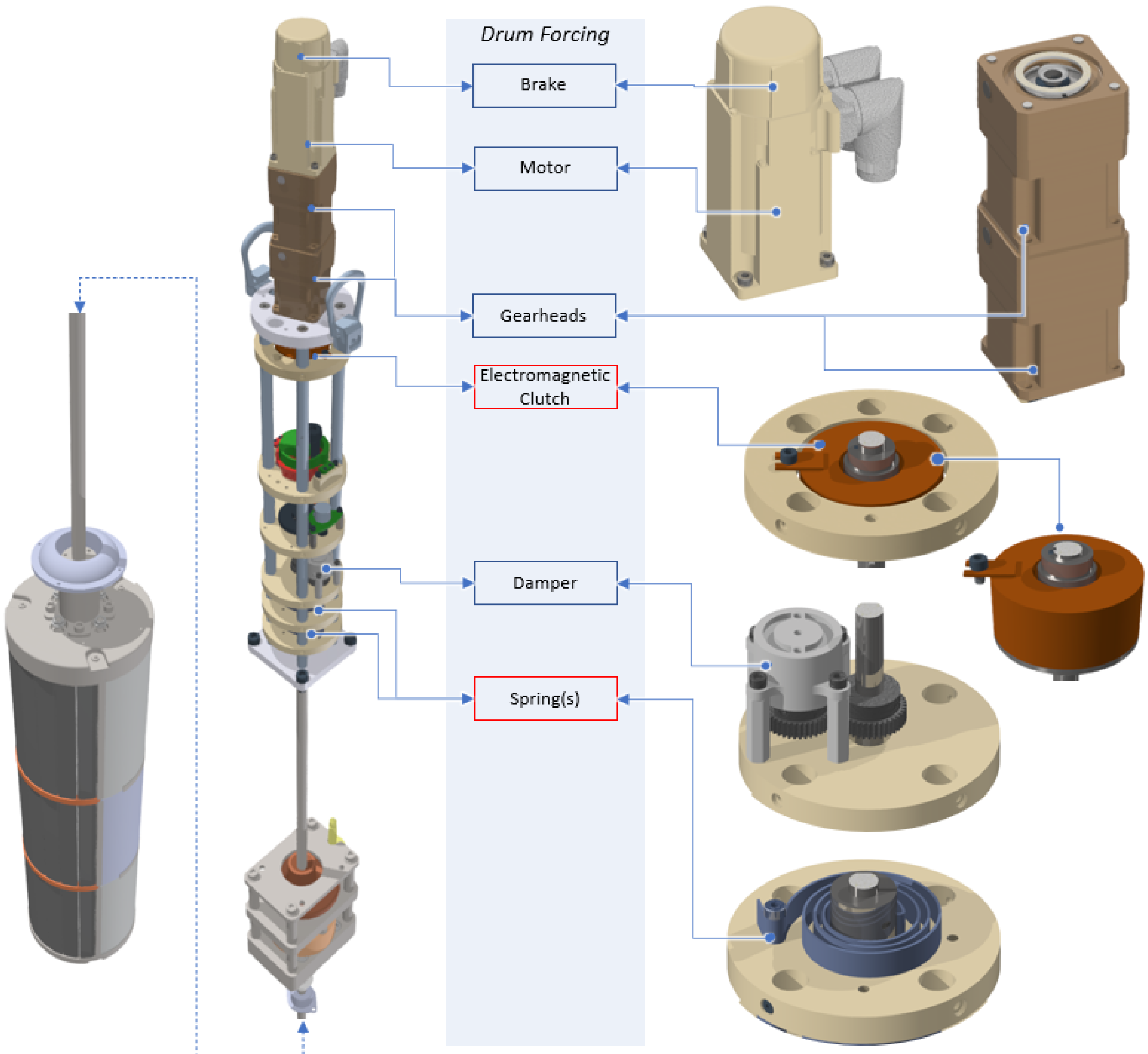

Figure 39: Illustration of CD actuator forcing components (SR components in Red).

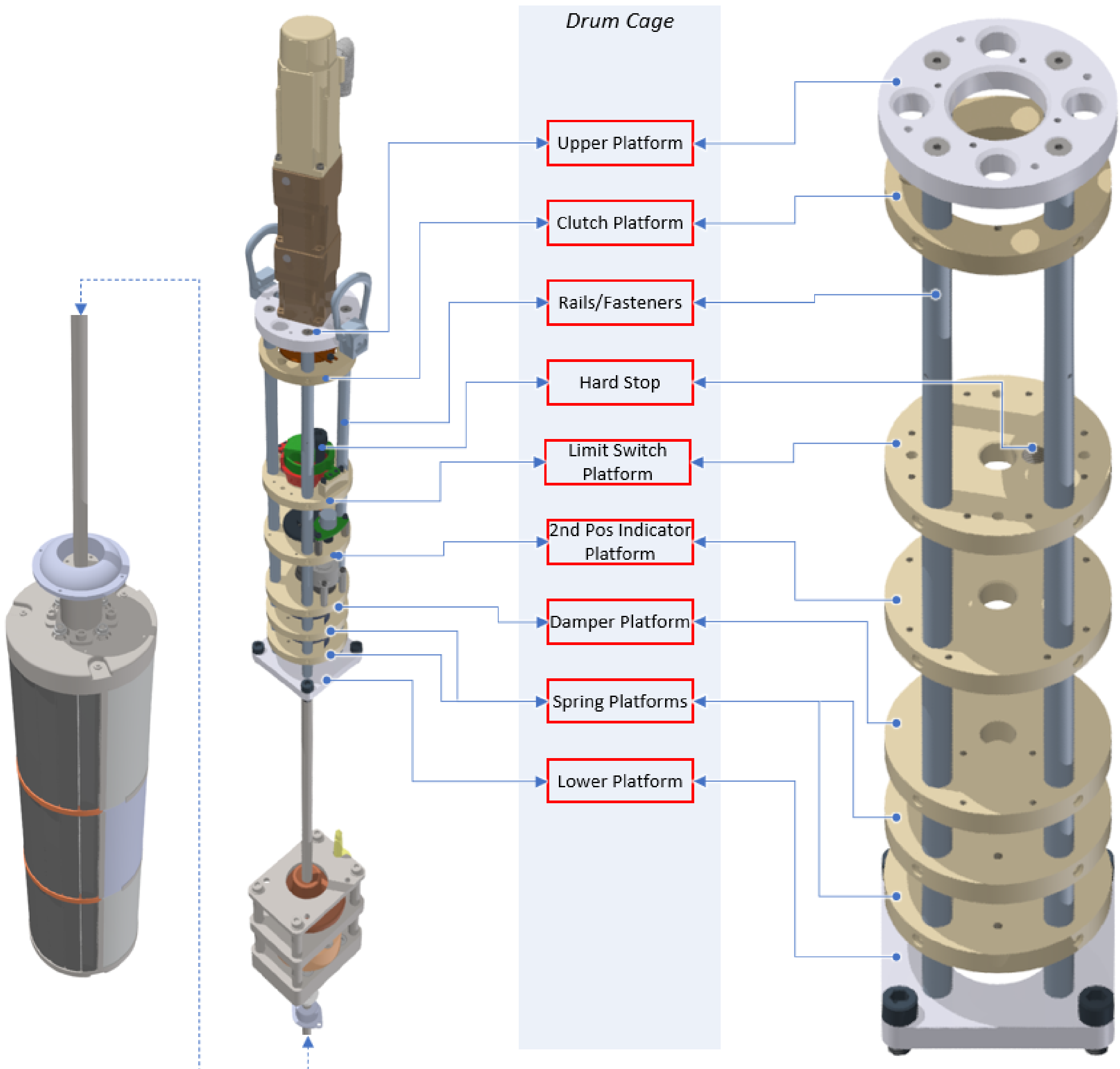

Figure 40. Illustration of CD actuator cage components (SR components in red).

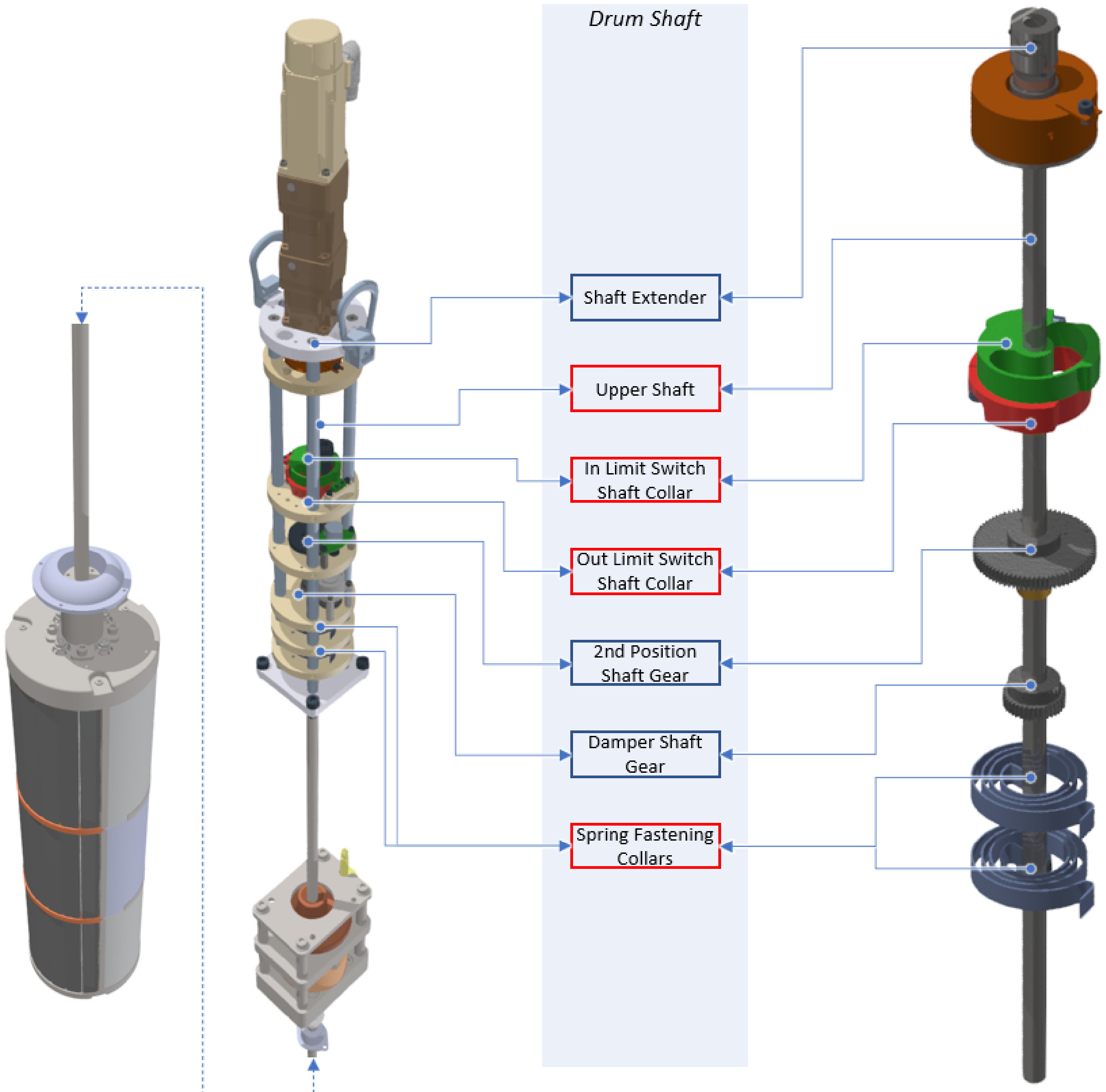

Figure 41. Illustration of drum shaft components (SR components in red).

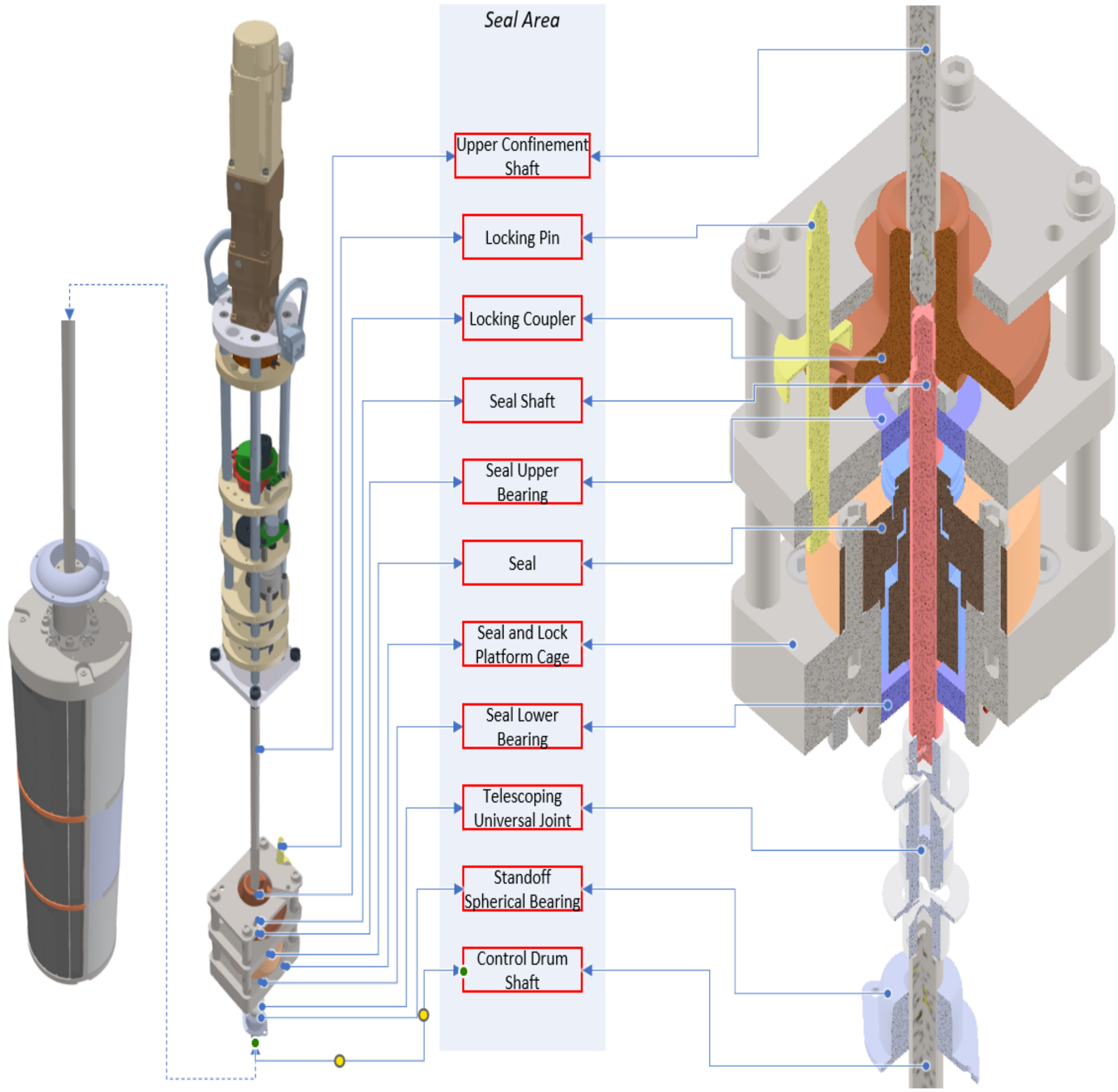

Figure 42. Illustration of CD seal, standoff, and platform with lock.

3.5.3.7 Shaft Connections as Part of DSS. The control drum shaft assembly (Figure 43) is a complex feature that includes an actuator coupler, an upper confinement shaft, a locking coupler, a seal shaft which translates through the seal and is supported on either side by sleeve bearings, a telescoping universal joint, and a secondary confinement shaft which is secured at the top by a spherical bearing and its splined bottom portion is placed in a keyed spline connection to transmit rotation, and terminates above a large flat plate pressing down on the top of the disk stack. The keyed spline connection is located at five points in the shaft including the top of the drum, the top/bottom of the locking coupler, and top/bottom of the coupler between the top of the secondary containment shaft and the bottom of the actuator shaft.

The upper portion of the secondary confinement CD shaft will experience both longitudinal and axial movement with respect to the CD standoff during heat up and cooldown of the reactor, due to differences in thermal expansion of the components of the reactor, drum and shaft. These movements are expected to exceed those allowed by the design of the shaft seal. To avoid transmitting those movements to the CD shaft seal, a telescoping zero-backlash universal joint is incorporated into the shaft design. This universal joint allows relatively large displacements of the shaft without transmitting forces or movements to the seal.

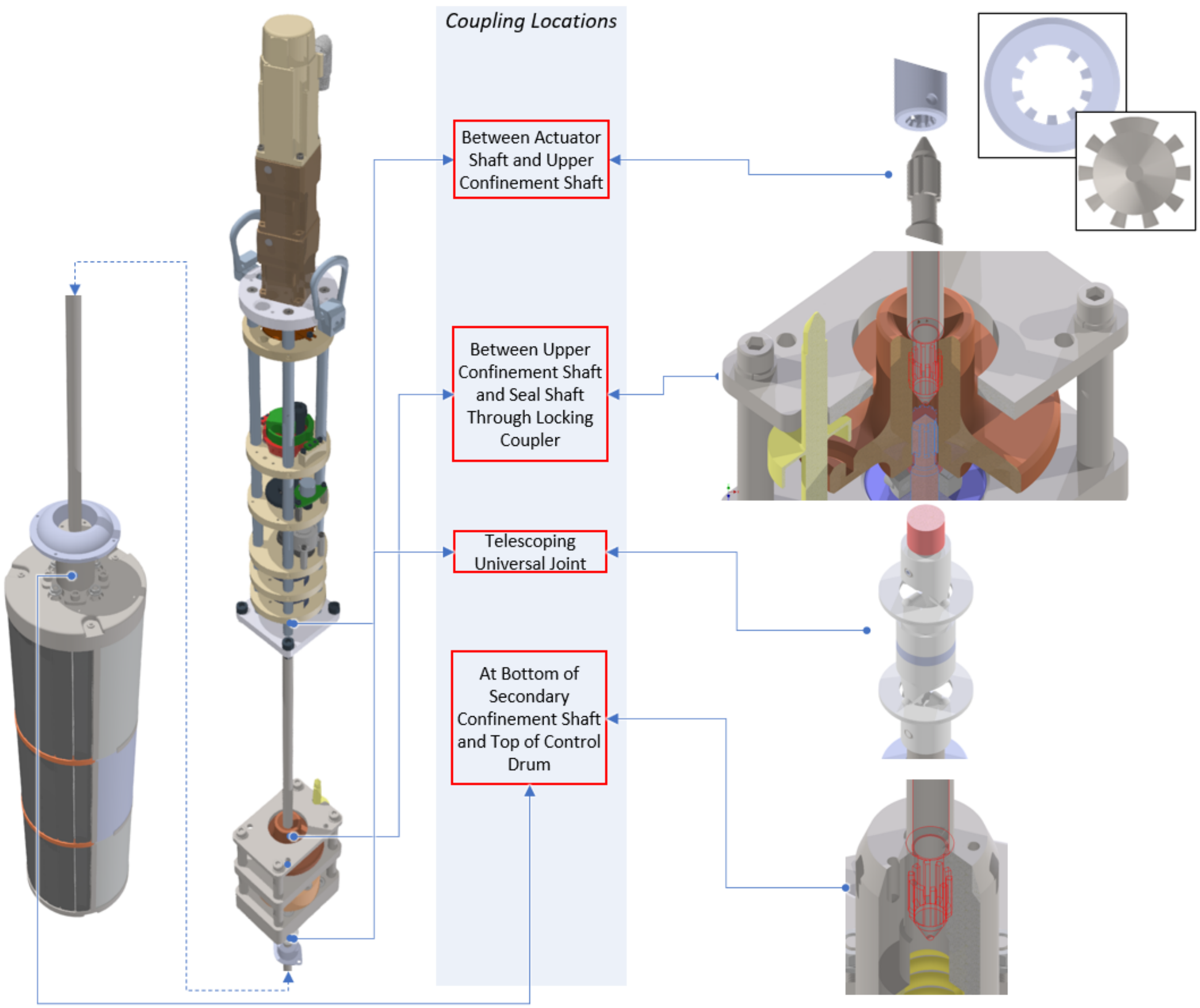

Figure 43. Shaft assembly couplings including five spline couples and a telescoping universal joint.

The connection between the lower end of the secondary confinement CD shaft and the drum is also a keyed spline connection and although the drum is very inaccessible this shaft can be removed with the seal and its attachments with the short shaft through the seal and the telescoping universal joint connections.

3.5.3.8 Control Drum as part of DSS and DNS. The Control Drum (Figure 44) is a complex structure designed to vary the amount of reflected radiation in order to control reactivity. The Shaft Coupler allows the spline end of the CD Shaft to be inserted after assembly of the reactor structure and transmits rotation from the CD Actuator. A Pressure Plate is attached to the Shaft Coupler using six shoulder bolts to provide positive pressure to the reflector disks to reduce the amount of slippage during operation. The shoulder bolts transmit the rotation from the Shaft Coupler to the Pressure Plate. Three alignment rods ensure the reflectors are correctly aligned, provide additional rotational support, and hold the CD assembly together. Beryllium oxide Reflectors provide the necessary radiation reflection. Boron carbide Poison Plates provide the necessary radiation absorption. Support brackets position the Reflectors and Poison Plates. Reflector Shields physically contain the Reflector Plates. Shield Bars position the Reflector Shields. Upper and Lower Caps provide support for the assembly. Metallized graphite bushings mounted in the Reactor Support Structure will interface with the Shaft Coupler and Lower Cap to support the CD while allowing rotation. The top of the CD is supported by a spherical graphite bearing and the bottom is supported by a thrust supporting sleeve bearing.

3.5.3.9 CD Hard Stops as Part of DSS. The CD full-out hard stop (Figure 45) provides a robust reference which to limit the total excess reactivity and ensure that reactivity insertion does not challenge fuel and temperature limits. It consists of a fixed bolt in the actuator frame which interfaces with the full-out limit-switch cam at the end of travel.

The CD full-in hard stops provide a robust shutdown position reference. The hardware implemented to accomplish this task include two dowels at the top and bottom of the CD itself as well as the same fixed actuator bolt used as the full-out hard stop as was just discussed.

The CD full-in and full-out hard stop bolt in the actuator can be the same due to a slot in the cam where one end acts as the full-out limit and the other as the full-in limit. However, there are two cams present because the full-out limit is expected to be adjusted through the life of the reactor. This is achieved by having the lower cam fabricated such that its D-shaped center hole is at an angle which positions the end of the slot to engage with the hard stop bolt at the target maximum allowed reactivity. The angle is initially set based on the initial zero-power reactor physics measurements and the reactivity limitations in Addendum Chapter 15.

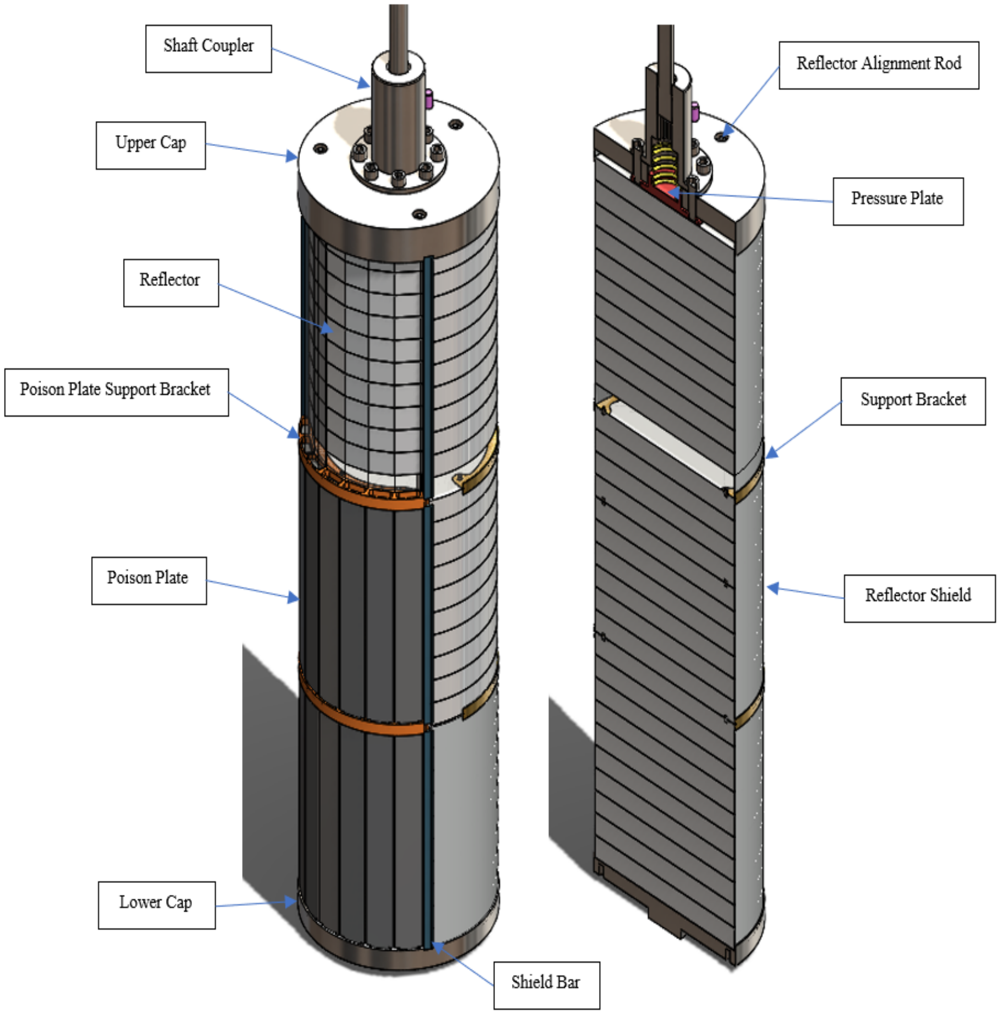

Figure 44. Illustration of CD structure.

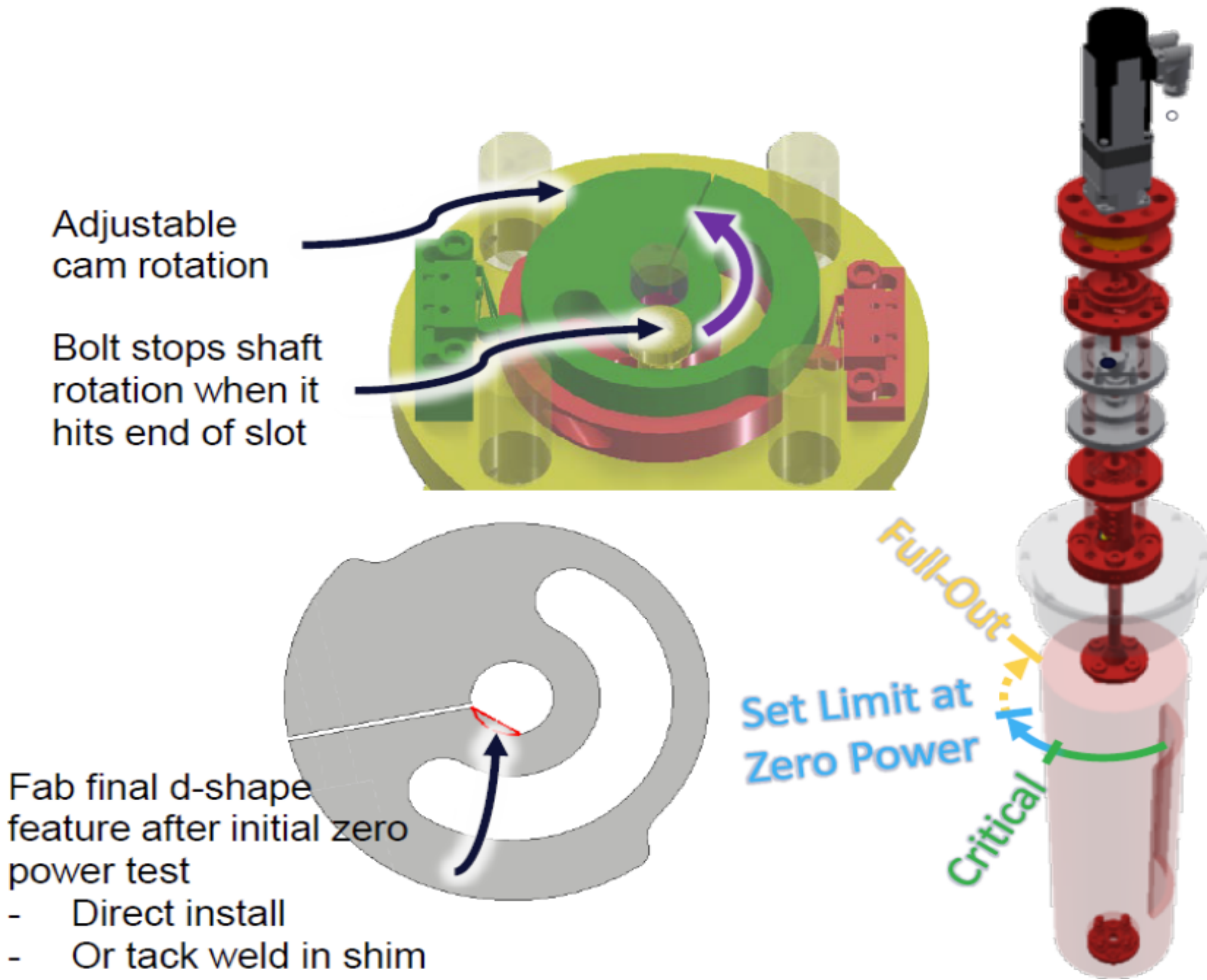

Figure 45. CD hard stop details.

3.5.3.10 CIA Actuator Position Measurement Components as part of DPMS. The CIA actuator position measurement components (Figure 46) provide information about the CIA positional status. The subsystem is to be designed to acquire system data which ensures that conventional motion control can be achieved while accommodating the additional constraints imposed by a nuclear reactor environment. The design implements multiple position and signal processing instruments including the limit switches actuated to a spring deflected rod extending the length of the actuator, the motor resolver as influenced by the 0.2-inch pitch balls screws transferring rotational motor motion into linear motion, and the 2nd position indicator which monitors the position of the lower platform.

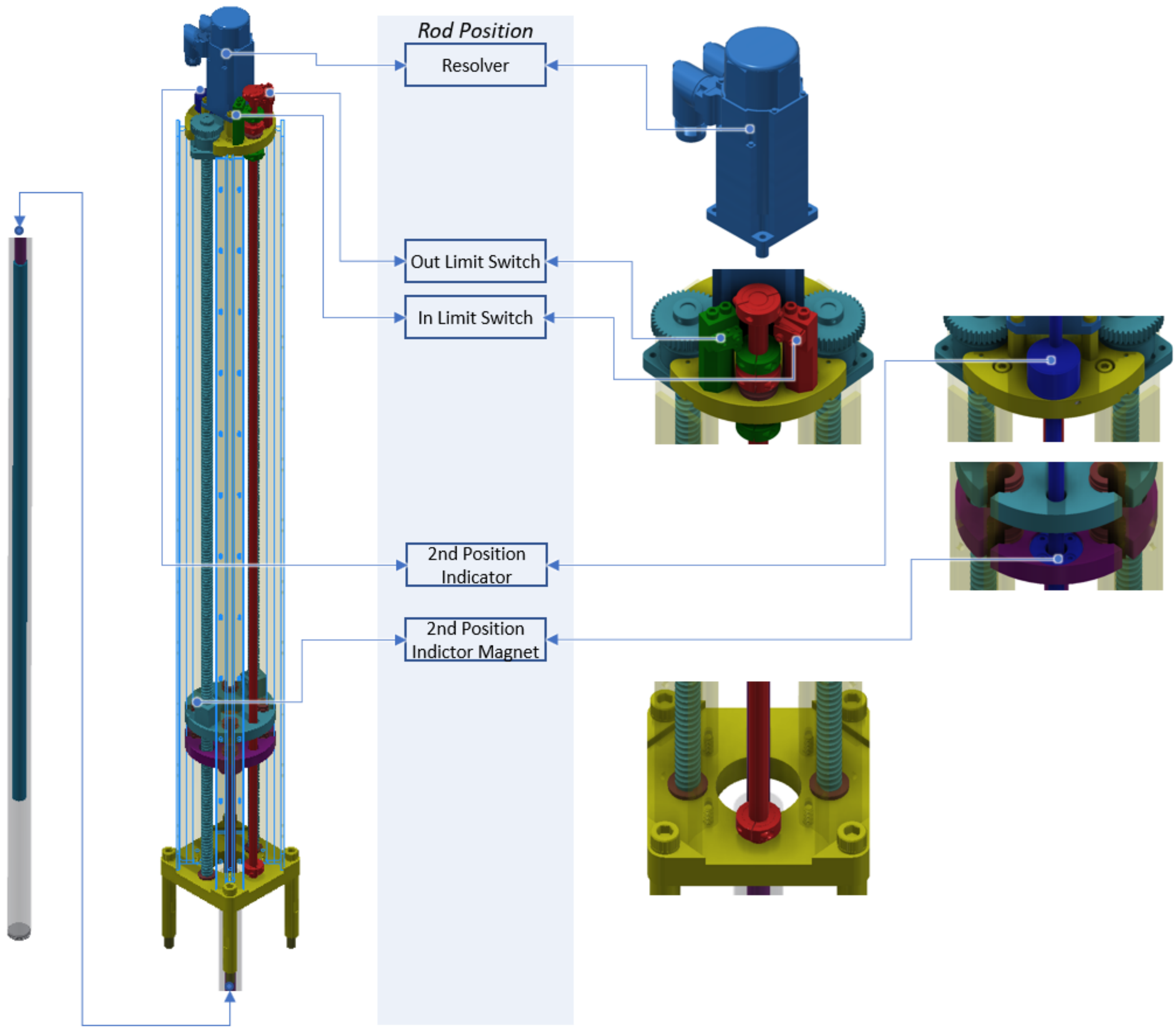

Figure 46. CIA actuator position measurement components.

3.5.3.11 CIA Actuator Forcing Components as Part of DFS. The CIA actuator forcing components (Figure 47) control the CIA motion by applying force to the rod. This forcing is designed to either assist or resist the resulting motion and configured/sized such that they accommodate all operational/accident modes. These components include the motor, two 0.2 inch pitch balls screws, an electromagnet, and general gravity influence.

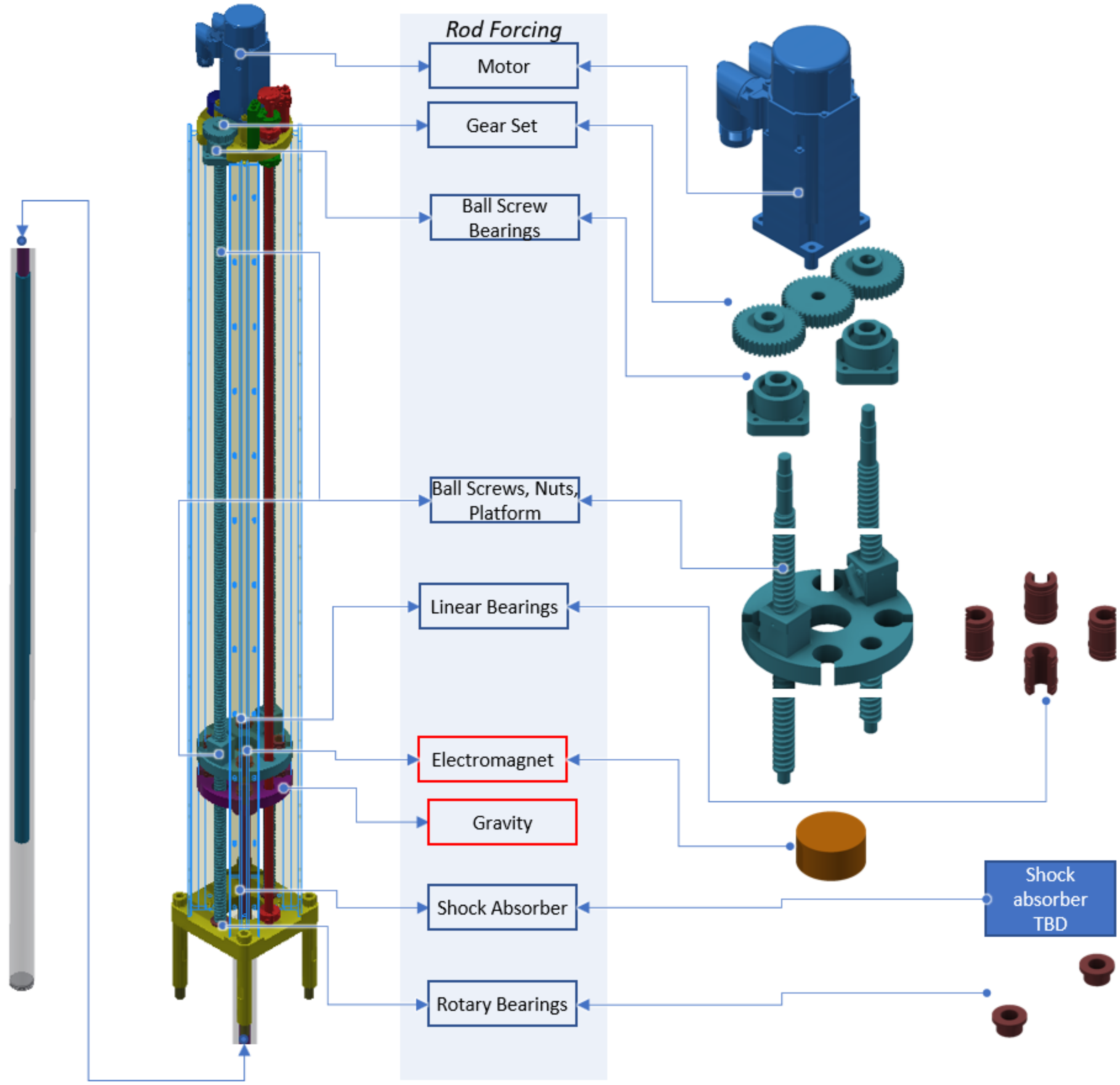

Figure 47. CD actuator forcing components (SR components in Red).

3.5.3.12 CIA Actuator Cage as Part of DSS. The CIA actuator cage structure (Figure 48) provides the necessary structure to support the imposed forces while providing a reliable reference frame. The more radiation and temperature sensitive 2nd position indicator, limit switches, and motor components are placed near the top as it provides better wire routing options and better mitigates environmental effects. The structure mitigates the radiation effects via distancing the components from the reactor and each platform provides shielding for all the components above it. The cage can also be disconnected between its lower platform and the reactor head or shielding platform which enables the entire actuating assembly to be easily removed while still maintaining the seal.

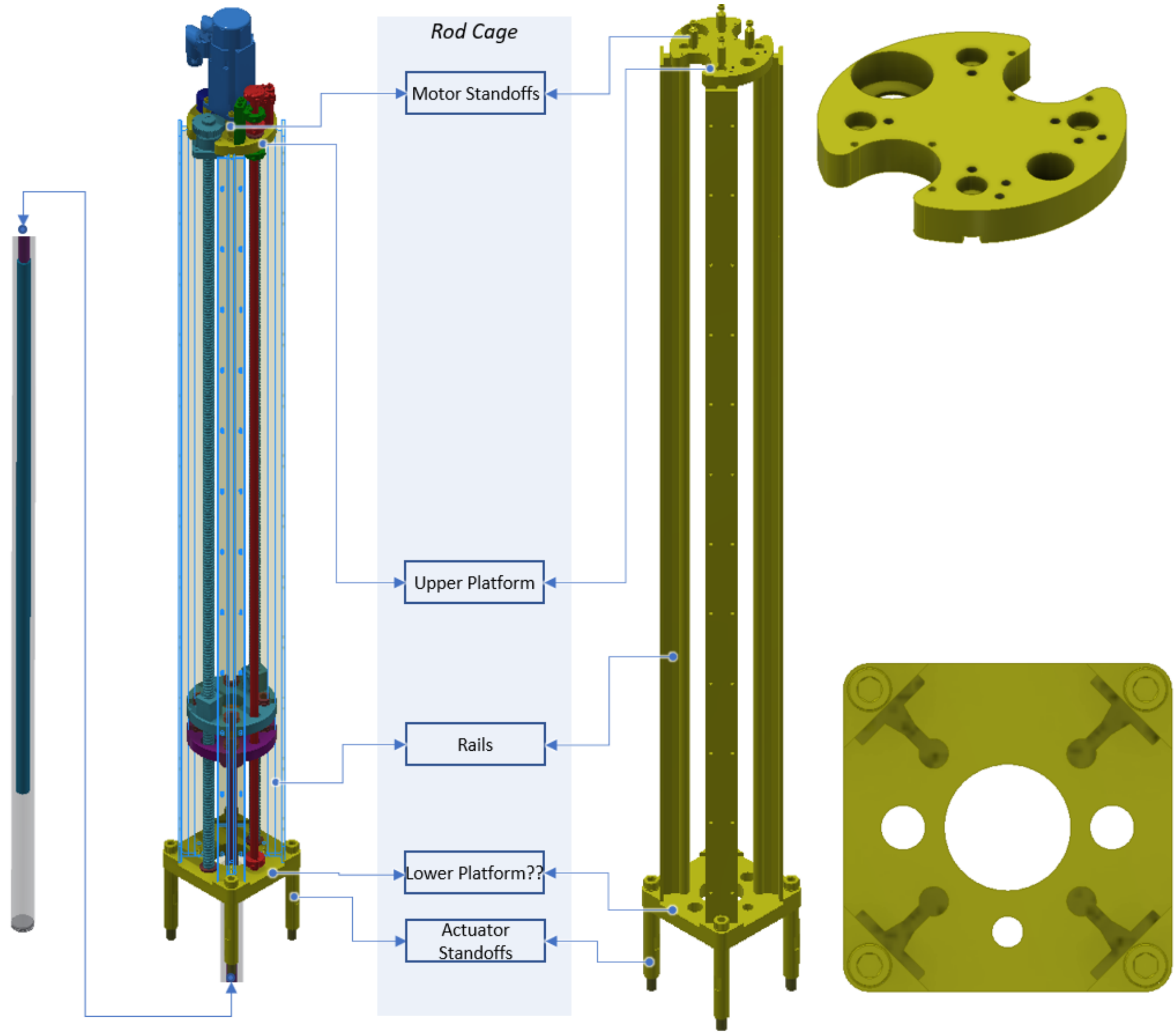

Figure 48. CIA actuator cage components.

3.5.3.13 CIA Actuator Lower Platform and Rod as Part of DSS. The CIA actuator's lower platform (see Figure 49) provides the link between the forcing components, position measurement components, and the lower shaft directly attached to the rod. The lower platform shall be designed to properly couple/interact with the various forcing components and instruments while still accommodating the environmental constraints.

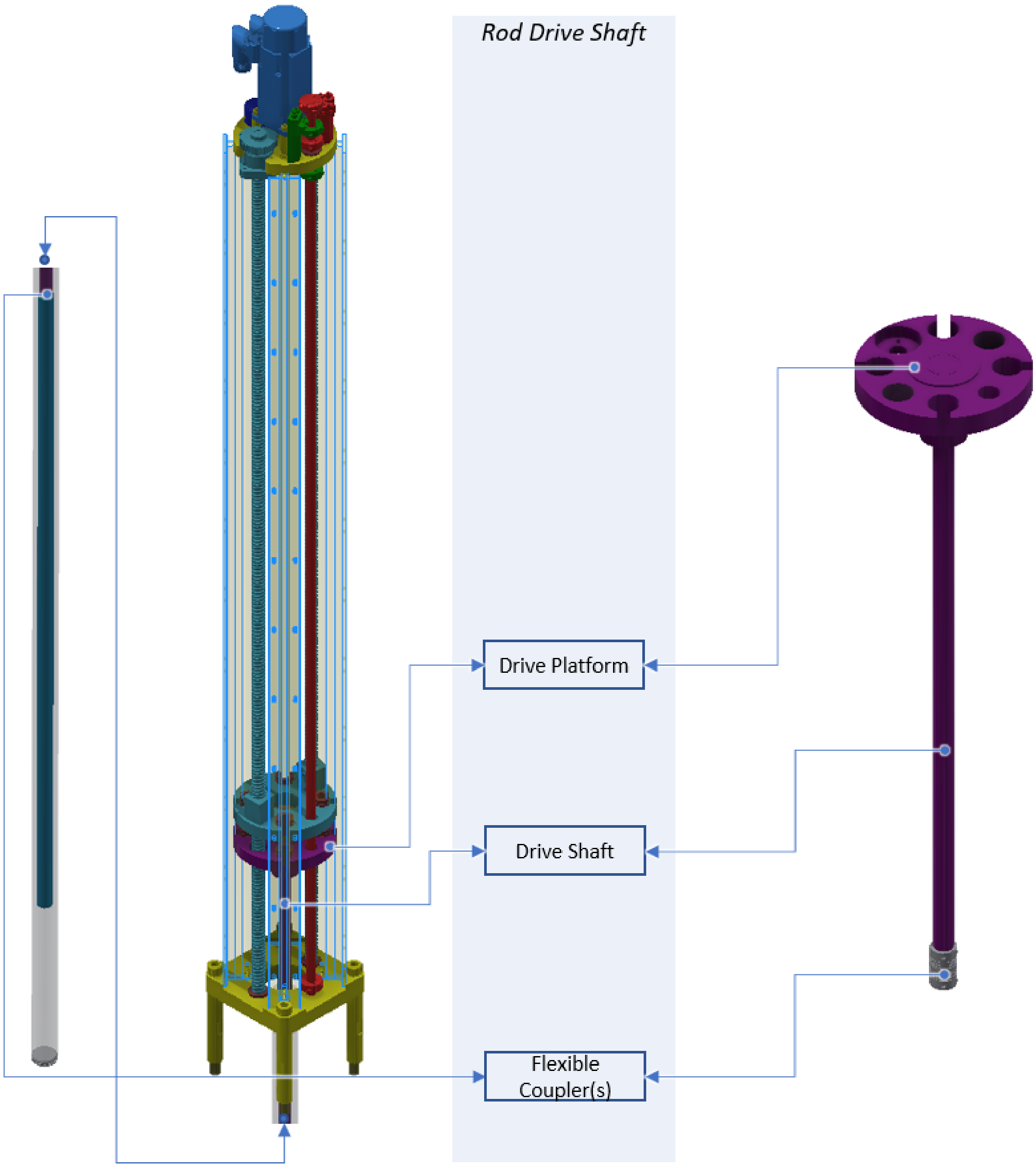

Figure 49. CIA shaft components.

3.5.3.14 CIA Hard Stops as Part of DSS. The CIA rod hard stops (Figure 50) function to limit the total excess reactivity possible and provides a reliable shutdown position reference. Unlike the CDs, the CIA system is not being used for refined operational reactivity control but rather as either a pre-operation reactivity hold-down mechanism or to support reactor SCRAM. As such the full-out hard stop is such that the CIA rod is completely removed from the core and the full-in hard stop is such that the CIA rod is fully inserted into the core.

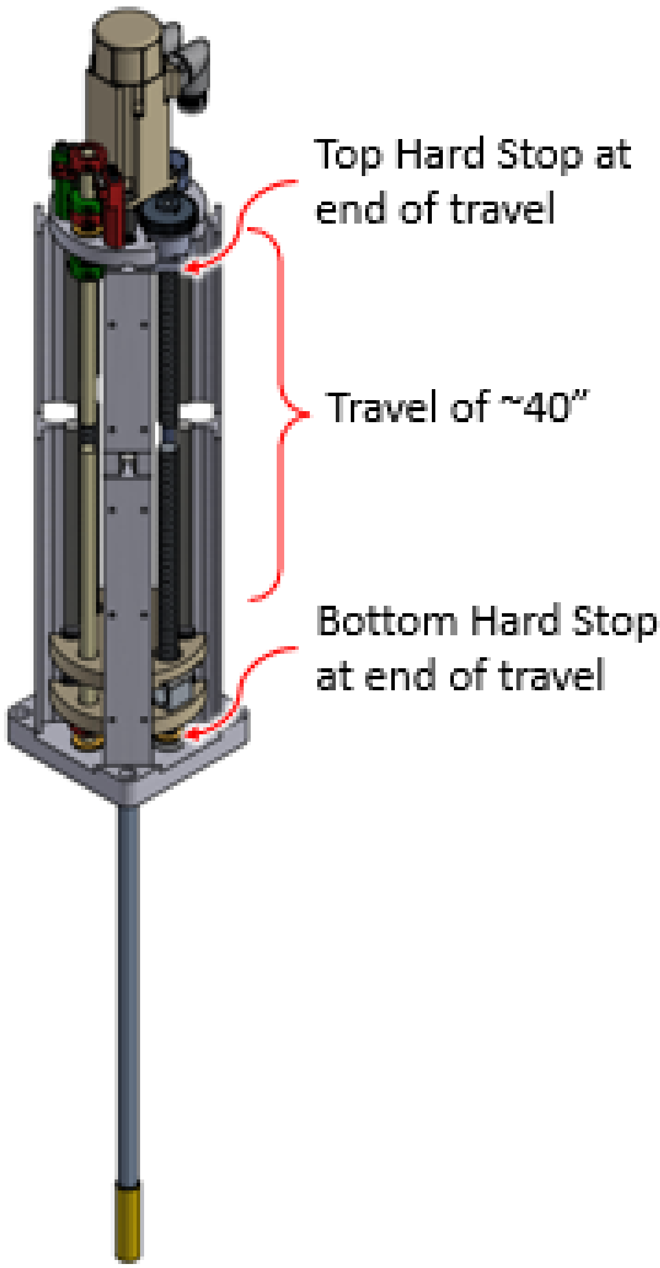

Figure 50. CIA interlock hard stops.

3.5.4 Design Evaluation

The evaluation of the RCS's CD and CIA system designs are embodied in the ECAR-7228 analysis and the PLN-6874, "MARVEL Reactivity Control System Assembly and Checkout (Phase I) and Functional Testing (Phase II)."[25] Both documents were constructed to specifically addresses the requirements contained in FOR-868 And TFR-2578. To be consistent with the above format the analysis results are presented by subsystems including the DPMS, DFS, and DSS.

The CD components contributing to the DPMS include the resolver, limit switches, and potentiometer. The resolver, in combination with the 100:1 x 100:1 = 10,000:1 gearhead chain has a precision that well exceeds the +/-0.1-degree requirement. The selected potentiometer has a 3-turn (1080 degree) range which is coupled with a 5:1 gear ratio between it and the shaft which when considering the voltage input range and noise characteristics produces a precision of +/-0.054-degrees. The limit switches are more simple mechanisms and can only report signal continuity or discontinuity and thus was not evaluated based on precision but rather it was shown that cam mechanism which actuates the limit switch lever arm compliments the device's workspace. The CD components contributing to the DFS include the motor, two 100:1 gearheads in series, the electromagnetic clutch, the two torsional springs, and the damper. The analysis found that although the system consisted of six motion influencing components it was actually the clutch's maximum static torque (80 in-lbf.) that drove the DFS component's sizing. The clutch's torque limit protected the rest of the system from the motor and gearhead's excessive amount of available torque, which came as a byproduct of the effort to successfully limit the motor's maximum physically achievable speed at the gearhead to be less than 2 deg/s set as an operational human factor parameter.

Although the torsional spring stiffness of 0.165 lbf*in/deg is thought to only be influenced by the less than 2 second SCRAM requirement, it was the clutch's torque limit which set the maximum torque allowed at the maximum possible full-out position of 180 degrees to be less than the 80 in-lbf. minus the system's present frictional torque and thus also influenced the spring's pre-set torque and thus deflection to be 90 degrees. Per the figure below, without a damper to supplement the system's base frictional torque the system would be able to SCRAM in about 1 second. However, this was at the expense of a large impact on the hard stops. With a damper the SCRAM time increases to ~1.8 seconds but the impact speed on the damper was significantly decreased. With the damper and 1 spring the system SCRAMS to near or below critical at around 4 seconds (Note: this condition would only exist if 1 spring were to become unavailable thus leaving only 1 spring available for SCRAM which is a very unlikely condition and one would not expect such a state for all 4 CDs.

Although the two hard stop dowels above and below the CD also share the SCRAM impact this analysis conservatively assumed that the actuator's hard stop bolt would absorb all the impact. Consequently, the hard stop bolt diameter was increased from 3/8" to 5/8", which provided marginal performance, and a damper was introduced which extended the SCRAM time to just below 2 seconds but significantly decreased the hard stop bolt's impact demand to capacity ratio to 0.2.

The CD components contributing to the DSS were evaluated as part of a detailed FEA analysis which was segmented into thermal, structural, and eigenvalue analyses. Like the analyses, the qualification plan is constructed of tasks which generate quantitative evidence that the CD and CIA systems meet the requirements. In that vein each action references the requirement which it supports. The plan will allow one to demonstrate that when the CD and CIA systems are subject to the expected thermal induced deflections, they are still able to achieve the required position control and required SCRAM characteristics. Most notable are the system's ability to move to various positions at target speeds, achieve those positions with the appropriate precision and to SCRAM within the specified less than 2 seconds time interval. Although the basic set of tasks is applied to each system, an extra system will also be subjected to a statistically appropriate number of cycles to verify design robustness and dependability.

3.6 Instrumentation and Control System

3.6.1 Design Description

This section describes the MARVEL ICS design. Figure 51 as taken from TFR-2574, shows the overall system architecture of the ICS and how each of the subsystems interface. Yellow lines represent electrical interfaces and teal lines represent I&C interfaces.

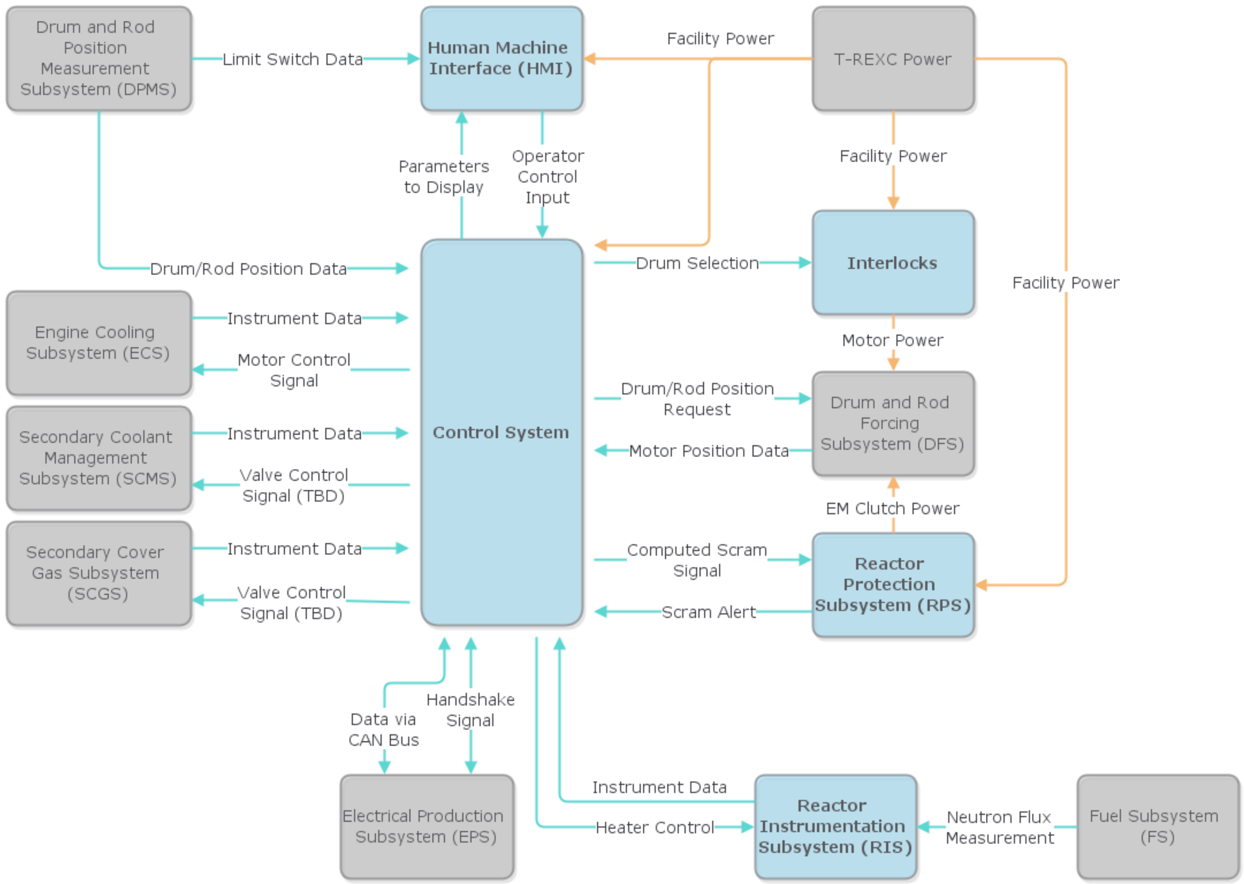

Figure 51. Instrumentation and control system architecture and interfaces.

The Reactor Instrumentation System (RIS) is responsible for measuring critical operating parameters of the reactor systems including temperature, pressure, the presence of NaK outside the primary coolant boundary, and the neutron count. These measurements are routed to the control system for processing and subsequent action.

The control system serves as the primary integrator between sensors and motors in the Power Generation System (PGS), sensors and controllers in the Reactivity Control System (RCS), and sensors within reactor systems. The control system accepts parameters from all operating instruments, performs calculation and logic solving, and transmits outputs to various actuators. It also accepts control requests from the operator via the HMI.

The Reactor Protection System (RPS) initiates the immediate shutdown of the reactor via a relay circuit. The system accepts signals from either the T-REXC seismic sensors, the manual scram button, or the control system and removes power from the electromagnetic clutches of the control drums and electromagnet of the CIA rod to begin the passive insertion of negative reactivity. The system will also passively trip the reactor in the event of a loss of power.

The interlocks ensure that only one control drum (or CIA rod) can be moved at a time. The system consists of a set of relays that route facility power to the control drum and CIA rod motors. The relays are controlled by signals from the control system.

The HMI provides the interface for operators to either route control requests to the reactor or to monitor system parameters transmitted by the control system.

3.6.2 Design Bases

The ICS SSC safety classifications are provided in Appendix B. Key safety functions of the ICS SSCs are to:

- Prevent simultaneous uncontrolled withdrawal of more than one CD as a result of equipment or operator error.
- Prevent uncontrolled withdrawal of the CIA rod as a result of equipment or operator error.
- Provide human interface to software to ensure CD/CIA insertion under normal operating and transient conditions.
- Provide reactor pressure indication under normal operating and transient conditions.
- Provide CD/CIA full in and full out position indication under normal operating and transient conditions.
- Provide control system support.
- Run control system.
- Supply uninterrupted power to control system
- Provide control system support and power instruments
- Shut down the reactor and maintain it in a safe shutdown condition by manual operator scram.
- Provide DC power to clutch mechanisms.
- Provide an additional level of protection to ensure reactor not started without permission.
- Sense a seismic event and provide RPS actuation signal to shutdown reactor by insertion of the CDs.
- Receive input signal and initiate a reactor shutdown by insertion of the CDs.
- Upon loss of off-site power (LOOP), initiate a reactor shutdown by insertion of the CDs.
- Indication of a NaK leak in the upper confinement
- Provide dynamic power indication.
- Measure across the upper plenum to verify dT within what was used in the stress analysis.
- Detect NaK leaks.
- Sense pressure differential between primary and guard vessel.

3.6.2.1 Design Criteria. In addition to the overall principal design criteria (PDC) 1-5 in Appendix A, the ICS SSCs shall be designed to meet the following PDC:

- The ICS shall have the ability to maintain MARVEL systems within their normal operating ranges [PDC-10].
- The ICS shall have the capability to automatically shut down the reactor safely to ensure the fuel design limits are not exceeded during any condition of normal operation or anticipated event [PDC-10].

- The I&C instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated events, and for postulated accident conditions, as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the primary coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges [PDC-13].
- The ICS shall have the capability to automatically shut down the reactor safely to ensure the fuel design limits are not exceeded during any condition of normal operation or anticipated event [PDC-15].
- Normally accessible ICS SSCs important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features [PDC-18].
- A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under postulated accident conditions [PDC-19].
- The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that design limits for the fission product barriers are not exceeded as a result of anticipated events and (2) to sense accident conditions and to initiate the operation of SR and NSR-AR SSCs [PDC-20].
- The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred [PDC-21].
- The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function [PDC-22].
- The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, NaK and NaK reaction products, pressure, steam, water, and radiation) are experienced [PDC-23].
- The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired [PDC-24].
- Active ICS SSCs important to safety shall be designed to meet the single failure criterion defined in IEEE 379 [PDC-25].
- The protection system shall be designed to ensure that design limits for the fission product barriers are not exceeded during any anticipated events accounting for a single malfunction of the reactivity control systems[PDC-26].

- A minimum of two independent reactivity control systems or means of different design principles shall be provided [PDC-26].
- The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity postulated accidents can neither (1) result in damage to the primary coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core [PDC-28].
- The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated events [PDC-29].

3.6.2.2 Functional and Operating Requirements. The requirements and bases for the ICS are discussed in detail in FOR-868 and TFR-2574. Key system functional requirements are as follows:

- The ICS shall monitor reactor variables and systems over their anticipated ranges for normal operation, for anticipated events, and for postulated accident conditions, as appropriate to ensure adequate safety.
- Means shall be provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage.
- The ICS shall have the ability to maintain MARVEL systems within their normal operating ranges.
- The ICS shall have the capability to automatically shut down the reactor safely to ensure the fuel design limits are not exceeded during any condition of normal operation or anticipated event.
- The ICS shall provide data from monitored parameters to the control room, including those parameters necessary for post-accident monitoring.
- The ICS shall initiate a reactor scram under postulated accident conditions.
- The ICS shall provide the capability to start power generation equipment safely and in a controlled fashion when prompted by an operator.
- The ICS shall provide the instrumentation necessary to start the reactor safely and in a controlled fashion by an operator using manual control.
- The ICS shall provide the capability to change the reactor power safely and in a controlled fashion based on the input from an operator, while limiting the potential amount and rate of reactivity increase.
- The ICS shall provide the capability to change the power generation output level safely and in a controlled fashion based on the input from an operator.
- The ICS shall provide the capability to shut down the reactor safely and in a controlled fashion when initiated by an operator.
- The ICS shall have the capability to shut down power generation equipment in a controlled fashion when initiated by an operator.

Key system operational requirements are as follows:

- ICS SSCs within the secondary confinement boundary (guard vessel) shall be constructed of materials designed to withstand the credited environmental conditions (e.g., temperature, pressure, radiation, etc.) in which they are installed for at least 2 calendar years of operation without maintenance.
- ICS SSCs important to safety shall be designed to withstand the effects of seismic events without the loss of the capability to perform their safety functions.
- ICS SSCs important to safety shall be designed to fail into a safe state.
- Active ICS SSCs important to safety shall be designed to meet the single failure criterion defined in IEEE 379.
- The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.

- SR ICS SSCs must be designed to perform all safety functions with no failure mechanism that could lead to common cause failures under postulated service conditions in accordance with IEEE 323-2003 (R2008).
- ICS components shall be designed to facilitate decommissioning.
- ICS SSCs important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions, consistent with DOE-STD-1066.

3.6.3 Description of Systems, Subsystems, and Major Components

3.6.3.1 Control System. The control system (Figure 52) has input and output for various analog and digital signals. The Control System is an NSR-AR system. The function of the control system is to:

1. Take operator commands and relay the command to cause movement of the CD's and CIA rod for reactivity control
2. Let the operator select any single CD or CIA rod to be moved.
3. Communicate with the engine controller to set/control parameters
4. Control the engine cooling loops
5. Calculate Reactor Power from neutron detector sensors
6. Calculate Reactivity/Period from neutron detector sensors
7. Calculate Pressure in the Reactor Vessel and the Guard Vessel from pressure sensors
8. Calculate Reactor Temperature from temperature sensors
9. Indication that a NaK leak has occurred
10. Control heater elements to warm NaK using non-nuclear heating
11. Indicate if a seismic event occurred
12. Interface with the HMI
13. Perform additional DID measures as applicable in the software.
14. Record data

The control system is controlled with a computer program. The control systems interfaces with signals and communication of nearly all sub-systems inside of the I&C and RCS systems. The boundary between the control system and the sub-systems is considered the wire between them with the wire belonging to each sub-system. The network connection is considered part of the control system.

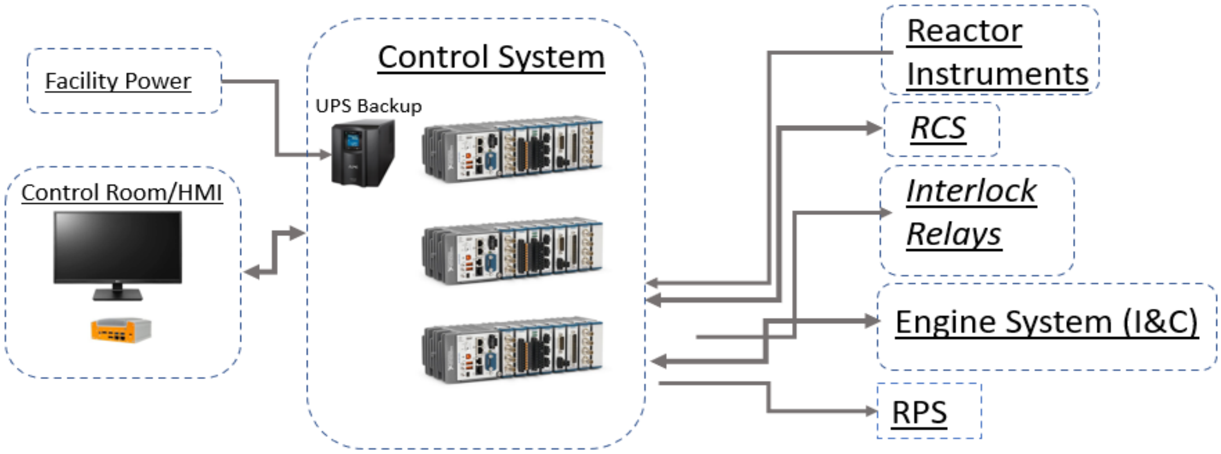

Figure 52. Control system boundaries.

3.6.3.2 Reactor Instrumentation. The reactor instrumentation (Figure 53) provides inputs to the control computer. The reactor instrumentation is NSR-AR with the exception of the pressure sensors which are SR. The functions of the reactor instruments are to provide the following to the Control System:

1. Thermocouples
   1) Signals proportional to temperature
   2) Signals indicative of the steady state reactor power
2. Neutron Detectors
   1) Sense neutron levels from source level to full power
   2) Dynamic indication of the reactor power
3. NaK leak detectors
   1) Signal triggered when in contact with liquid metal
   2) Detect a NaK leak into the guard vessel
   3) Detect if the NaK that has leaked into the guard vessel and has reduced the NaK inventory in the primary to uncover the core
   4) Detect if NaK has liked from the IHX into the secondary
4. Pressure Sensors (SR)
   1) Provide a signal proportional to pressure over the entire expected operational and transient pressure ranges
   2) Detect if the pressure is out of normal bounds to indicate the pressure differential between the guard vessel and the primary vessel has been compromised and the pressure cannot adequately perform its safety related function of maintaining the core covered in a LOCA accident.
   3) Provide pressure indication to operators
   4) Provide post-accident monitoring. Indication is on the analog meters and are not part of the control system.
5. Heaters
   1) Supply non-nuclear heat

The instruments sense physical phenomena like neutrons, pressure, and temperature. The leak detectors operate on the continuity principle and the fact that NaK and GaInSn are conductive. The welds and interfaces between the primary vessel or guard vessel and the instruments are SR. The pressure transducers are SR as well as the analog meters. The rest of the reactor instruments are NSR-AR. The reactor instrumentation also includes a heater element and a controller.

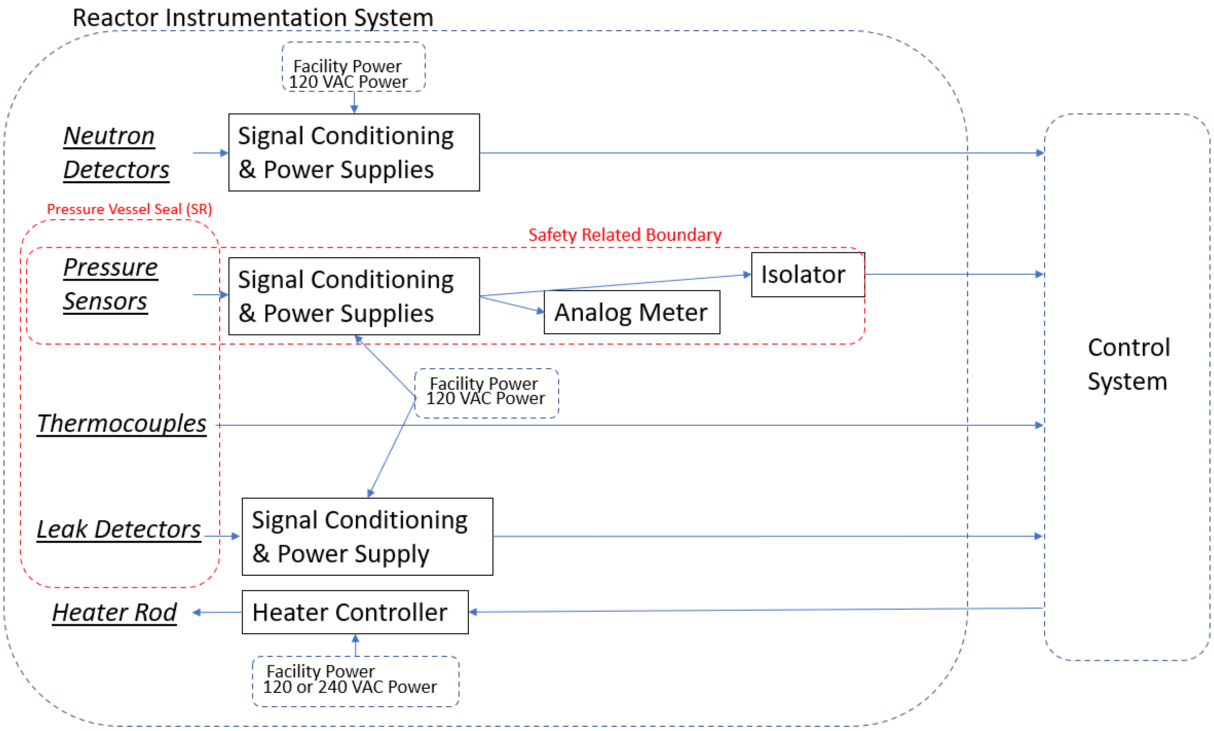

Figure 53. Reactor instrumentation system boundaries.

3.6.3.3 Engine System (I&C). This system (Figure 54) is the interface between the electrical generation system and the control system. None of the components are SR. The engines start after being heated and will stop when the heat diminishes. The cooling loops are turned ON/OFF by the operators and force coolant to circulate. Fans are also on a heat exchanger outside the building to dissipate the heat with the atmosphere.

There are no safety functions in the Engine System I&C. The system is NSR. The functions of this system are:

1. Provide signal I/O with engine cooling loop to cool the Stirling engine
2. Communicate with the engine controller to read/set parameters
3. Provide calorimetric information during steady state calibration of the reactor power for the neutron detectors

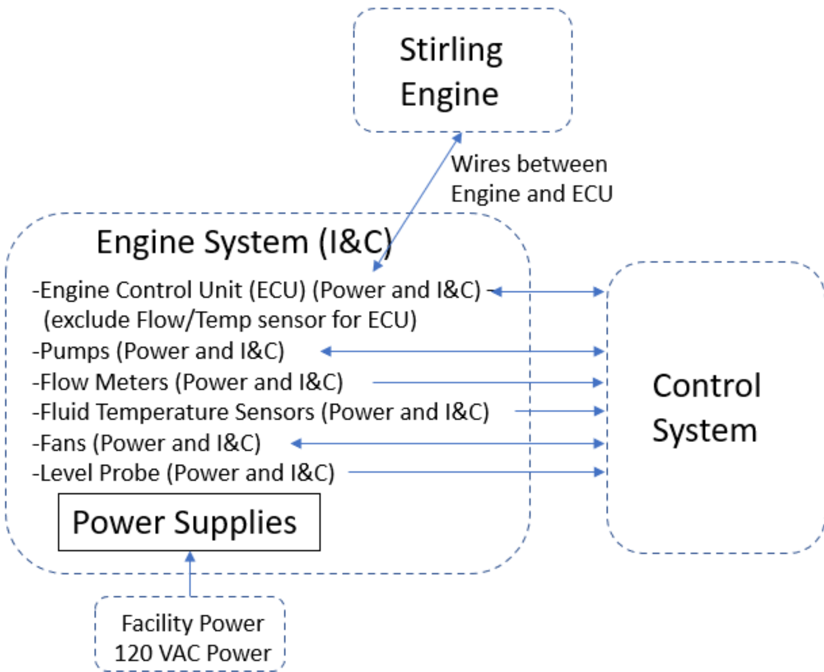

Figure 54. Engine system (I&C) boundaries.

3.6.3.4 RCS System. The RCS (Figure 55) systems function is to actuate the drums and rod. The system also has provisions to allow the drums and rod to return to the shutdown automatically with any scram event. A bulk of the SR SSCs are found in this system.

The RCS has both SR and NSR-AR functions. The RCS performs the following basic functions:

1. Insertion capability using stored energy (SR)
2. Provide reactivity insertion and magnitude control (NSR-AR)
3. Provide indication of the position of the drums and rod (NSR)

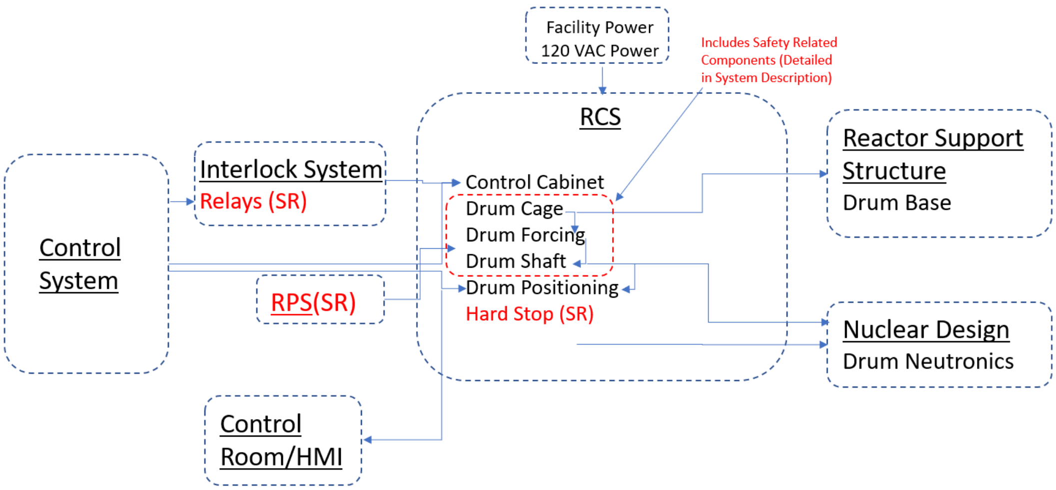

Figure 55. RCS boundaries.

3.6.3.5 Interlocks. The interlocks (Figure 56) are found in two locations and one in procedures. They are a set or relays that prevent more than one drum/rod from moving and mechanical hard stops to set allowable excess reactivity limits on each drum. The procedures also are used to prevent more than one drum from being rotated above the critical configuration with an appropriate amount of tolerance.

The interlocks have safety related functions. The interlocks perform the following basic functions:

1. Reactivity magnitude control by limiting one drum or rod to move at a time. (SR). The CIA rod selection is also part of the interlock circuit but is NSR-AR. However, the CIA rod selection functions just like another drum selection.
2. Reactivity magnitude control by setting of a hard stop to limit the maximum position of a drum. (SR). The CIA rod is also limited by a hard stop but is NSR-AR. The hard-stop is physically part of the RCS system but is an interlock concept.
3. Reactivity magnitude control by not allowing operators to move more than one drum above the critical position. (SR).

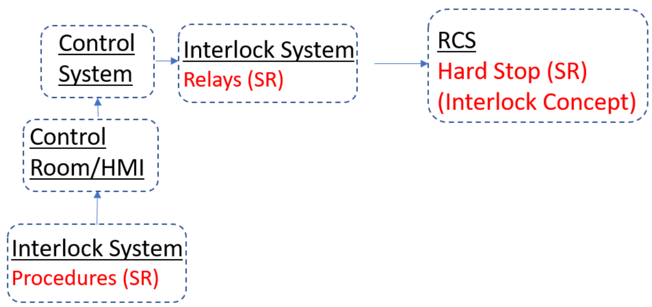

Figure 56. Interlock boundaries.

3.6.3.6 RPS. The RPS (Figure 57) is a set of relays that when not actuated will released the power to electromagnet clutches in the drums and rod. Releasing the clutch will allow the drum or rod to be sent back to the shutdown condition by means of a spring in the case of the drums or gravity in the case of the rod. The RPS is housed in the RCS cabinet. The SR parts of the RPS are the seismic and manual scrams and any associated part in its path. All other parts are NSR-AR.

This system has safety related functions. The system performs the following functions:

1. Reactivity insertion capability using stored energy. (SR) The RPS is comprised of a series of relays with activate the electromagnetic connections near the actuating motor in the RCS system. The electromagnets link the actuator to the CDs and CIA and this connection is lost upon the deactivation of any of the relay inputs. The subsequent release allows the now decoupled CDs and CIA to be driven into their default shutdown state via a spring or gravity respectively.
2. Provide reactivity insertion from manual scram. (SR) The depressing of the manual scram button disconnects a set of redundant relays in the RPS system to allow the RCS to move the drums and CIA to the shutdown position.
3. Provide reactivity insertion from a seismic sensor. (SR) The seismic sensor disconnects a set of redundant relays in the RPS system to allow the RCS to move the drums and CIA to the shutdown position.
4. Provide reactivity insertion from computer trips. (NSR-AR) The control system can provide a signal based on a criteria to cause a reactor scram which disconnects a set of redundant relays in the RPS system to allow the RCS to move the drums and CIA to the shutdown position.
5. Provide reactivity insertion from a loss of power event. (SR) The relays in the RPS system disengage with loss of power allowing the RCS to move the drums and CIA to the shutdown position by means of a spring and gravity, respectively.

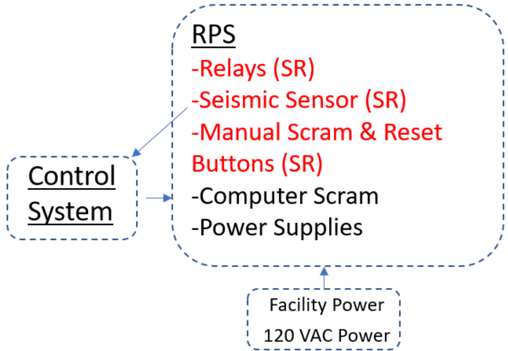

Figure 57. RPS boundaries.

3.6.3.7 Facility Monitoring. The facility monitoring systems (Figure 58) are designed by the facility. However, the equipment used by the facility monitoring provide useful information to operators in case of emergencies or accidents. For this reason, this equipment is being mentioned. These sensors relay information to the control room for display. The Control Room/HMI system does not include these functions readouts as part of the system. The facility monitoring is considered NSR-AR because of its functionality for post-accident monitoring.

This system is NSR-AR. The functions provided by this system are:

1. Readout of the radiation levels using Radiation Area Monitors (RAMs) and Continuous Air Monitors (CAMs) and provide indication to Radiological control in the control room. This function is also part of post-accident monitoring.
2. Provide indication if smoke is present outside of the reactor. If smoke is present, that is indication of a NaK leak.

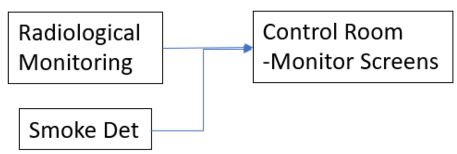

Figure 58. General diagram of facility monitoring.

3.6.3.8 Facility Power. The facility power interfaces wherever power is needed. This in not considered a system and has no safety related functions. The functions of the facility power are to provide:

1. Standard 110VAC power
2. Standard 220VAC power
3. Standby power for equipment protection (i.e., diesel generators)
4. Bridge power between the loss of facility power and the starting of standby power. (i.e. uninterrupted power supply UPS)

3.6.3.9 Control Room/HMI. The Control Room/HMI general diagram can be seen in Figure 59. The control room/HMI interfaces with the control system, the RPS system at the manual scram button, facility monitoring and indicators from RCS for the LED lights. All the Control Room/HMI system parts are NSR or NSR-AR.

The control room has a mix of analog and computer based display. The computer based HMI has no safety related functions. The analog gauges and LEDs are used for post-accident monitoring. The analog pressure gauges are credited for verifying the pressure is within the expected operational bounds and are SR.

Figure 59. Graphic of control room with MARVEL screens and analog meters.

The functions of the control room/HMI (Figure 60) are to provide:

1. Interfaces between the operators and the control system
2. In particular the analog pressure meters and LED lights for the shutdown position are used for post-accident monitoring (NSR-AR)
3. Provide warnings and alarms

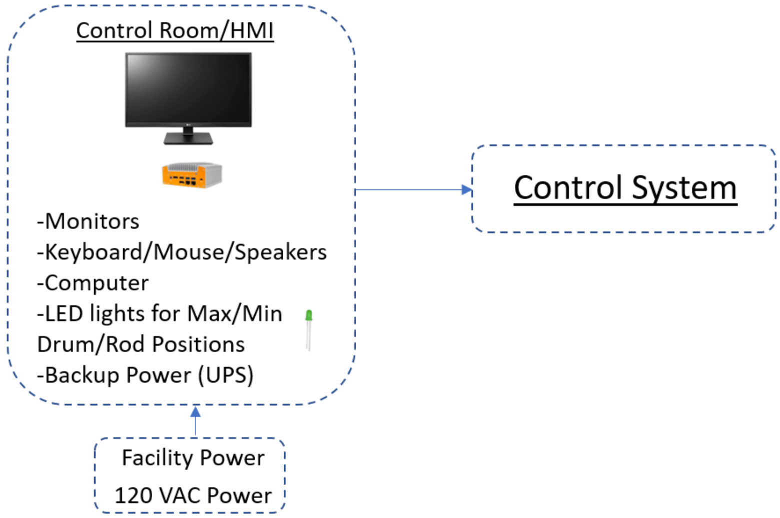

Figure 60. Control room/HMI boundaries.

3.7 Power Generation System (PGS)

3.7.1 Design Description

This section describes the MARVEL Power Generation System (PGS). Figure 61 as taken from TFR-2575, shows the overall system architecture of the PGS and how each of the subsystems interface. Boxes surrounding the system boundary represent interfacing subsystems not within the scope of the PGS. Red lines represent thermal interfaces, yellow lines represent electrical interfaces, teal lines represent instrumentation and control (I&C) interfaces, and black lines represent important mechanical or structural interfaces.

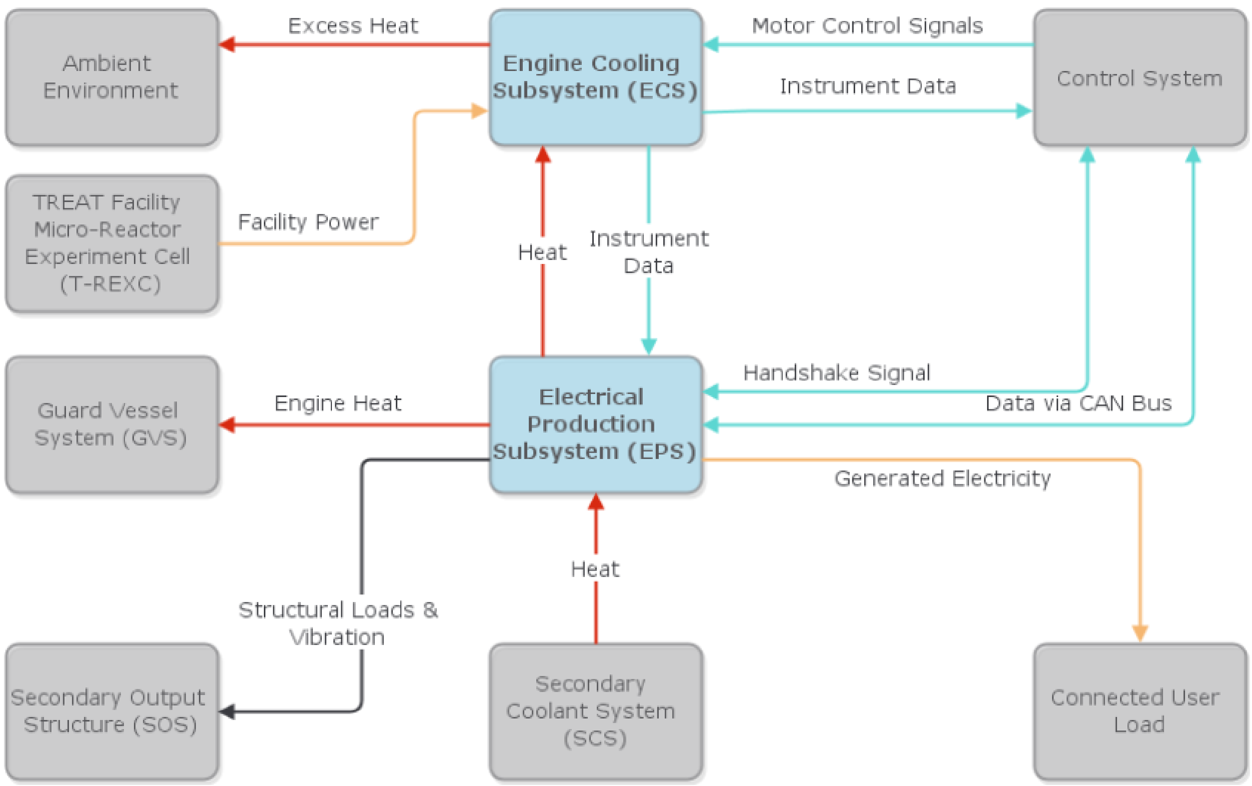

Figure 61. Power generation system architecture and interfaces.

The Electrical Production Subsystem (EPS) consists of the stirling engines and their associated engine controllers. The engines absorb high-grade heat from the SCS media through passive conduction when not actively operational due to the natural circulation of the primary and secondary coolants. Once activated via a DC impulse, the engines extract additional heat that is absorbed into the closed power cycle of the engines. This heat is converted first into pressure variation within the working media, then into sympathetic, concurrent motion of the resonant internal components, and finally into either of low-grade heat from the power cycle or gross electric AC power delivered to the engine controllers. Low-grade heat is removed by actively circulating engine cooling loops. Engine position and frequency data is transmitted to the engine controller to adequately monitor and control the unit. The engines are mounted to the Secondary Output Structure (SOS) which is designed to dampen the vibration caused by the inertial force of the engines.

Gross electrical power produced in the engine is then conditioned by the engine controller according to the unique user loads present. Output power can be conditioned to be DC or AC, frequency may be selected for AC loads, and voltage may be chosen for each load. The engine controller is also capable of modulating the SES in order to dynamically follow end user loads. Any excess gross electricity is converted back into thermal energy and removed by the engine cooling loop. The engine controller also serves as the primary communication mechanism between the engine and the control system using controller area network (CAN) bus protocols and an e-stop “handshake” signal. The CAN bus interface is used to command the power generation equipment from the control system and to report back engine and conditioned electrical output parameters. The bi-directional e-stop is used to sync operational readiness between the PGS and the control system. The signal from the engine controller to the control system confirms that the engine has passed all pre-operational health checks and that the engine is operating as expected. The signal from the control system to the engine controller confirms that the reactor subsystems are ready for active heat extraction from the secondary coolant.

Independent Engine Cooling Subsystem (ECS) loops circulate a coolant through each set of engine and engine controller to remove heat from the power generation process. These inner coolant loops, fully located within the TREAT Facility, then transfer heat to outer loops that reject heat to the ambient environment outside the TREAT Facility building envelope. Control of the engine coolant loops is performed by the control system, which provides the ability to modulate the pump and heat rejection unit (HRU) motor speeds. Select data from the inner cooling loops is provided to the engine controller to support the health checks prior to engine start up. All instrument data is provided to the control system so that it can perform overall MARVEL heat balance / calorimetry calculations.

3.7.2 Design Bases

The PGS SSC safety classifications are provided in Appendix B. There are no key safety functions for PGS SSCs.

3.7.2.1 Design Criteria. PGS SSCs shall meet the overall principal design criteria (PDC) 1-5 in Appendix A.

3.7.2.2 Functional and Operating Requirements. The requirements and bases for the ICS are discussed in detail in FOR-868 and TFR-2575. Key system functional requirements are as follows:

- The PGS shall be capable of extracting heat from the secondary coolant.
- The PGS shall produce electrical power from heat provided by the secondary coolant.
- The PGS shall be capable of distributing net electrical energy to user loads.
- The PGS shall reject excess heat or electricity from the power conversion cycle.

Key system operational requirements are as follows:

- PGS SSCs within the secondary confinement boundary shall be constructed of materials designed to withstand the credited environmental conditions (e.g., temperature, pressure, radiation, etc.) in which they are installed.
- Each PGS train shall provide at least 90% availability during its expected operational life.
- The PGS shall have sufficient controls to interface with other systems for startup, shutdown and power ramping.

3.7.3 Description of Systems, Subsystems, and Major Components

3.7.3.1 Qnergy QB80 Power Convertor. MARVEL's baseline PCRS implementation is based on a QB80 to provide both HTHX+HCS functions. Qnergy's 80-tube power convertor provides both function groups within one component. QB80 has heat exchanger (tubes) surface area to extract high temperature heat from the SCS. Phased motion of the two internal Power Convertor components ultimately converts absorbed heat into useful electric power. An image of QB80 exterior is shown in Figure 62 and a schematic of the generalized internals are shown in Figure 63 below. The illustration clearly separates the two moving internal subassemblies and shows sub-system boundaries relevant to QB80 within PCRS.

Figure 62. Qnergy PCK-80.

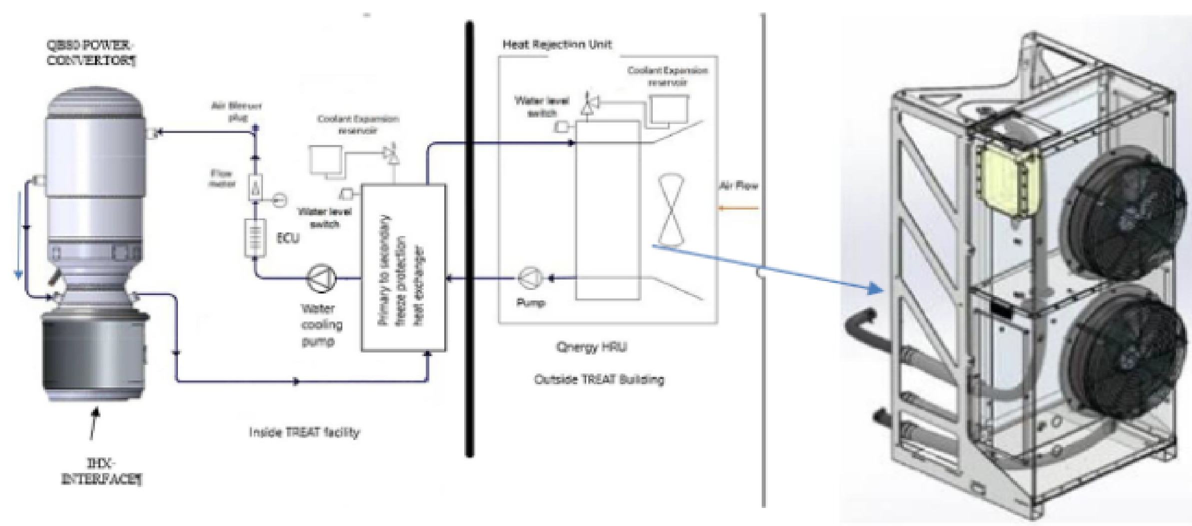

Figure 63. Power conversion and heat rejection equipment.

Qnergy power convertors are based on a gamma-configuration, pseudo-Stirling implementation, and the two internal moving assemblies are supported, isolated, and aligned using flexible plate bearings (flexures). Flexure-bearing machines minimize potential for component rubbing contact, and ensure long, maintenance-free service lifetime for Qnergy hardware.

The MARVEL heat source is designed for natural flow of coolant in both the primary and secondary cooling circuits. The HTHX+HCS provide the cooling required to drive natural flow. The physical position of the QB80's, at the top of the assembly, ensures that cool fluids will fall due to gravity. Heating from the source happens down low, resulting in warm fluids 'naturally' rising back up to ensure circulation of the coolant. Individual heat engines vibrate during operation, which may additionally enhance flow of the coolant through paths of asymmetric directional pressure drop.

3.7.3.2 Qnergy Engine Controller (QEC). Qnergy power convertors are 'free-piston' architecture, which means no crankshafts, and no mechanical wearing parts. Motion of the two internal moving assemblies is controlled and maintained electromagnetically. The 'engine controller' maintains this control while also removing gross electric power from the HCS function and delivering that as net, conditioned electric power to the user load. Terminal voltage at the QB80 connection 'runs wild,' with minimal regulation of voltage or frequency of operation. Power export to the user from QEC is conditioned and controlled to the user's specific interface requirements.

3.7.3.3 Qnergy Heat Rejection Units (HRU). Equipment is required to deliver waste heat to the ultimate ambient heat sink (air). The Qnergy HRU provides modular LGHX function, removing heat from the PCK. The MARVEL initial design is based on directly implementing equipment from Qnergy PowerGen hardware, since that equipment is already sized for PCK and has a well-known field service record at customer sites. LGHX HRU equipment is basically a set of pumps and radiators and may optionally be reconfigured in the future, without meaningful impact to the rest of the MARVEL installation.

Figure 64, Figure 65, and Figure 66 show the major systems relationships. These diagrams break each box out of the overall system architecture and break them down into lowest-level form.

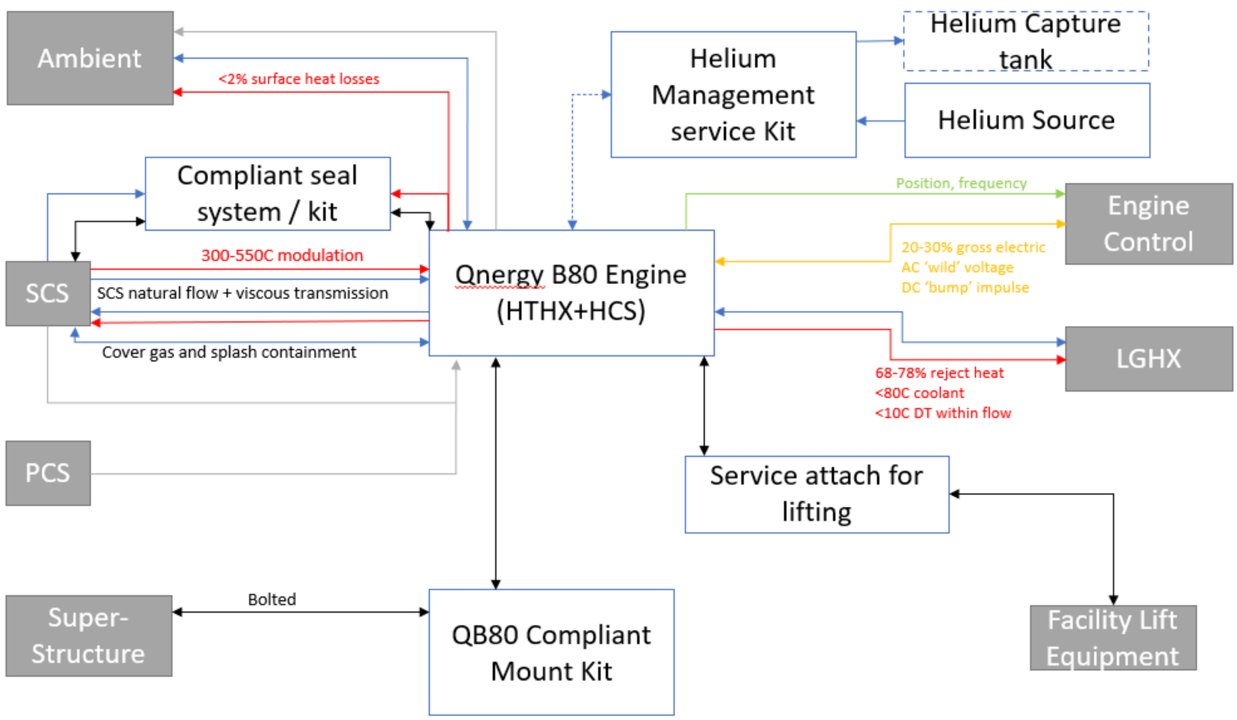

Figure 64. QB80 (HTHX+HCS) subsystem diagram.

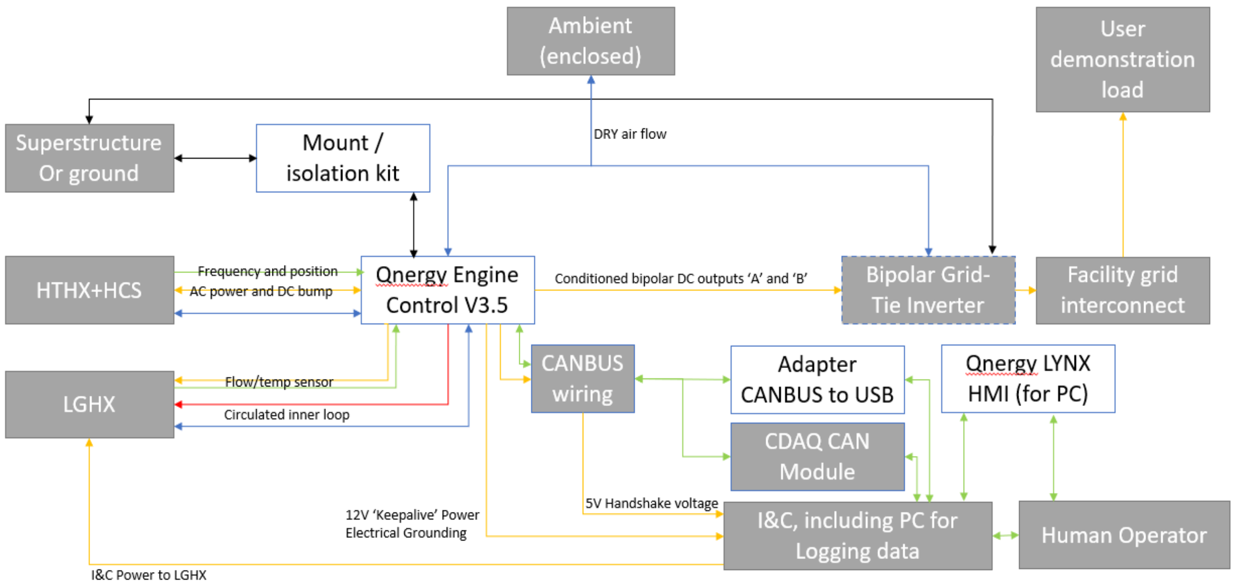

Figure 65. Engine Control (QEC) subsystem diagram.

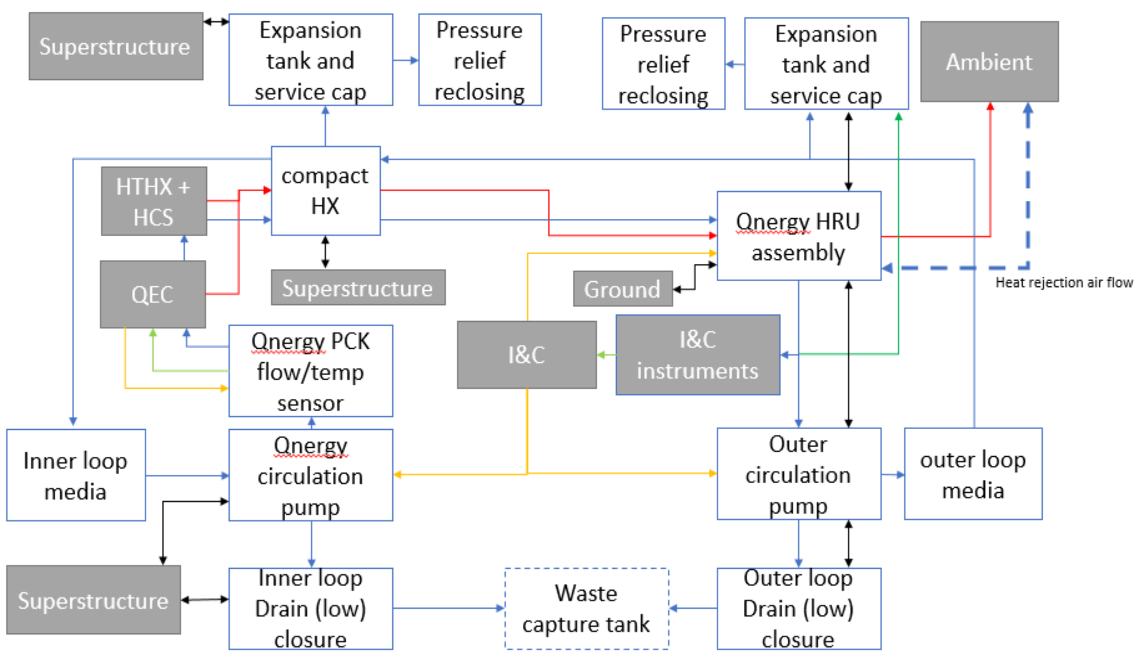

Figure 66. Heat Rejection and Cooling (LGHX) subsystem diagram.

3.7.3.4 Electrical Production System (EPS). The electrical production system consists of two major components, the engine and the engine controller. The engine connects directly to the engine controller. The wires between the engine and controller are for the electrical power and signals from two hall sensors on the engine. The controller is called the Qnergy Engine Controller (QEC) and it performs all the control functions. The operator can communicate with the QEC by means of a Control Area Network (CAN) bus. The QEC has water flowing through it because it has a built-in load-bank to dissipate any unused electricity as well as a means of adjusting the engine amplitude. The engine oscillates at a fixed frequency near 60 Hz and the only variation is the amplitude.

Th process of starting the engine is called bumping. Once the engine has “bumped” successfully the frequency will ramp to the operational frequency. The operators have two setpoints that are used to control the engine. These two setpoints are not independent and thus there is a priority on the setpoints. The first setpoint is the Heater Head Temperature (HHT). This is an estimated temperature that the engine is seeking. The next setpoint is the electrical power level. The electrical power setpoint is subservient to the HHT setpoint. If the HHT setpoint is higher than the estimated temperature than the amplitude is decrease in an effort to build up heat. If temperature is lower than the HHT setpoint then the amplitude is increase in order to pull more heat. Once the HHT setpoint is close to the estimated temperature, then the power setpoint takes over.

It is worth noting that the heater head temperature is not directly measured, instead it is inferred. The only signal inputs to the QEC are flow, temperature of the water line and the hall sensors. The QEC is designed based on the readings from a specific flow and temperature sensor that is required to be installed.

3.7.3.5 Engine Cooling System (ECS). The primary purpose of the engine cooling system is to cool the engine. The ECS also has a major function in the power calibration method. Each engine has an independent set of cooling loops. There are two loops per engine with the first being water and the second loop being glycol. The purpose of the water loop is to keep the loop from exiting the building for potential contamination issues. Glycol will also break-down in a radiation environment. The glycol loop exits the building so that the heat can be dissipated to the environment.

The major components needed to fulfil the purpose of heat extraction are pumps on each loop, a reservoir with a radiator cap for overprotection, a heat exchanger between the water and glycol loops, and a radiator with fans on the glycol loop. However, there are other component added for the power calibration function. These include resistance temperature detectors (RTD) and a flow meters with better accuracies than the flow and temperature sensor required by the QEC. The RTD’s are placed in positions to measure the heat input across the engine and the engine controller for measuring the heat balance.

4. T-REXC DESIGN DESCRIPTION

4.1 TREX-C Design Description

The project to establish T-REXC will equip the TREAT reactor building (MFC-720) pit (see Figure 67) with the following SSCs to fulfill the indicated functions for any supported microreactor:

- An electrical panel, to provide normal AC power to support the following:

  Temperature control equipment

  Load handling equipment

  Facility general loads

  Reactor startup and operation

  Instrumentation and controls.
- A backup AC power system to supply backup power (interrupted) to protect experiment assets/resources.
- A simple set of generic instrumentation and controls, including a panel installed into the TREAT control room (MFC-724), to provide basic instrumentation and control (I&C) services.
- Seismic and manual trip circuits as safety related SSCs/signals.
- SSCs to detect low intensity seismic ground motions (p-wave) and initiate a reactor trip signal.
- Supplied plant air from the TREAT plant air system, to support non-safety actuation devices.
- A heat rejection system that will transfer heat to the outside environment, which is separate from the TREAT HVAC system.
- A filtered ventilation system that establishes a continuous airflow pattern from the outside T-REXC environment (Either outside the TREAT building or from the TREAT Highbay) into T-REXC from noncontaminated areas to potentially contaminated areas through HEPA filters to the outside environment, which is separate from the TREAT HVAC system.
- TREX-C ventilation system exhaust monitoring, which is separate from the TREAT HVAC system, to ensure compliance with the requirements of 40 CFR 61, Subpart H, "National Emission Standards for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities."[26]
- A radiation monitoring system, to monitor radiation fields and any released contamination emitted from the microreactor. [NOTE: This system likely will include duct detectors in the T-REC ventilation systems to provide indication of combustion events in either the Upper Confinement or T-REXC pit]
- Control room equipment to remotely monitor and control T-REXC facility equipment, as needed.
- A fire detection system, to detect fires at or near the microreactor and send alarm signals to the on-site fire station.
- A fire barrier if determined to meet HAD-470, "TREAT Fire Hazard Analysis,"[27] requirements [Note: an exemption from the requirement to have a fire barrier has been prepared and if approved, this SSC will not be installed].
- A pit cover/shielding system, to serve as an upper radiation shield and to provide access to the Stirling engines.
- Radiation shielding, to line the pit walls and protect the pit concrete from radiation-induced degradation.

- An equipment pad and gas manifold outside the building, to support microreactor-specific systems.
- BeO control drum components and reflector panels, to be incorporated into microreactors installed into T-REXC.

The T-REXC project will equip the pit with the above SSCs to accommodate the quickest possible testing of small microreactor designs and associated critical tests. The installation of T-REXC SSCs will be addressed in separate TREAT engineering change (EC) documentation from that of MARVEL. Installation of MARVEL into T-REXC (see Figure 68) and application of T-REXC SSCs specific to MARVEL will be addressed in the safety bases documents to be submitted to DOE for approval as described in SDS-119, "Safety Design Strategy for the Microreactor Applications Research Validation and Evaluation (MARVEL) Project."[28]

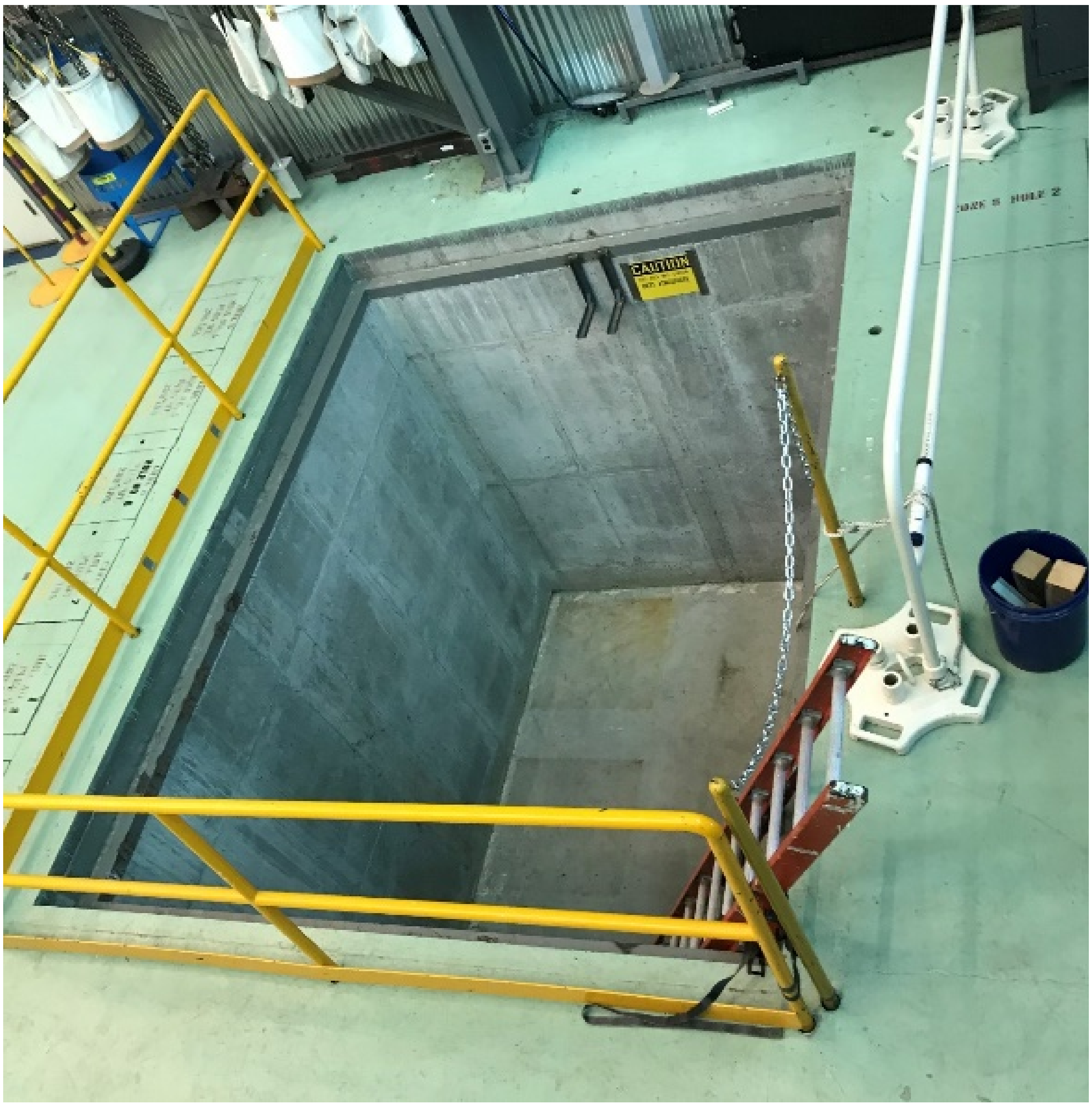

Figure 67. Photo of the TREAT north high bay equipment pit where T-REXC will be located.

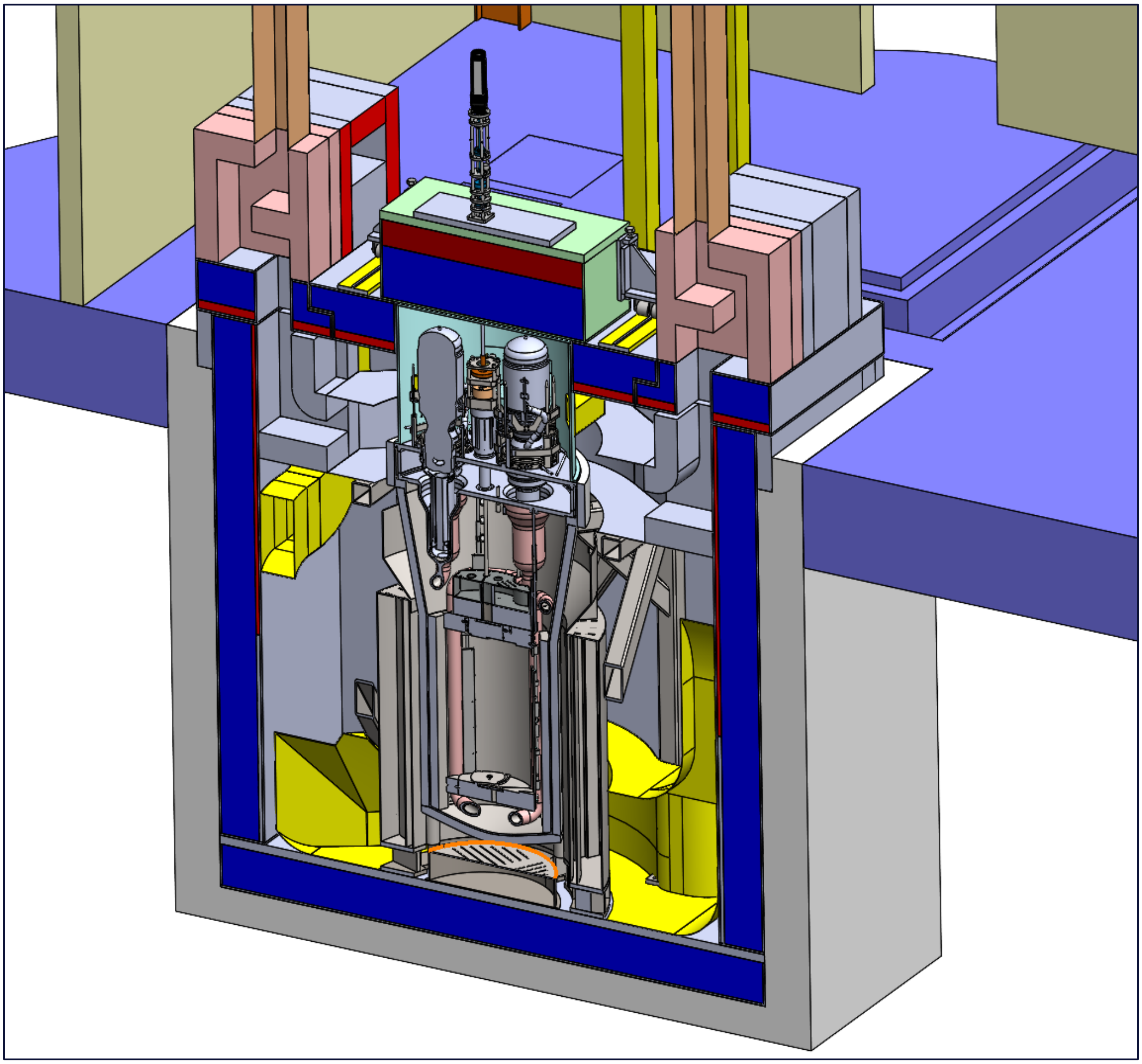

Figure 68. Rendering showing MARVEL configuration inside T-REXC.

4.2 Design Bases

The T-REXC SSC safety classifications are provided in Table C-. Key safety functions for T-REXC SSCs include:

- Reactivity control – passive IRF
- Heat removal – passive heat rejection
- Reactivity control – CD insertion
- Control direct radiation exposure – shielding
- Confinement of radioactive and hazardous material release – fission product barrier
- Heat removal – passive heat rejection.

4.2.1 Design Criteria

T-REXC SSCs shall meet the overall principal design criteria (PDC) 1-5 in Appendix A.

4.2.2 Functional and Operating Requirements

The requirements and bases for the T-REXC are discussed in detail in FOR-684. Key system functional and operational requirements are as follows:

- Each component or assembly being installed in T-REXC shall have a weight limit of 27,000 lbs. (including shielding and rigging equipment).
- Demonstration micro-reactors shall be installed in a 10 ft L × 7 ft W × 9 ft H space constraint.
- An air-cooling system shall be provided with the capability to remove 10kW of heat during operations.
- The average internal air temperature of the micro-reactor area shall be maintained at or below 40°C (104°F) during reactor operation.
- The internal air temperature of the micro-reactor area shall be maintained above 0°C (32°F).
- A micro-reactor heat rejection system shall transfer a minimum of 150 kW from the demonstration reactor to the outside environment (excluding air cooling capacity).
- All ventilation exhaust from the micro-reactor area shall pass through HEPA filters prior to being released to the outside environment.
- The ventilation system shall discharge the filtered effluent from the micro-reactor area through a monitored stack where both the stack and monitoring system are compliant with ANSI N13.1.
- T-REXC systems shall be designed with sufficient margin to assure safe micro-reactor shutdown upon loss of normal power, without needing a separate electrical power source (e.g., backup power). (2 kW decay heat for 48 hrs. - The current backup plan is to provide a blower system on a UPS. This is subject to change)
- A minimum capacity of 50KW for a minimum of 48 hours shall be allocated from the existing TREAT backup power supply (interrupted).
- Reactor controls and monitoring circuits shall be physically isolated (air gapped) from networks external to TREAT.
- T-REXC Instrumentation and Control (I&C) equipment in the TREAT Reactor Building (MFC-720) shall be coordinated with TREAT facility management.
- T-REXC shall detect low intensity seismic ground motions (p-wave) and initiate a reactor trip signal.

- TREX-C dedicated post-accident monitoring (PAM) signals shall be provided in the TREAT control room. At a minimum, the following signals will be required for the T-REXC facility:
- Each demonstration micro-reactor shall be designed such that the bounding radiological and non-radiological postulated consequences will be within evaluation guidelines.
- Facility equipment shall not be damaged from radiation exposure during and after operation of a demonstration micro-reactor.
- The INL criticality safety program, LRD-18001, shall be followed.

4.2.3 T-REXC Interface Requirements

Support and infrastructure systems are at times required for safety system operation. Major facility interfaces between the T-REXC and MARVEL project that are critical to design and the safety design basis have been identified in SDS-119 as follows:

<u>The MARVEL safety interfaces with T-REXC are</u>:

- The toxicity hazards with the introduction of BeO and Be metal components into the T-REXC for storage prior to their incorporation into MARVEL shall be evaluated.
- The potential neutron moderation of BeO and Be metal components into the T-REXC for storage prior to their incorporation into MARVEL shall be evaluated.
- The standard industrial hazards with the introduction of electrical and mechanical equipment and process gases, and the toxicity hazards with liquid metal coolants (NaK and eutectic Galinstan [R]) shall be evaluated.
- MARVEL SSCs with the potential to interact with T-REXC SSCs during a seismic event shall be designed to the same safety design category for seismic events and shown by seismic analysis to not interfere with any T-REXC or TREAT building SSCs.

<u>The T-REXC safety interfaces with T-MARVEL are</u>:

- T-REXC upper confinement HVAC and filtration systems shall be capable of adequately filtering postulated airborne radioactive material releases during normal operation or accident conditions.
- T-REXC pit HVAC system shall be capable of adequately remove decay heat during normal operation or accident postulated conditions.
- T-REXC radiation monitoring systems shall be capable of monitoring direct radiation fields and any released contamination emitted from MARVEL.
- T-TREXC fire detection system shall be capable of detection of fires at or near MARVEL and send alarm signals.
- T-REXC pit lid shall adequately serve as an upper radiation shield for protection of the immediate worker.
- T-REXC pit radiation shielding shall line the pit walls and protect the pit concrete from radiation-induced degradation.
- T-REXC SSCs have been identified as systems with the potential to interact with MARVEL systems during a seismic event. These potential interactions will be evaluated further and mitigated based on the guidance presented in in Section 6.3.2.4, “System Interaction,” of American National Standards Institute (ANSI)/American Nuclear Society (ANS)-2.26-2004, “Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design.”[29]

The above interfaces of MARVEL SSCs installed into the T-REXC pit shall be evaluated in the MARVEL preliminary Documented Safety Analysis (PDSA).

4.3 Description of Systems, Subsystems, and Major Components

4.3.1 T-REXC Shielding System

The shielding concept mentioned involves using a combination of materials to reduce the radiation emanating from the reactor. The materials in the shielding system serve to either moderate (reduce the energy of) neutrons, or to absorb gamma radiation, both of which can be harmful if not properly contained. Here's a more detailed explanation of the shielding concept:

Water-Extended Polyester (WEP): WEP is a type of material that is essentially a polymer (the polyester) that has been extended or diluted with water. It's used here for neutron moderation, which is the process of slowing down fast neutrons by scattering them off of light atoms or molecules, reducing their energy and making them less penetrative. This is crucial in a reactor setting because fast neutrons can cause more damage and are more likely to cause materials around the reactor to become radioactive (neutron activation). Water is an excellent neutron moderator because of its light hydrogen atoms. By incorporating water into polyester, the WEP acts as an effective and moldable neutron shield. This shielding material is placed in steel cans to provide structural support and additional gamma shielding.

Steel/Lead for Gamma Attenuation: Gamma rays are high-energy photons (light particles) that are often produced in nuclear reactions. They are highly penetrative and can cause ionization in the tissues of living organisms, potentially leading to harmful effects like radiation sickness or an increased risk of cancer. Steel and lead are heavy, dense materials that are very effective at absorbing gamma radiation. They work by interacting with the gamma rays and reducing their energy to a level where they are no longer harmful. The shielding concept incorporates these materials to provide effective gamma radiation protection.

Shielding Placement: The shielding materials are placed in prefabricated steel cans. These cans are likely chosen because they can be easily manufactured and transported to the reactor site, where they can then be quickly assembled to create the shielding system. The cans are then used to line the interior of the pit, which is the area surrounding the reactor. This provides protection for the surrounding TREAT infrastructure, preventing it from becoming radioactive and reducing the gamma dose to safe levels. In addition to lining the interior of the pit, the top lid of the pit is also lined with these cans. This is important as it helps to contain radiation within the pit and prevent it from escaping into the TREAT high Bay .

Shield Configuration: Figure 69 provides a visual representation of this shielding concept. It would illustrate the placement of the steel cans filled with WEP and steel/lead in the pit's interior and on the top lid. By reviewing this figure, one can better understand how the shielding system is arranged and how it works to protect the surrounding infrastructure.

Please note that T-REXC shielding is a conceptual design at this stage, which means it's an initial idea or proposal for how the shielding system could be implemented. It may be subject to further changes and refinements as the project progresses.

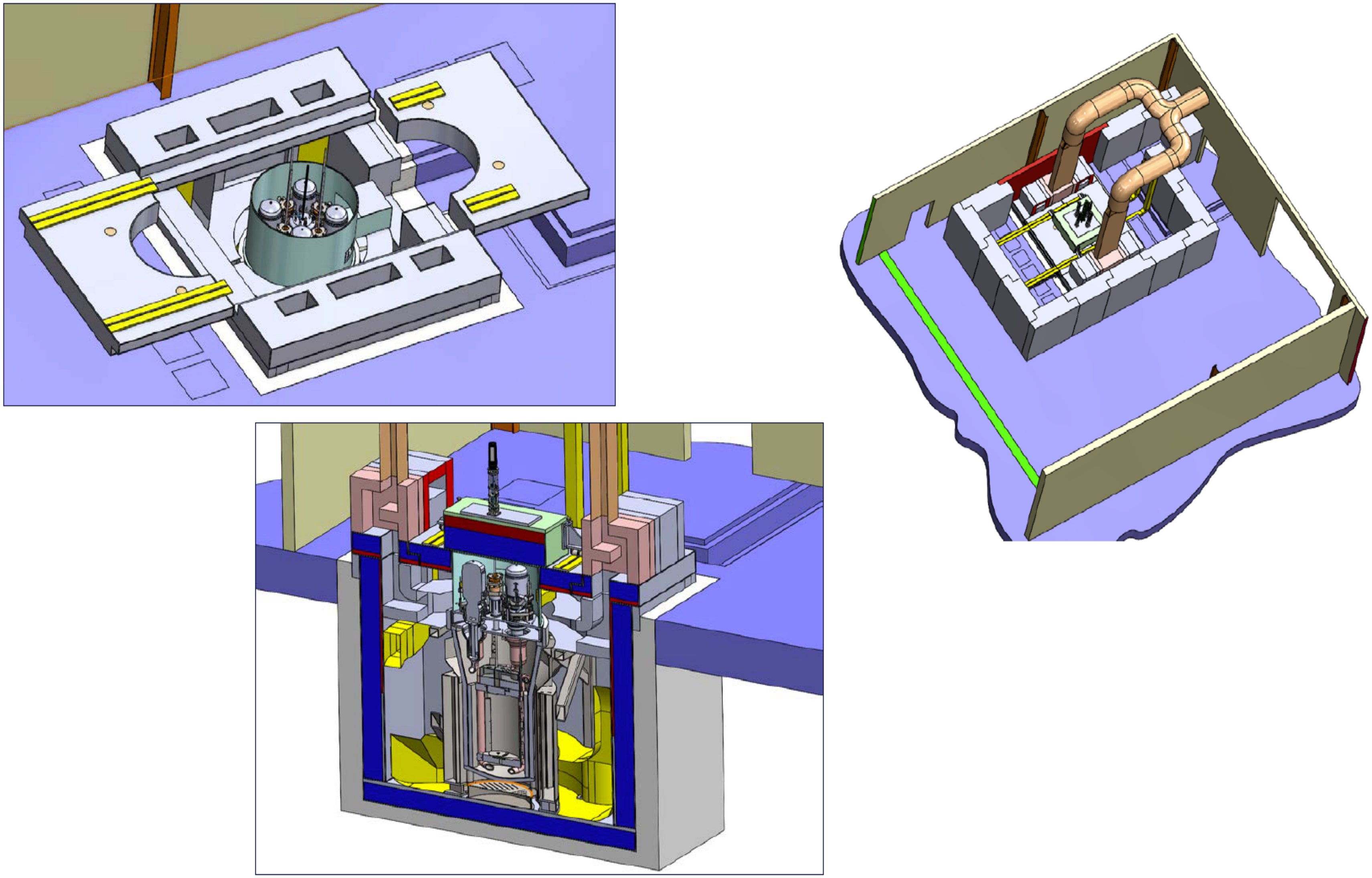

Figure 69. T-REXC shielding concept.

4.3.2 T-REXC Ventilation System

The design of the ventilation system is conceptual at this stage.

For the MARVEL Micro-Reactor, T-REXC offers two primary HVAC systems, namely the Upper Confinement HVAC and the Main HVAC, as illustrated in Figure 70. The Upper Confinement HVAC system's chief function is to extract heat from the Upper Confinement. This is achieved by employing forced airflow, wherein air is drawn into the system. The incoming air is conditioned to maintain desired temperature and humidity levels within the space. To ensure air quality, the system is equipped with HEPA filters that purify both the supply and exhaust air. This system is also adaptable for maintenance activities, providing clean and conditioned air to maintenance personnel. The heat removal requirement for normal operations of this system is up to 10 kW of continuous heat removal, while no decay heat removal is required. HEPA filters in the supply and exhaust streams address the filtration requirement for the Upper Confinement. This approach is particularly important due to the absence of differential pressure, which is expected due to the desired temperature gradient.

The Main HVAC system, on the other hand, is responsible for heat removal from the main pit. The system uses both forced airflow and natural drafting to achieve this, with the drafting function considered a Safety Related (SR) feature and is predominantly used for decay heat removal. The incoming air is conditioned as per the desired parameters. Filtration, in this system, is aimed at controlling dust and bugs in accordance with FMEA, but HEPA filtration is not mandated. Additionally, this system has an exhaust stack to release air from the pit. For integration with the MARVEL Tertiary Cooling Annulus, three primary options are under consideration: no plenums, a plenum either to the inlet (Option 2a) or outlet (Option 2b) of the Tertiary Cooling Annulus, and plenums to both the inlet and outlet of the Tertiary Cooling Annulus. The heat removal requirements for normal operations are up to 10 kW of continuous heat removal, while for decay heat removal, it's 2 kW via natural convection for the first 48 hours, reducing to 1 kW for extended periods, as specified in the ECAR-6332 RELAP5-3D Thermal-Hydraulic Analysis of MARVEL Micro-Reactor - Final Design Sections 4.3.3 and 4.3.8. The filtration requirement is solely for FMEA control of dust and bugs; no HEPA filtration is necessary. Two primary methods are suggested to reduce the filter pressure drop: increasing the filter area or opting for a lower MERV rating.

These HVAC systems are vital in maintaining the optimal operating conditions for the MARVEL Micro-Reactor, providing continuous heat removal and ensuring the safety and efficiency of reactor operations. Future design iterations are expected to further refine the HVAC configurations based on detailed thermal-hydraulic analyses and other pertinent evaluations.

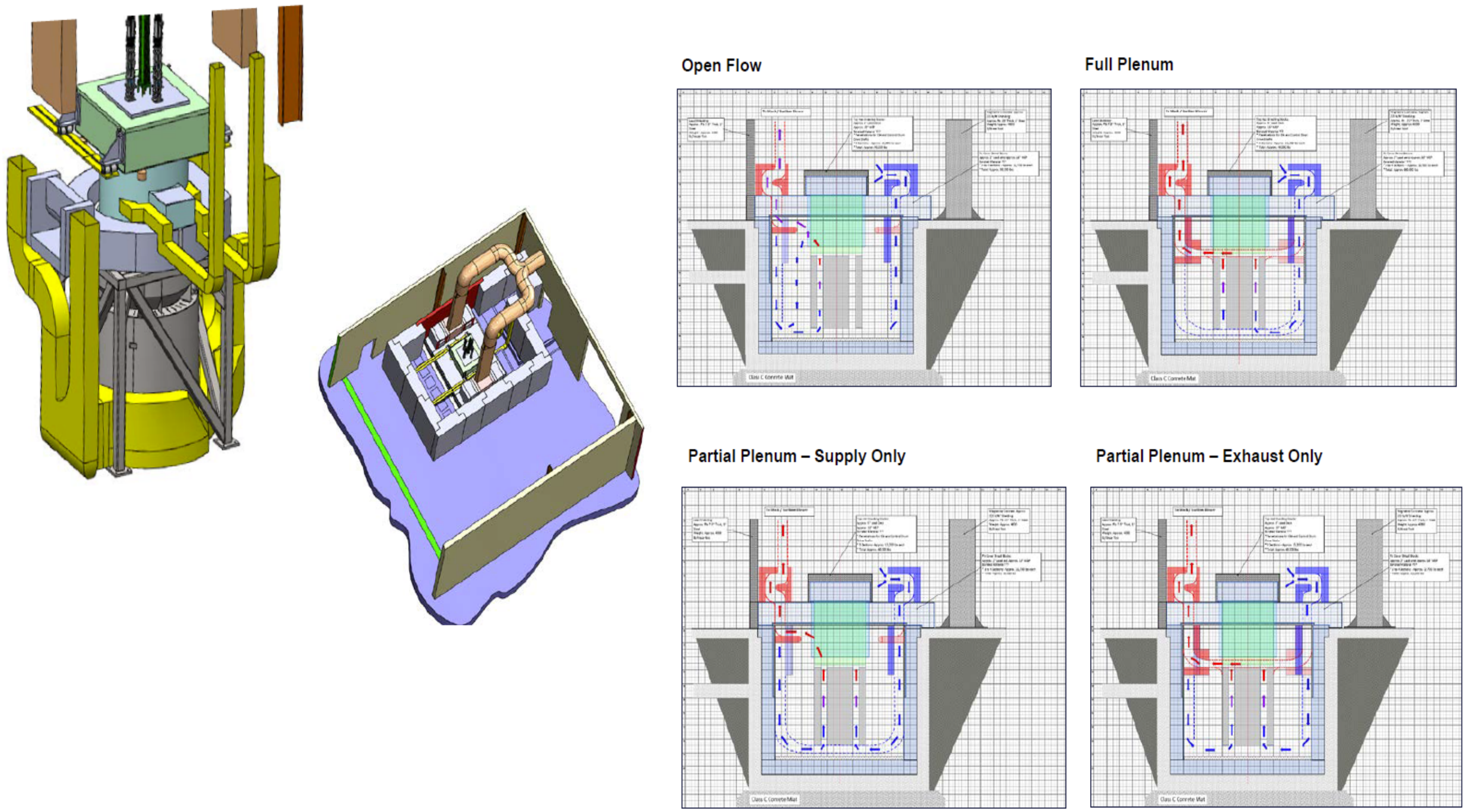

Figure 70. T-REXC ventilation concept.

5. MECHANICAL AND STRUCTURAL INTEGRITY

The PCS design is governed by ASME Section III Division 5 Subsection HB Subpart B as a Class A metallic pressure boundary component seeing service loading temperatures greater than 800°F (425°C). Design and Service Level A, B, and C checks can be grouped into four categories – (1) primary load design or load-controlled stress limits, (2) ratcheting strain limits, (3) creep-fatigue damage accumulation limit, and (4) time-independent and time-dependent buckling limits and special stress limits. Load-controlled stress limits are evaluated using elastic-perfectly plastic (EPP) and simplified inelastic analyses approach that is provided in ASME Code Case N-924. Ratcheting design follows the EPP methodology in ASME Code Case N-861. Creep-fatigue design also relies on EPP methodology as detailed in ASME Code Case N-862. Buckling design is not prescribed by ASME Section III Division 5; instead, a method of isochronous curves is used. Special stress limits such as bearing stresses and triaxial stresses are considered using elastic structural models.

To meet the MARVEL project objective of an accelerated design schedule while achieving technical excellence and fulfilling rigorous quality standards, the PCS pressure boundary is divided into two sections for Design and Service Level A, B, and C code checks. The dividing boundary is determined from the transient temperature inputs that are required when considering time- and temperature-dependent failure modes. RELAP5-3D is a thermal-hydraulics software that simulates all the design-basis transients for the MARVEL reactor in [ECAR-6332]. It takes hot-channel factors from MCNP neutronics data and generates time-dependent temperature data for the reactor coolants and structural components. The mapped temperature distributions from RELAP5-3D on the relatively simple geometry of the Lower Downcomer, Bottom Head, and Reactor Core Barrel to solve the design checks using a Finite Element Analysis (FEA) software called MOOSE. RELAP5-3D is, however, limited to one-dimensional models – this is a concern for the complex geometry and three-dimensional thermal-hydraulic behaviors in the PCS Distribution Plenum and IHX. Star-CCM Computational Fluid Dynamics (CFD) software is employed to provide characteristic temperature distributions in these regions. The CFD tool relies on boundary conditions from RELAP5-3D to generate time-dependent fluid fields with fluid and structural temperature data. Due to model size, resource requirements, and data transmission challenges, CFD is not practical for complete transient runs on a full model of the pressure boundary for the current project objectives; however, this model sectioning approach limits the resource load to an actionable level. In addition, the sectioned model takes advantage of symmetry to further reduce the computational burden. Modelling and results of the CFD analysis are documented in ECAR-6594, "CFD Analysis of the PCS."[30] Temperature data on the pressure boundary is sourced by [ECAR-6580, "ASME Section III, Division 5 Analysis of PCS"[31]] to perform the design checks on the remaining, more complex regions of the vessel using ABAQUS FEA software.

Service Level D code checks are performed separately on a full model of the PCS using an elastic analysis method. Secondary stresses due to temperature gradients are not considered for the limiting faults, but the maximum temperatures from [ECAR-6332] are used to determine allowable stress limits.

6. CHEMICAL AND MATERIALS COMPATIBILITY

The secondary fluid is GaInSn eutectic. Gallium based alloys are very good for heat transfer, minimum biological hazard and many are liquid at room temperature. However, one very large disadvantage of gallium-based alloys is that they corrode other metals very quickly. For this reason, a sacrificial liner is being utilized to provide a buffer between the fluid and the primary vessel. The secondary coolant is expected to come in contact with Stainless Steel 316H (sacrificial liner), Haynes-230 (engine dome) and Haynes-282 (engine tubes). Other stainless steels are also generically expected to be encountered.

The corrosion of SS is mainly a function of temperature. The corrosion rates are much worse past 400°C. The corrosion rates of stainless steel below 350°C would allow components to far exceed the anticipated MARVEL life. The liner and the engine will have a limited life because of the corrosion. The engine is expected to run until it fails and is expected to fail safe for pressure relief. The liner is to be replaced on a preventative maintenance basis.

Using a temperature of 430°C for the secondary coolant and allowing the liner to corrode to 1 mm before replacement, the liner is calculated to last 1460 hours [ECAR-6126, "Gallium Corrosion"[32]]. However, the corrosion rates will be verified with a prototype test [PLN-6772, "MARVEL Engine Prototype Test Plan"[33]] to show the liner and engine lifetime in GaInSn.

The engine tubes and dome are made of Haynes superalloys which are primarily made of nickel. A literature review showed that nickel-based alloys appear to have similar corrosion rates as stainless steel. Using the stainless-steel corrosion rates the expected lifetime of the engine is bounded by the tubes which are anticipated to be around 584 hours at 430°C. The engine life can greatly be extended by the fact that the tubes are cooler than the bulk fluid and by reducing the temperature of the fluid to lower than 400°C. The same test in PLN-6772 will be used to verify the engine life using the assumed corrosion rates for Haynes.

Besides the engine and the liner, GaInSn will come in contact with thermocouples which will have stainless steel sheaths and may require replacement once they fail. When the coolant has cooled down the GaInSn does not corrode very fast. The one major exception is aluminum. Any aluminum that comes in contact with gallium can be corroded in a matter of seconds to minutes. No tools should be made of aluminum. When the coolant is at room temperature it can come in contact with a variety of plastics and could be removed using a plastic pump and a plastic storage vessel. GaInSn should be maintained under an inert cover gas to prevent oxidation. Helium is used as the cover gas during normal operation. During maintenance or other activities, argon can be used as the cover gas. The reason for not using argon during normal operation is to avoid the argon activation.

The primary fluid is NaK. NaK is a unique liquid metal, it is liquid at temperatures above -12°C, and has a high boiling point of 785°C. This in addition to its desirable heat transfer properties make it a desirable heat transfer fluid for reactor systems. NaK is composed of two alkali metals and retains many of the properties of its constituent metals. As such careful handling is required and compatibility with materials that it will be in contact with must be considered. NaK is reactive with air and water, which can result in violent reactions, including smoking, fire, and explosions, as such the environment NaK is exposed to must be controlled. The primary coolant system houses the NaK and is evacuated of all air and water, then an inert atmosphere is achieved utilizing dry argon. During NaK transfer an inert atmosphere of argon is maintained throughout the process, and during all operational states while NaK is within the primary coolant system. This reduces the risk of NaK contamination to those caused by primary coolant boundary failures.

The primary materials where compatibility is essential are 316H stainless steel, and 304 stainless steel. The primary coolant boundary is composed of 316H, it is a ASME BPVC Section III Division 5 approved material. The fuel is previously qualified U-ZrH developed by General Atomics and procured through TRIGA international, the qualified design utilizes 304SS as cladding. Stainless steels are considered to have near indefinite life in sodium environments sub 1000°F, with oxygen contents below 20ppm. Oxygen requirements for NaK within the PCS are prescribed to be at or bellow 15ppm. This allows for corrosion data presented in the "Sodium NaK Handbook, Volume V,"34 to be utilized to estimate corrosion rate. This rate is predicted to be less than 1 thousandth of an inch per year.

Care must be taken with welds to minimize risk of welding related corrosion issues, particularly with respect to the high carbon 316H stainless steel. Carbide precipitation resulting in the localized depletion of chromium is common in austenitic stainless steels and is prevalent in high carbon variants. Carbide precipitation typically forms within the heat affected zone (HAZ) generated by the heat inputed into the bulk material by the welding process. The carbide precipitation around the weld within the HAZ, and the resulting region of depleted chromium results in increased susceptibility to corrosion. This is commonly referred to as sensitization. To minimize risks of sensitization solution annealing has been prescribed for strategic weldments where sensitization poses the greatest risk, along with welding procedures which minimize the size of the HAZ.

Through chemistry control of the environment and primary coolant, along with procedural adjustments to welding of components exposed to NaK. Risk of corrosion, and chemical reactions occurring within the primary coolant system are minimized, allowing for MARVEL to accomplish its anticipated two year operational lifetime without NaK related compatibility issues.

7. REACTOR SAFETY

7.1 Hazard Evaluation Summary

The detailed hazard evaluation may be found in ECAR-6440. The hazard evaluation process for the MARVEL project for compliance with the requirements in 10 CFR 830, "Nuclear Safety Management,"35 follows a process similar to the Licensing Modernization Project (LMP) as outlined in Nuclear Energy Institute (NEI)-18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,"36 and supporting documents. The LMP process was adapted to fit Department of Energy (DOE) reactor regulatory requirements as applicable and appropriate using a graded approach based on the MARVEL microreactor design. This approach provides reasonable assurance of meeting the requirements of 10 CFR 830 for protection of the public, workers, and environment.

ECAR-6440 evaluated the impacts of MARVEL operations, hazards, and postulated accidents. The hazard evaluation of MARVEL events and associated operations was performed for selection and evaluation of safety classification of systems, structures, and components (SSCs) and SSC safety functions, and for selection of design basis accidents (DBAs) applicable to the MARVEL microreactor design. The level of detail and analysis in ECAR-6440 was based on the 90% reactor design, and, where detail was unavailable, appropriate simplistic or bounding assumptions were made. As such, safety SSCs were identified for consideration in the MARVEL design effort. With these SSCs in place, the evaluation concludes that MARVEL can be built and operated safely in the TREAT facility. The final hazard and accident analysis and selection of safety SSCs will be documented in the MARVEL PDSA.

The following briefly summarizes the major tasks that were implemented in ECAR-6440:

- A systematic approach was used to identify initiating events (IEs) that challenge at-power MARVEL plant operation and require successful mitigation to prevent radionuclide release.
- Fundamental safety function (FSFs) necessary to keep the IEs identified from progressing to end states that could result in core damage and release of radioactive or hazardous material, were identified.
- Safety basis event (SBE) sequences were qualitatively modeled to obtain an understanding of accident progression; response of structures, systems, and components (SSCs) performing the FSFs; and sequence end states.
- The full set of SBEs were examined to verify that the SSCs performing the FSFs are sufficient to ensure that the evaluation guidelines (EGs) are met. For each of these safety functions, a decision was made on which SSCs should be classified as safety SSCs.
- Each SBE identified was mapped to a DBA that includes the FSF challenges represented in the SBE sequence but assumes that the FSFs are performed exclusively by safety-related (SR)-SSCs, and all nonsafety-related (NSR)-SSCs that perform these same FSFs are assumed to be unavailable.

7.2 SSC Classification Summary

Preliminary safety SSCs identified SDS-119 and in the ECAR-6440 SBE sequence analyses were further evaluated for SSC classifications and identification of required SSC safety functions as a result of the SBE analyses. ECAR-6440 demonstrates that passive reactivity control, decay heat removal, and confinement FSF SSCs are sufficient to eliminate the Internal event sequences resulting in a radiological release without reliance on active reactivity control and decay heat removal FSF SSCs. As such, the passive reactivity control, decay heat removal, and confinement FSF SSCs were selected as SR-SSCs.

Since the passive reactivity control, decay heat removal, and confinement FSF SSCs alone are sufficient to eliminate the event sequences resulting in a radiological release, the active reactivity control and active decay heat removal FSF SSCs were selected as non-safety-related (NSR) SSCs. The DBA analysis that will be performed and documented in the MARVEL SAR-420 Addendum Chapter 15, Accident Analysis, and summarized in the next section, demonstrates these conclusions, as the DBAs credit only SR-SSCs, and will identify the specific performance requirements for those passive FSF SSCs for inclusion in the MARVEL Technical Specifications document.

As discussed in ECAR-6440, MARVEL SSCs will be further classified as SR, NSR, or NSR with augmented requirements (NSR-AR), consistent with the SSC classifications in SAR-420 Section 3.2, based on the criteria in Table 7 below. Table B-1 provides the final results of safety SSC classifications. Based on the analyses in ECAR-5127 and ECAR-6332, there are no SBEs that could result in radiological or non-radiological consequences that could exceed the evaluation guidelines in ECAR-6440. Therefore, no SSCs are required to reduce the risk of the public, and no SSCs meet the SR-2 classification criterion for internal SBEs. However, the SSCs identified in Table B-1 may be designated as SR based on criteria 1, 3, 4 in Table 7.

For non-reactor nuclear facilities, DOE uses the SSC classifications of safety class (SC) and safety significant (SS) as defined in 10 CFR 830, “Nuclear Safety Management.” Table 7 provides a crosswalk between the MARVEL SR and NSR-AR SSC and DOE safety-class (SC) and safety-significant (SS) SSC classifications. If potential accidents could result in offsite consequences greater than evaluation guidelines (SR-2 criterion met), all SR SSC’s are equivalent to SC SSC’s and NSR-AR-1 SSCs are equivalent to SS SSC’s. If potential accidents do not result in offsite consequences greater than evaluation guidelines (SR-2 criterion NOT met), all SR SSCs are SS, and NSR-AR-1 SSCs are non-safety.

If potential accidents could result in onsite consequences greater than evaluation guidelines (NSR-AR2 criterion met), all NSR-AR2 SSC’s are equivalent to SS SSC’s. If potential accidents do not result in onsite consequences greater than evaluation guidelines (NSR-AR2 criterion NOT met), all NSR-AR-2 SSCs are non-safety. All NSR-AR-3 SSCs are non-safety. T-REXC SSC preliminary classifications are found in Table C-1 based on the criteria in Table 7.

Table 7. Safety SSC classification criteria.

SSC Classification	Criteria			DOE Safety SSC Classification
	Criterion	Type	Description	
Safety Related (SR)	SR-1	Deterministic	Is the SSC required to shut down the reactor and maintain it in a safe shutdown condition?	SC if SR-2 is triggered, otherwise SS
	SR-2	Risk-Informed	Is the SSC required to ensure capability to prevent or mitigate the consequences of accidents that could result in potential off-site consequences greater than the evaluation guidelines?	Exceed off-site EGs, then SC
	SR-3	Deterministic	Does the SSC contain an item required to establish an SR/NSR interface such that an SR system is isolated from a NSR system?	SC if SR-2 is triggered, otherwise SS
	SR-4		Could failure of the SSC prevent reactor shutdown or inhibit a SR SSC function?	SC if SR-2 is triggered, otherwise SS
Non-Safety Related with Augmented Requirements (NSR-AR)	NSR-AR-1	Defense-in-Depth	Is the NSR-SSC assumed in the accident analyses to provide a layer of protection to (1) shut down the reactor and maintain it in a safe shutdown condition, (2) monitor the status of the reactor, or (3) monitor and filter reactor effluent?	SS if SR-2 is triggered, otherwise Non-Safety
	NSR-AR-2	Risk-Informed	Does the NSR-SSC prevent or mitigate the consequences relative to the safety or protection of the facility or collocated worker?	Exceed collocated worker EGs, then SS
	NSR-AR-3	Deterministic	Is the NSR-SSC otherwise designated by management to support operational commitments or key assumptions in the safety analysis report?	Non-Safety
Non-Safety Related (NSR)	All MARVEL SSCs not classified as SR or NSR-AR per the above criteria, shall be classified as NSR with no augmented requirements required.			Non-Safety

7.3 DBA Analysis Summary

MARVEL DBAs are postulated event sequences that are used to set design criteria and performance criteria for the design of SR-SSCs. The DBAs were derived from the SBEs identified in the event sequence analysis in ECAR-6440 for internal and external events. Each postulated DBA is assigned to one or more of the following overall categories:

- Transient Overpower (TOP)
- Loss of Heat Sink (LOHS)
- Loss of Flow (LOF)
- Loss of Offsite Power (LOOP)
- Seismic Event (g ≤ SSE)
- Loss of Coolant Accident (LOCA)
- NaK Spill and Fire
- Radioactive or Hazardous Material Release, or Direct Radiation Exposure, from a System, Subsystem or Component.

One beyond design basis accident (BDBA) was identified for further analyses:

- Seismic Event (g >SSE).

Only SR-SSCs listed in are assumed to be available in each DBA scenario.

ECAR-6332 reports the thermal hydraulic analyses results for the MARVEL microreactor, final design, including normal operation, operational transients and a set of very-low probability transients caused by accident conditions. The ultimate scope of this document is to demonstrate the MARVEL microreactor thermal hydraulic performances and its inherent safety. First, the list of the operational and accidental transients with the corresponding safety criteria are recalled. Then, details of the final design, the key input parameters, the assumptions, and the methodology used for performing the deterministic safety analyses (DSA) are presented. In particular, the DSA has been performed using a best-estimate code (RELAP5-3D) plus uncertainties. Finally, the analyses results are provided, calculating the safety margins for each transient and demonstrating the satisfaction of the corresponding safety criteria.

The calculations demonstrate MARVEL thermal hydraulic performances for normal operation and anticipated operational occurrences, and the inherent safety of the reactor even during accidental conditions caused by beyond extreme unlikely events.

The DBA safety analyses in ECAR-6332 show that with only the applicable SR-SSCs available (a conservative assumption on systems availability), the MARVEL design provides sufficient prevention and mitigation such that there are no radiological or non-radiological releases from the TOP, LOHS, LOF LOP, LOCA, or seismic DBAs, and there are no radiological or non-radiological consequences.

8. MODELS, VERIFICATION AND VALIDATION

In any reactor design and simulation exercise, the credibility of the model results hinges critically on the underlying model's accuracy, which is confirmed through a process known as model verification and validation (V&V). Verification is the process of determining that the model implementation accurately represents the developer's conceptual description and specifications. It ensures the computational accuracy of the model and examines if the model is correctly implemented in a computer code without errors. Validation, on the other hand, is the process of substantiating that the model within its domain of applicability possesses a satisfactory range of accuracy consistent with the intended application. It tests how well the model can reproduce the reality it represents.

This section provides the overview of a detailed verification and validation of the main simulation tools used in the MARVEL project, covering MCNP, RELAP5-3D, ABAQUS, and MOOSE.

8.1 MCNP Verification and Validation

The Monte Carlo N-Particle (MCNP) code was subject to four key areas of verification and validation. First, the Sawtooth-specific installation was verified using standardized test suites supplied with the code. Second and third, validation was performed using standardized test suites for criticality and shielding applications, respectively, included in the MCNP distribution. Lastly, individual analysis of benchmarks from the International Handbook of Evaluated Reactor Physics Benchmark Experiments and the SNAP 10A reactor core benchmarks was used to validate MARVEL-specific features: UZrH fuel form, beryllium/beryllium oxide reflected systems, and liquid sodium coolant. The verification and validation of MCNP is documented in ECAR-7300, "Verification and Validation of MCNP 6.2 for MARVEL Neutronic Analysis."[37]

Next phase of validation is to use the as-manufactured data for core components (fuel, beryllium oxide neutron reflectors, reactor vessel, control drums, etc.) from their respective manufacturers. Critical parameters of the fuel include uranium content, U-235 enrichment, initial hydrogen loading in UZrH fuel and cladding straightness. Other critical parameters for the neutron reflectors and neutron poison plates are chemical composition and density. The final phase of MCNP validation will be conducted during startup testing, as outlined in Section 11.1 and elaborated in the PLN-6816, "MARVEL Startup Plan."[38]

8.2 RELAP5- 3D Verification and Validation

The use of the Reactor Excursion and Leak Analysis Program (RELAP5-3D) for liquid metal reactor simulations has been significantly explored in several Idaho National Laboratory (INL) reports. These in-depth analyses confirmed the sufficiency of the existing RELAP5-3D models, indicating that only minor modifications were required for an accurate simulation of liquid metal reactors and related experimental facilities. The competency of the RELAP5-3D tool in this domain was further evidenced by its successful application in a myriad of liquid metal reactor technology projects and in international benchmarking activities.

The next phase in the validation process involves leveraging experimental data from the Primary Coolant Apparatus Test (PCAT) apparatus. This test loop, represented in Figure 71, is a full-scale, electrically heated, thermal hydraulic setup, specifically designed to simulate the conditions within the MARVEL reactor. The PCAT facility plays an instrumental role in the validation of the thermal hydraulic model within RELAP5-3D by providing a highly controlled and measured environment for collecting experimental data.

The PCAT test plan, detailed in document PLN-6573, "MARVEL PCAT Test Plan,"[39] aims to: (i) demonstrate natural circulation within the constraints representative of the MARVEL reactor and (ii) gather crucial data for model validation of RELAP5-3D. Natural circulation is a significant parameter in reactor safety analysis as it indicates the coolant's ability to circulate without mechanical assistance, thereby preventing overheating of the reactor core during shutdowns.

The final phase in the validation process, as detailed in the startup plan, involves heating the MARVEL primary coolant using CIA heaters and no nuclear heat. This exercise ensures the model's capability to accurately replicate the thermal hydraulic behavior in the reactor under conditions of heat-up without nuclear fission heat. Any necessary adjustments to the model assumptions based on this testing are then made.

Once this validation step is satisfactorily completed, the model is then used to predict the initial startup using nuclear fission heat. The validated model, thoroughly vetted through this multi-step process, can then confidently be used in simulations to predict the behavior of the MARVEL reactor under different operational scenarios, including the initial startup phase, thereby ensuring its readiness for effective use in the safe and efficient operation of the reactor.

Figure 71. PCAT Test Loop, as assembled at Creative Engineers Inc. facility in Pennsylvania.

8.3 ABAQUS Verification and Validation

Finite element analysis (FEA) is a numerical technique extensively used for solving complex structural and thermal problems in engineering. In this project, the FEA software ABAQUS 2021.HF6 was used for conducting both heat transfer analysis and structural analysis.

To validate the accuracy of the simulations performed with ABAQUS, several test problems were selected. These test problems were benchmark cases with known analytical solutions, providing a reliable means for comparison and validation. For each of these test problems, ABAQUS simulations were conducted, and the resulting outputs were meticulously cross-verified against the corresponding published analytical solutions or hand calculations. This comparison process confirms that the program can generate accurate results within the bounds of known physics and mathematics., applicable to MARVEL reactor operation.

To streamline and automate the validation process, scripts were developed. These scripts automated the sequence of tasks, including the execution of the test problems, collection of data from the outputs, and calculation of the relative error between the ABAQUS results and the benchmark analytical solutions. Automation of these tasks reduces the potential for human error in the validation process and ensures consistency in the execution of each test. All the scripts were executed on High-Performance Computing (HPC) system, specifically on the "Sawtooth" configuration. HPC systems provide substantial computational power, allowing for efficient processing of complex and large-scale problems, like those often encountered in FEA. The results of the validation process using these scripts on "Sawtooth" were satisfactory, with the outputs from ABAQUS closely aligning with the known solutions to the test problems. Finally, the verification and validation of ABAQUS is captured in ECAR-5540, "ABAQUS V&V Plan."[40]

8.4 MOOSE Verification and Validation

Blackbear, alongside the entirety of the MOOSE (Multiphysics Object-Oriented Simulation Environment) framework, operates under a comprehensive software quality assurance (SQA) plan. This meticulous plan is publicly documented and can be accessed at https://mooseframework.inl.gov/sqa/. An intrinsic component of this SQA plan is the inclusion of a series of automated tests. These tests are intricately designed to verify and validate not only the physics modules present within the MOOSE framework but also specific applications based on MOOSE, such as Blackbear. These V&V tests are multifaceted, encompassing several layers of scrutiny. Firstly, they involve benchmarking against established analytical solutions, ensuring that the software's outputs are consistent with known results. Next, they employ the method of manufactured solutions, a technique which involves creating artificial yet plausible problems to further challenge and test the software. Additionally, convergence rate analysis is used to ensure that the software provides progressively accurate results as the computation's granularity is increased.

Another noteworthy aspect of the V&V plan for Blackbear is its comprehensive test suite, which draws comparisons between its outputs and those from other renowned structural analysis software packages, ABAQUS being a prime example. This comparative analysis guarantees that Blackbear's results are consistent with industry standards and best practices. To further enhance the efficacy of these tests, an automatic test harness has been deployed. This harness not only ensures the seamless functioning of the installed application but also acts as a sentry, maintaining rigorous regression testing. This is crucial as both the MOOSE framework and Blackbear application continuously evolve through software development iterations. It is worth noting that Argonne National Laboratory (ANL) administers these tests every time Blackbear is installed on a new computational platform, ensuring consistent performance across diverse hardware configurations.

8.5 STAR-CCM+ Verification and Validation

Simcenter STAR-CCM+ is a computational fluid dynamics (CFD) software package originally developed by CD-adapco but acquired by Siemens Digital Industries Software in 2016. It is utilized to simulate components and systems operating under real-world conditions. For MARVEL, it was specifically used to perform thermal-fluidic analyses of the PCS and various components within it.

As part of the SQA plan for STAR-CCM+ a subset of test problems was selected from a suite of problems provided by the software developer for V&V purposes via comparison with previous runs and measured data. The main objective of this comparison is to V&V STAR-CCM+ by demonstrating the results of these runs match the provided benchmarks within an acceptable tolerance. These problems have been documented in ECAR-3020, "STARCCM Multi-Physics Validation,"[41] and have recently been re-run on the latest configuration of INL's HPC system Lemhi as described in ECAR-6594, "Computational Fluid Dynamics Analysis of the Primary Coolant System of MARVEL Microreactor," "Computational Fluid Dynamics Analysis of the Primary Coolant System of MARVEL Microreactor."[42] Final validation of STAR-CCM+ for use in the MARVEL project will follow the last two phases of the validation process for RELAP5-3D as outlined in Section 8.2. This includes leveraging experimental data from the PCAT apparatus tests as well as data acquired during the initial startup of MARVEL where it's planned to utilize the CIA electrical heaters prior to use of nuclear heating.

9. QUALITY ASSURANCE

This section describes the Quality Assurance controls that shall be applied during the design (in addition to the verification and validation activities described in Section 8) and manufacture of the Microreactor Application Research Validation and Evaluation (MARVEL) project.

The overarching Quality Assurance program employed at INL is defined in PDD-13000, "Quality Assurance Program Description."43 This document is written in accordance with the requirements of 10 CFR 830, Subpart A 'Quality Assurance Requirements' and the directives of DOE 414.1D, "Quality

Assurance."[44] PDD-13000 has been approved by DOE Idaho Operations Office (DOE-ID) and defines how quality-affecting work shall be performed at INL and the application of QA requirements. The document also defines the application of a graded approach based on significance and risk.

In conjunction with 10 CFR 830 a full safety analysis was performed, and MARVEL was classified as a Category B reactor. Furthermore, MARVEL is to be installed at INL's TREAT complex - a Hazard Class 2 nuclear facility. This analysis and classification determined that the suitable code of construction for the fabrication of MARVEL is ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications."[45] ASME NQA-1 governs quality assurance protocols for any structure, system, component, activity or organization that is essential to the safe, reliable, and efficient performance of a nuclear facility. As such, the standard serves as the appropriate guidance on how MARVEL is designed, procured, manufactured, and tested. All project activities shall be performed in conformance with INL's Quality Assurance Program which implements ASME NQA-1 requirements. Specific notable applications of the NQA-1 Program are described in the following sections.

9.1 Design

In accordance with INL's Quality Assurance Program (PDD-13000) and Conduct of Engineering (PDD-10000) engineering and design inputs for MARVEL are controlled and verified to be in accordance with regulatory and performance requirements. Design inputs are performed by qualified personnel and verified via an independent peer review and/or software validation.

Before formally proceeding with design, INL's engineering team performed a thorough analysis of the MARVEL concept and safety performance. These documents included a Safety Design Strategy (SDS-119), MARVEL Code of Record (COR-0011), and an Environmental Assessment (EA-2146). These documents, among others, served as guiding principles for design inputs for the MARVEL engineering considerations.

The engineering performer is responsible for identifying and specifying the appropriate quality standards, requirements, inspections and testing4 related to MARVEL. Given the unique features of MARVEL it was deemed appropriate that ASME Boiler and Pressure Vessel Code, Section III, Division 5 would be the suitable code application for both design and manufacture.

Materials of construction for the Guard Vessel and Primary Coolant System shall be procured under the provisions of Division 5, whereas materials for the Support Structure(s) shall be purchased under NQA-1 requirements. In both cases materials would be purchased from a qualified supplier, validated via third party testing, and traceable to the originating mill. Where appropriate Commercial Grade Dedication shall be performed on Safety Significant Items when not purchased from an approved NQA-1 supplier.

As discussed in the MARVEL Code of Record (re: COR-0011, Rev 0) the MARVEL microreactor will not receive an N stamp, however the appropriate QA rigor has been applied to assure an equivalency of requirements.

9.2 Supplier Oversight

The MARVEL project has assigned a full-time INL Quality Engineer who shall be resident at the manufacturer's facility during the fabrication and testing of the Guard Vessel, the Primary Coolant System, and the Support Structure(s). The Quality Engineer shall work independent of cost, schedule, or the direction of work and shall serve in a supplier oversight capacity. The QE role and responsibilities have been written into the contract specification7 and shall verify that MARVEL is manufactured in accordance with the design drawings, specifications, and the code references therein. The QE shall also be signatory to any contractual change requests (both fiscal and technical) that may be flowed down to the supplier.

The oversight role shall include (but is not limited to); verification that materials of construction are in accordance with design and PO requirements, witness and validate dimensional inspection(s), validate calibration of M&TE, witness weld activities, verify that welding is performed within the parameters defined on the Weld Procedure Specification, verify that the supplier is maintaining configuration management and control of documents, and witnessing of test activities.

This oversight shall be supported via Surveillance Reports, Source Inspection Reports, electronic spreadsheet tracking, and Weekly Status Reports back to the MARVEL project stakeholders. Where appropriate, these reports shall be supplemented with photographs and/or exhibits (test reports, weld maps and dimensional inspection reports etc.) In accordance with INL procedure LWP-1202 these documents shall be managed and retained as a project record.

INL Quality Assurance oversight shall also be applied for the manufacture and supply of the Beryllium Reflectors, Zirconium-4 Debris Shields, and Stirling Engines. The contracts for these suppliers have a 'right of access' condition written into the specifications therefore have provision for INL surveillance and source inspection. The provider of these components will also submit the appropriate QA documentation – M&TE calibration records, manufacturing plan/traveler, qualifications of inspection & test personnel, inspection and test reports, certificate of conformance, and shipping & handling plan. Documentation from these suppliers shall be transmitted via INL's Vendor Data System, reviewed, dispositioned with a code status, and returned to the supplier.

9.3 Inspection and Testing

Inspection and testing of the Guard Vessel shall be performed in accordance with the appropriate criteria defined in ASME Boiler and Pressure Vessel Code, Section III, Division 5. Weld inspection and acceptance shall follow the criterion listed in Article NCD-5300 of ASME Boiler and Pressure Vessel Code III.1.NCD-2021. Radiography, Dye Penetrant, and Ultrasonic tests shall be performed using qualified inspection personnel and pre-approved procedures. Guard Vessel structural and leak tightness integrity shall be verified by pneumatic pressure testing (per HCB-6000) and Helium Leak Testing - the latter shall be performed in compliance with Article 10 of ASME Boiler and Pressure Vessel Code, Section V.

The Primary Coolant System will also be inspected and tested in accordance with ASME Boiler and Pressure Vessel Code, Section III, Division 5, however given that the Primary Coolant System is the boundary for NaK, the rigor of HBB-5000 is applied (Requirements of Class A Vessels Containing Hazardous Substances). The integrity of welds shall also be inspected via radiography (RT) inspection. In locations where radiography is not possible, welds shall be inspected using ultrasonic (UT) examination. Weld inspections shall be performed by a qualified inspector using pre-approved test procedures. Test inspections shall be documented and submitted via the Vendor Data System. These documents shall be retained as a project record.

10. CONSTRUCTABILITY AND ASSEMBLY LOGISTICS

The Idaho National Laboratory (INL) is embarking on the MARVEL project to construct the first microreactor in the United States in over 50 years, which will be located within the existing Transient Reactor Test Facility (TREAT). This ambitious initiative presents unique logistical challenges, as it must be executed without disrupting TREAT's ongoing operations. Moreover, the TREAT building doesn't have sufficient storage for the MARVEL Systems, Structures, and Components (SSCs) under the necessary temperature-controlled and protected conditions. Consequently, the MARVEL project demands a carefully devised receipt, storage, and assembly plan to ensure the safe, efficient, and uninterrupted construction and assembly of the microreactor within the TREAT facility.

10.1 Long Lead Procurement

MARVEL is an operating-funded project and as such, the long-lead procurement (LLP) requirements are governed solely by the guidelines provided by 10 CFR Part 830 and DOE-STD-1189-2016.

In compliance with DOE-STD-1189-2016 and Section 830.206 of 10 CFR Part 830, LLP of structures, systems, and components (SSCs) is permitted with prior authorization from DOE, even before approval of a preliminary documented safety analysis (PDSA), provided the activities are not detrimental to public health and safety and align with the best interests of DOE. Procurements during the design phases, as approved by DOE, should ensure that (i) the approval basis is upheld as design and procurement activities progress, and (ii) any changes are communicated and reconciled in a manner consistent with managing design changes post-PDSA approval.

There are several benefits to early material procurement for the MARVEL project, including:

- Cost savings due to reduced impacts from the current volatile market conditions, as well as a shorter total project duration, which reduces the project's "hotel load."
- Additional time to address potential supply chain issues, reducing the risk of not receiving critical components on time.
- Improved integration with other programs waiting for MARVEL to commence operations and enable demonstrations.
- Early involvement of long-lead vendors in final design constructability reviews, minimizing the risk of rework or redesign.

However, there are also risks associated with proceeding with LLPs. Significant changes to the final design are not anticipated, but minor alterations in material selections are possible, and more substantial changes may affect final fabrication dimensions. Small adjustments in dimensioning and tolerances are possible, and not all ASME analysis is complete. Nonetheless, preliminary finite element analysis (FEA) models have confirmed that the geometry dimensions should meet the ASME boiler and pressure vessel code (BPVC) Section III Division 5 design requirements.

To manage the risk associated with LLPs, MARVEL has developed a tailored procurement strategy that features phased approvals, optimizes the schedule, and balances the risks and opportunities of early procurement. LLPs are divided into two categories – those with the longest delivery times and those with more moderate delivery times. The first group of LLPs are shown in Table 8.

Table 8. LLP summary for Group I LLPs.

Material	LLP SSC	Rationale for early procurement
Stainless Steel (SS) 316H	Request for Proposal (RFP) No. 3727044, MARVEL Primary Coolant System	Procurement of SS 316H and fabrication duration significantly extends project duration, if procured after final design and PDSA approval. Phased approach to contracting (design participation, material procurement, fabrication) mitigate the risk of procuring early and improves safety by involving vendor in the design
	RFP No. 3716594, MARVEL Guard Vessel	
	RFP No. 3729344, MARVEL Support Frame and Outer Shell	
HALEU Fuel	MPO No. 00268251, MARVEL Fuel Slugs	High-assay low-enrichment uranium (HALEU) production is long, so is the UZrH fuel fabrication, shipping, and receipt.

In the MARVEL project, a phased procurement approach was implemented for Group I Long-Lead Procurements (LLPs). The first phase involved issuing requests for proposals, evaluating respondents, and awarding contracts. Contracts for stainless steel SSCs included hold points requiring written authorization to transition from material procurement to fabrication, ensuring fabrication aligns with the final design. The vendor took part in the final design for constructability reviews, completing the first contract phase. With DOE authorization, the next phase commenced, and stainless-steel materials were procured after final drawings were released to the vendor.

Additionally, DOE's authorization led to the awarding of a memorandum purchase order (MPO) for the provision of metallic HALEU from Y12, based on an approved material specification used for MARVEL neutronic modeling, not the final design. The MPO was not phased like the stainless steel contracts, and LLP for fuel fabrication will be discussed in subsequent sections.

Approval for a second set of LLPs (Table 9), referred to as Group II, was requested post-final design review. These SSCs have long lead times, though not as significant as Group I's. Fuel fabrication may take up to a year post-HALEU delivery to TRIGA International (TI) but finished fuel won't be needed until after fabrication and readiness. Thus, HALEU procurement approval was requested as part of Group I LLP, while fuel fabrication approval was requested as part of Group II LLP.

Table 9. LLP summary for Group II LLPs.

Long Lead Procurement SSC	Status on design stability	Rationale for early procurement
Fabrication of 316H SS SSCs described in Group I	Final Design Review (FDR) complete, final drawings issued	Procurement of SS 316H and fabrication duration significantly extends project duration, if procured after final design and PDSA approval. Phased approach to contracting (design participation, material procurement, fabrication) mitigate the risk of procuring early and improves safety by involving vendor in the design
RFP No. xxxxxxx MARVEL Fuel Fabrication	Stable design with significant fabrication history; Fuel cladding and fuel element specifications (SPC-2999 and SPC-3000) and drawings issued; MARVEL Fuel Performance Report issued	Fuel fabrication is available from a single contractor in France and must be integrated with other programs; fabrication may take as long as year
Reactivity Control System	Stable Design, 90% final design; Several full-scale prototypes were constructed and tested for viability	System needs to be built to conduct qualification tests. This is the primary means of controlling reactivity. Installation of the reactivity control system occurs very early in the construction sequence
Stirling Engines and Controls	Stable, “off-the-shelf” Design, 90% final design;	Delivery may take up to six months; design is stable and “off-the-shelf;” significant programming time will be required after receipt
Beryllium Metal Reflector	Stable, “off-the-shelf” Design, 90% final design; simple component with dimensions determined by fuel diameters for which design has been completed	Delivery time is significant and installation is one of the first activities in the fabrication sequence

10.2 Fabrication

Fabrication is a crucial part of the MARVEL project and encompasses a wide range of activities, from vendor selection to project closeout. This process involves a comprehensive approach to ensure that the materials and components created meet the required quality and safety standards.

1. Fabrication Vendor Selection: This step involves identifying and selecting a vendor capable of fabricating the necessary components. 1.1. Selection Criteria: The criteria for selecting a fabrication vendor might include their technical capabilities, experience, adherence to required standards, cost, and schedule. 1.2. Award: Once a suitable vendor is identified based on the selection criteria, a contract is awarded to them to proceed with the fabrication.
2. Fabrication/Design Requirements: These requirements are based on specifications provided by DOE/INL and are essential for ensuring the components meet the desired standards. 2.1. DOE/INL Requirements: These are specific requirements provided by the Department of Energy and Idaho National Laboratory. 2.2. NQA-1: This refers to the Nuclear Quality Assurance standard that the fabrication process must adhere to. 2.3. ASME: The American Society of Mechanical Engineers standards also need to be followed in the fabrication process.
3. Material Acquisition: This involves procuring the necessary materials for the fabrication. 3.1. Long Lead Procurements: These are materials that have a significant lead time and must be ordered well in advance.
4. Qualified Personnel: The fabrication process should be conducted by personnel who are qualified and experienced in the relevant field.
5. Fabrication Shop Material Control: This involves managing and controlling materials in the fabrication shop. 5.1. Inventory Storage: Proper storage of inventory is essential to prevent damage or loss. 5.2. Material Segregation: Different materials must be kept separate to prevent contamination or mix-up. 5.3. Material Control through Fabrication: As the fabrication progresses, materials must be tracked and controlled to ensure that the right materials are used.
6. Fabrication: This is the actual process of creating the components based on the design specifications.
7. Quality Control/Quality Assurance: These are measures to ensure the fabricated components meet the required standards. 7.1. Vendor: The vendor is responsible for ensuring quality control in the fabrication process. 7.2. Owner: The project owner also plays a role in quality assurance, typically through inspections or audits.
8. Acceptance: Once the fabricated components are complete and have passed all quality checks, they are formally accepted by the project owner.
9. Shipment to Owner’s Facility: The fabricated components are then shipped to the owner's facility for installation or use.
10. Closeout: This is the final step where all project activities are completed, documentation is finalized, and the project is formally closed.

10.3 Receipt of Materials

The materials and components required for the MARVEL project at the Idaho National Laboratory (INL) are supplied by diverse vendors. These items' receipt, storage, and handling are critical in the construction and assembly process. To ensure the safe and proper receipt of these materials, the following plan outlines the steps to be followed upon arrival at TREAT. Vendors have been instructed to securely package their materials and components in containers to protect against damage during transit.

Upon receipt, each shipment will be inspected for accuracy, completeness, and any signs of damage. Following inspection, the materials and components will be transported to designated storage areas based on their requirements:

- Non-Temperature Sensitive Components: The majority of the components that do not have temperature sensitivities will be stored in Conex containers located behind the TREAT facility.
- Temperature Sensitive Components: Some components have specific temperature requirements and will be stored accordingly. These components include electrical equipment and neutron detectors.
- Criticality Components: Certain components, such as beryllium, neutron sources, and gas canisters, have specific safety and security requirements. These will be stored in specially designated areas.
- Outside Storage: Items such as bottle racks, NaK containers, and QL-1 components will be stored in secured areas outside the TREAT facility.
- Fuel Pins: The fuel pins will be stored in 7M casks within a shielded and locked shipping container outside the TREAT facility.

The designated storage areas will provide the necessary protection against environmental factors, unauthorized access, and ensure compliance with safety regulations. The possibility of using TREAT QL-1 storage will be explored based on space requirements and suitability.

10.4 Assembly

This assembly plan outlines the construction process of a TREAT Assembly unit. It breaks down the procedure into different components, with detailed steps and estimated timeframes for each step. The process can be divided into various categories, as illustrated in Figure 72, which will be highlighted in the following sections.

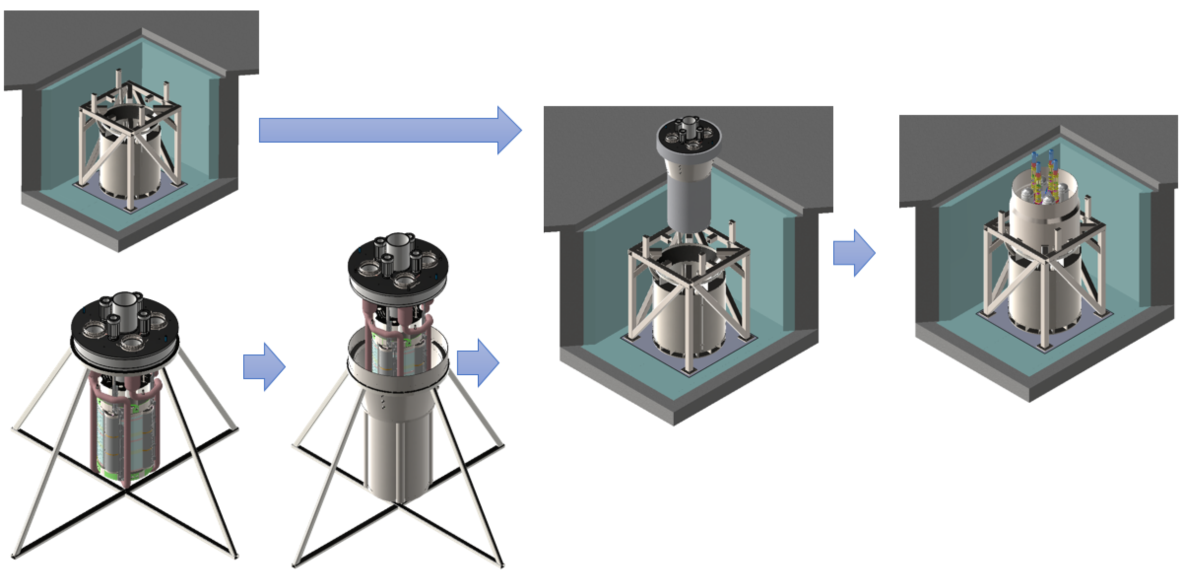

Figure 72. Illustration of MARVEL assembly steps.

10.4.1 TREAT Assembly Preparation (Figure 73)

1. Pit Base Plate Installation: The pit base plate, provided by TREXc, is lifted and lowered into its designated position, taking about 1 hour. This serves as the foundation for the reactor installation.
2. Pit Wall Shield Assembly: Shields, also provided by TREXc, are assembled around the wall of the pit over the course of 12 hours to provide radiation protection.
3. Reactor Bottom and Radial Shield Installation: The reactor bottom shield and radial shields are lifted and lowered into position, taking about 1 and 4 hours, respectively. These shields protect from radiation emanating from the reactor's bottom and sides.
4. Air Plenum and Reactor Support Frame Installation: The air plenum and reactor support frame are lifted and lowered into position over 1 and 2 hours, respectively. The support frame is anchored to the bottom base plate using screws, and neutron detector lifts are attached. Ceramic insulation plates are also attached to the support frame.
5. Installation of Stirling Engine Heat Rejection Unit and Related Components: The Stirling Engine Heat Rejection Unit is installed on the North Wall in 2 hours, followed by the installation of heat exchangers, heat rejection units, pumps, and hoses connected to applicable loops.
6. I&C Cabinet for Reactor Components Installation: The Instrumentation and Control (I&C) cabinet for reactor components, assumed to be assembled in the lab, is rolled into location and connected. This process takes approximately 1 hour.

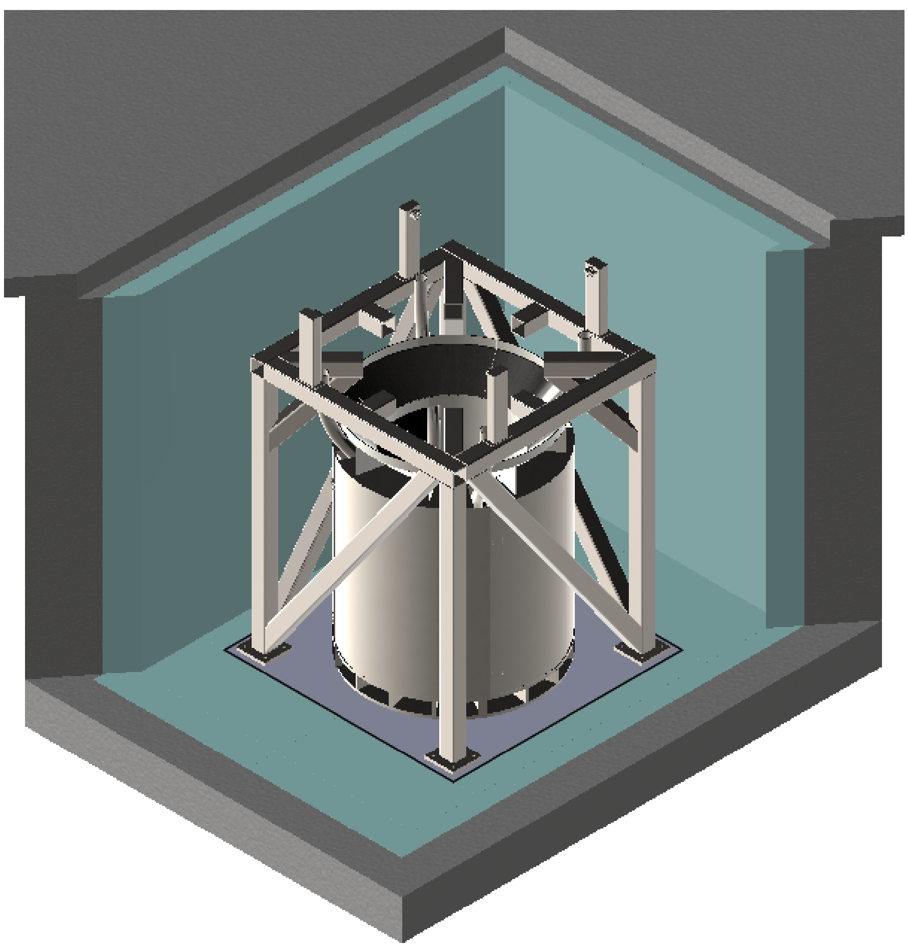

Figure 73: Illustration showing initial TREAT Pit Assembly.

10.4.2 Reactor Internal Assembly (Figure 74)

1. Primary Coolant System (PCS) Installation: The PCS is lifted and lowered onto a temporary support frame using stand-offs for SSS as lifting eye mounts in about 1 hour. The PCS distribution plenum is leveled and the opening in the upper core barrel is sealed with plastic.
2. Upper Reflector Support Plate Installation: These support plates are brought into position and attached to lifting rings in 2 hours. A floor crane performs the initial lift and the plates are moved into position and screwed together.
3. Upper Support Plate Hardware Installation: Pins and springs are placed in holes with reflector support plates, and set screws are installed. The upper control drum bushings, bushing retainers, and alignment ring segments are screwed in. The upper support plate is leveled and the alignment rings are tightened to fit the core barrel.
4. Axial Gamma Shield Blocks and Axial Neutron Shields Installation: The gamma shield blocks and neutron shields are lifted and moved into position on the support plate in 1 hour. The neutron shields are filled with B4C powder for radiation protection.
5. Lower Reflector Support Plate Installation: The process involves lifting rings, a floor crane, and stands for lifting the plates onto stands and screwing them together in 2 hours.
6. Lower Support Plate Hardware Installation: The lower control drum bushings are inserted and installed, and alignment rings are attached in 1 hour. The lower reflector support plates are leveled and the alignment rings are tightened in position.
7. Fixed BeO Reflector Installation: Over 12 hours, steel reflector plates are placed on each quadrant and BeO reflectors and their thermocouples are installed in stacks separated by debris shields.
8. Control Drum Installation: Lifting caps and floor cranes are used to raise and move the control drums into position and seat them onto bushings in 4 hours.
9. Installation of Upper and Lower Straps: The upper straps are lowered until they contact BeO and preload plates, and the pins from the reflector preload plates are removed. The upper reflector support plates are raised to their final position, and the lower straps are installed. The lower support plates are raised to their final position, and leveling is checked and adjusted as necessary in 2 hours.
10. Control Drum Temporary Locks: While the control drums are all visible and able to rotate by hand, temporary locks need installed with the control drums in their shut down orientation, to prevent motion. Each of them interfaces with their own control drum, and the process should take about 4 hours.
11. Downcomer and Distribution Plenum Insulation: Thermocouples need installed in the various thermowells around the reactor, then the downcomers and distribution plenum need insulation blankets installed. The field-wrap around the downcomers will be the most time-consuming, but overall all of these should be done within 5 hours.

Figure 74. Illustration of reactor internal assembly on temporary support frame.

10.4.3 Guard Vessel (Figure 75)

1. The guard vessel must be lifted and lowered into a second temporary support frame on the TREAT floor, then leveled with shims. After that is complete, the instrumentation wires need to be routed through their appropriate penetrations in the guard vessel. Then the reactor internal assembly can be lifted and lowered into position within the guard vessel, paying attention to clocking feature locations and interfaces and pulling cable slack out through the guard vessel penetrations. This should happen over the course of about 5 hours.
2. A circumferential weld needs to be performed between the Primary Coolant System and the Guard Vessel, ensuring both weldments are level. After that is done, it needs inspected using volumetric testing, either Ultrasonic or Radiograph, as well as visual and penetrant testing. The sealing braze on the instrumentation wires needs installed, and then the threads need seal-welded to prevent leaking. Remaining penetrations may be used as pressure taps, which can be installed at this point. The guard vessel may be leak-tested as necessary here, when the welds are exposed for possible repair, then the lower section of the guard vessel can be wrapped with insulation. Overall, this will take about 20 hours.
3. The fully welded assembly can be lowered into position on the reactor support frame in the pit next, paying attention to clocking features, moving the insulating plates as necessary to accommodate the vessel, then shimming the vessel into place to prevent sliding. At that point, the mid-section and upper section of the guard vessel will be wrapped in insulation. This should be able to happen in 3 hours.
4. Install pressure taps from PCS and GVS.

Figure 75. Illustration of reactor internal assembly welded with guard vessel.

10.4.4 Fuel Loading

For ease of reference since several of the steps need repeated multiple times, the fuel loading procedure is numbered below.

1. Preliminary Steps (1 hr)
    1.1. Double check drums are still locked in shutdown
    1.2. Install Neutron Detectors in initial positions and connect to cabinet for monitoring readings
    1.3. Remove plastic from PCS upper core barrel
2. Shell Assembly (4 hr)
    2.1. Place steel spacers on lower grid plate
    2.2. Place pins and beryllium reflectors in sequence on steel spacers
    2.3. Place upper spacers on top reflectors
    2.4. Place pins and vertical tie plate on top spacers
    2.5. Attach neutron source to lower grid plate
    2.6. Lift shell assembly using lifting device and overhead crane
    2.7. Lower into core barrel
    2.8. Perform initial calibration of neutron detectors, adjusting position for optimal counting
3. Fuel Pin Subassembly (72 hr)
    3.1. Screw one fuel pin into alignment plate
    3.2. Slide upper alignment system lower piece over fuel pin
    3.3. Insert capture pins on either side of upper alignment system
    3.4. Thread capture pin closure until hand tight to lock capture pins in place
    3.5. Repeat steps 3.1 through 3.4 for five more fuel pins
    3.6. Slide upper alignment system grid plate over capture pin closures

4. Screw grid plate locks into upper alignment system grid plate
    4.1. Connect lifting device to upper alignment system
    4.2. Lift with overhead crane and lower into position in reactor
    4.3. Check neutron detector measurements
    4.4. Disconnect lifting device using extended reach tooling and remove from reactor
    4.5. Repeat steps 3.1 through 4.4 for five other fuel pin subassemblies
5. Zero Power Physics Testing (12 hr)
    5.1. Begin phase one of zero power physics testing, for coarse adjustments to reactivity
    5.2. Remove fuel rod and replace with dummy rod if necessary
6. Locking core (2 hr)
    6.1. Lift and lower top grid plate onto core, ensuring it interfaces with pin subassemblies
    6.2. Lower core seals onto core and tighten with captured bolts
7. CIA Housing (4 hr)
    7.1. Lift CIA housing using threaded holes
    7.2. Lower onto position, taking care of interface with fuel core
    7.3. Inspect seating surface between upper core barrel and CIA housing
    7.4. Perform any necessary weld prep, such as placing a backing ring or removing debris
    7.5. Weld CIA housing to upper core barrel using an orbital welding machine
    7.6. Perform inspections on weld

10.4.5 Upper Reactor Components

1. Control Drum Lock Installation: The temporary locks are removed from the control drums and replaced with the sealing and locking assembly, one at a time to minimize any reactivity insertion. Then each of the locks are locked in the shutdown position

2. Secondary Support Structure Installation: The secondary support structures are lifted and lowered into position and screwed into place on the PCS distribution plenum in 1 hour.

3. Upper Air Plenum Installation: The upper air plenum is lifted and lowered onto the reactor support frame using swivel hoist rings and an overhead crane in 1 hour.

4. Pit Lid Installation: The neutron detectors are temporarily disconnected, and the pit lid, possibly provided by TREXc, is lifted and lowered into place in 4 hours. The neutron detector wires are routed through the pit lid and reconnected.

5. Upper Confinement: All applicable components are connected to patch panels and seals are tested as necessary in approximately 4 hours.

10.4.6 Additional Steps

The remaining assembly steps include secondary coolant purification and filling, installation of Stirling engines and CD actuators, zero power physics testing and grey rod selection, purging and filling of PCS and GVS with argon gas, filling PCS with NaK, connecting ducting from upper confinement to TREAT stack, finishing shielding assembly, installing the top hat (provided by TREXc), gamma shielding assembly (provided by TREXc), fire barrier (provided by TREXc), cable connections to TREAT, and connecting Stirling engine water lines to heat exchanger loops.

Note: The elaborated steps cover a large portion of the assembly plan, but due to the extensive nature of the plan, some steps have been summarized for brevity. Please refer to the MARVEL Assembly and Construction Plan for further details.

11. TESTING AND OPERATION

11.1 Startup Testing

A startup test program is conducted to verify that MARVEL has been constructed per design and operates as intended. The program is anticipated to begin with Construction Testing, where the equipment's placement, cleanliness, and operation are verified. Instruments are calibrated. The electrical and instrument circuits are prepared for operation, then initially energized. The cleanliness of the internal portions of the reactor vessel, which is exposed to the NaK coolant and Argon cover gas systems, is verified. Discrepancies between components and systems required and actual operation are identified and resolved.

The program continues with Preoperational Testing, where the operation of components as part of systems is tested to the practicable extent permitted by existing conditions. The neutron monitoring instrumentation is energized, adjusted, and verified to operate as expected. Thermal expansion, area and airborne radiation surveys and area temperature measurements, and other data to describe the condition of the plant at various primary temperatures and core power levels are initiated and will continue throughout the startup testing program.

An initial set of plant instrumentation readings called 'state point measurements' are noted and evaluated for proper operation and to obtain baseline measurements. Whenever conditions significantly change, the state point measurements are repeated to support the evaluation of the plant and to identify and support the assessment of trends in the plant's systems and core.

The core is loaded. The responses of the neutron instrumentation to the loading of individual sections of the core are monitored and compared to predictions. The neutron monitoring instruments continue to monitor the core after completing core loading.

The reactor vessel head is installed, and preoperational testing verifies the expected operation of the Central Insurance Control Rod and other systems whose operation has been affected by the addition of the core. If required, the excess core reactivity is measured.

Cold preoperational testing of the cover gas control system is completed, and the primary coolant system is inerted and dried in preparation for receiving the NaK coolant. The NaK coolant is loaded, and preoperational testing of the NaK inventory management system and other systems, whose operation is affected by the addition of coolant, is completed.

The core is brought critical. During the approach to critical, control element positions and coolant temperatures are made, and the readings of the source range neutron monitoring instruments are noted and compared with predictions of the predicted critical position.

Cold Low Power Physics Testing is conducted. The Reactivity Computer's operation is verified, and the Low Power Physics Operating Power Window is determined. Tests are anticipated to include Isothermal Temperature Coefficient measurements, Reactivity Worths of the Reactivity Control Devices, the Control Drums and the Central Insurance Control Rod, and the Reactor Statepoint. Reactor Statepoint measurements collect readings of the position of the control elements, coolant temperature, and indicated power level. Tests being considered include a Control Drum Shadowing test, where the effect of the change in control drum position on the adjacent neutron detectors is noted and evaluated, and a Core Stability Test. In the Core Stability Test, variations on reactivity are made over a range of frequencies to develop a Transfer Function of the core. The responses of the neutron monitoring instruments are observed to the normal changes in power level produced by low-power physics testing, and following reactor scrams. The reactor is intentionally scrammed at least once to verify the proper operation of the reactivity control devices and neutron monitoring instrumentation.

The core is returned to critical at a power level to facilitate measurement of the Isothermal Temperature Coefficient. The plant is heated to hot standby, and electrical heaters above the core maintain a hot standby temperature. During the heat-up, thermal expansion measurements are made as required to verify expected thermal growth, and the natural circulation and thermal distribution of the primary coolant system are observed.

At hot standby conditions, the core is intentionally scrammed at least once to verify the proper operation of the reactivity control devices, the Reactivity Control Drums and the Central Insurance Control Rod, and neutron monitoring instrumentation. The Low Power Physics testing is repeated at hot conditions.

The core's power level is increased to a low level to observe and evaluate the passive decay heat removal pathways. Core power level is increased in several plateaus of essentially constant core thermal power level at approximately 20%, 50%, 80%, and 100% of total core power level.

The activities in each plateau are expected to follow essentially the same sequence. Testing that does not require an essentially constant power level and is unlikely to trip the core is conducted. In contrast, the concentration in the core of a transient neutron poison, Xe135, stabilizes during the first approximately 24 hours of essentially constant power operation. The core thermal power level is measured, and indications of reactor power by the neutron and other instrument systems are adjusted accordingly. Supporting systems such as cover gas management and Stirling engine are observed, and their control systems are adjusted as required for stable operation. Measurements include thermal expansion, system operation measurements and adjustments, and the area and airborne radiation level and temperature measurements.

Tests requiring constant power level and equilibrium Xenon concentration are conducted during the 'Stable Power' portion. Core State Point measurements are made at constant power levels, control element positions, and coolant temperatures. The Isothermal Temperature Coefficient is measured. Consideration is being given to conducting Core Stability Measurements at every power plateau.

Activities that could trip the reactor or involve other significant changes in power level are conducted during the third and final portion of a plateau, called the 'Transient Portion.' Consideration is being given to conducting asymmetric loop tests consisting of different control drum positions and asymmetric load on the primary coolant system by the Stirling generators. A reactor trip terminates each plateau, and data is gathered about the resulting changes in coolant temperatures, and operation of the connected systems, including the Stirling engines, and the neutron monitoring systems.

Near the end of the 100% power plateau, a demonstration of the ability of the plant to operate continuously and at essentially full power will likely be conducted. The trip at the end of the 100% power plateau signals the completion of the testing portion of the Startup Testing Program. The program will be documented in a 'Startup Report' for future reference.

11.2 Operation

The MARVEL reactor, designed for a 2-year operational life, follows a prescribed sequence of actions during a typical 3-day normal operation to ensure smooth functionality and safety. The details of this operation process are as follows:

1. Preparation for Startup: The operator first initiates the Stirling cooling systems pumps to facilitate heat rejection from the Stirling engines. The reactor, at this stage, is at a hot standby with an average coolant temperature of approximately 200°C, maintained from the previous weekend run by decay heat generated by the system. This temperature is further supplemented by the CIA heaters.
2. Start of Operation: The CIA heaters are turned on to gradually increase the average coolant temperature until the Stirling engines begin operating. Subsequently, the operator ensures the functionality of the power generation system by verifying that the Stirlings and their heat rejection loops are operating properly.
3. System Checks: The next step involves verifying the functionality of all critical monitoring, instrumentation, and control systems. These systems play a pivotal role in ensuring the reactor's safe and efficient operation.
4. Transition to Power Generation: With the checks completed, the CIA heaters are turned off, allowing the system to cool gradually until the Stirling engines can be shut off at their lower threshold temperature. Following this, the control drum startup sequence is initiated to gradually introduce nuclear fission heat, increasing from 0% power to 100% power over an approximate duration of one hour.

5. Steady State Operation: The reactor can then be maintained at a steady state for the duration of any applications tests, which typically last around 12 to 24 hours over a normal weekend. However, the reactor's design allows for longer operational runs, as long as they do not interfere with the Transient Reactor Test (TREAT) facility's regular operational schedule.
6. Shutdown Procedures: After the completion of all applications tests, the reactor enters the shutdown phase. The operator follows the shutdown sequence, performing a manual scram to shut down the reactor and allowing the system to cool until the Stirling Engines automatically shut off at the minimum threshold temperature.
7. Maintenance of Hot Standby: In the days following the shutdown, the CIA heaters utilize PID controllers to balance heat loss with decay heat and heater power, maintaining the system coolant in a hot standby state at an average temperature of approximately 200°C.

Detailed simulations of this normal operation process can be found in the ECAR-6332 document, conducted using the RELAP5-3D software.

Note: The procedures and checks outlined in the operation process are crucial for ensuring the reactor's safety and optimal performance during its 2-year operational life. The MARVEL team closely adheres to these steps, ensuring the reactor's reliable functioning while conducting applications tests and research.

12. MAINTENANCE STRATEGY

Like most reactors, MARVEL will involve some maintenance aspect. However, since the mission life of the MARVEL system is 2 years, regardless of capacity factor, the level and frequency of maintenance is limited compared to traditional powerplants. Those key maintenance activities are highlighted as follows:

12.1 Overhead Gas Sampling

In the design of the MARVEL reactor, the potential for cladding breaches during the reactor's standard operational lifespan is considerably minimized. However, as a precautionary measure and to ensure optimal reactor safety, it is vital that operators possess the capability to detect any possible cladding breach. Instead of relying on continuous, in-situ live monitoring tools embedded within the primary coolant system, MARVEL has adopted a different approach. Every week, post operational run, a gas sample is drawn from the overhead space of the primary coolant system. This procedure is dedicated to evaluating the gas's chemical composition.

The significance of this process lies in its ability to detect any presence of fission products. The identification of such products serves as a potential indicator of a cladding failure, as it can suggest that fission product gases have been released into the liquid NaK coolant. Further, this gas sampling can be analytically invaluable, offering relatively accurate insights based on burnup calculations and release fractions. In essence, these samples not only alert operators to immediate failures but also provide diagnostic data that can guide further investigations and responsive actions.

12.2 Stirling Engine Replacement

This 90% final design report section outlines the maintenance procedure for the Stirling engine used in the MARVEL (name of the system or facility) power generation system. The Stirling engine, a critical component in the system, is prone to failures due to corrosion and radiation damage. The purpose of this section is to provide detailed guidance on the replacement process, which is expected to be a complex and frequent maintenance activity on MARVEL. The section aims to ensure the safe and efficient replacement of the Stirling engine, thereby maintaining the system's optimal performance.

The Stirling engine is situated above the plenum top plate of the reactor, with its hot end heat exchanger tubes submerged into the intermediate heat exchanger that contains the Ga-In-Sn eutectic secondary coolant. To mitigate corrosion, the overhead gas space is confined with a helium blanket. Despite these preventive measures, Stirling engines are anticipated to fail under normal operating conditions due to the corrosive environment and radiation-induced damage to the Hall effect sensor, as documented in ECAR-6573, "PGS Stirling Engine Radiation Lifetime Estimate."[46] The Stirling engine replacement procedure (Figure 76) involves the following steps:

Step 1: Preparing for Maintenance- The replacement operation must be conducted at least two weeks after the shutdown of the system. This time frame allows for gamma radiation levels to subside to acceptable levels, ensuring the safety of personnel during the replacement.

Step 2: Disconnection of Actuators- To begin the replacement, the control drum and central insurance absorber actuators connected to the Stirling engine's shaft are disconnected and removed.

Step 3: Removal of Top Shield- The top shield that covers the Stirling engine is rolled off using the guided tracks on the lid, revealing the upper confinement.

Step 4: Unbolting Upper Confinement Lid- The upper confinement lid is unbolted and removed, providing access to the Stirling engine.

Step 5: Detachment and Lifting of Stirling Engine- Three bolts securing the Stirling engine to the reactor must be removed using a long handling tool. Subsequently, the Stirling engine is lifted 45 inches vertically upwards. It is crucial to ensure that fluid pipes, power, and instrument cables connected to the engine have enough slack to accommodate the lifting process safely. The engines may need depressurization by releasing the helium before lifting.

Step 6: Replacement of Orange Bracket- A total of 12 bolts are used to secure the orange bracket, an essential component for the Stirling engine's proper functioning. This bracket must be removed from the old engine and attached to the new Stirling engine.

Step 7: Welding of Flange Burner- Before lowering the new Stirling engine back into the reactor, the flange burner (highlighted in red) must be welded onto the engine securely.

Step 8: Reconnection of Cooling Pipes and Cables- The new Stirling engine is then connected to its cooling pipes, instrument lines, and power cables, ensuring a proper and secure fit.

Step 9: Purging and Sealing- The purged cooling line is re-filled with water, and excess air in the line is purged off to maintain optimal coolant flow and prevent airlocks. The overhead gas space is filled with helium to create a chemically inert environment, reducing the risk of corrosion.

Step 10: Disposal of Old Stirling Engine- The used Stirling engine, considered low-level waste, is safely hauled away in a shielded cask for proper disposal.

The maintenance procedure for the Stirling engine on MARVEL is a complex but essential activity to ensure the reliable and continuous operation of the power generation system. By following the outlined steps diligently and adhering to safety protocols, operators can successfully replace the Stirling engine. Regular and well-executed maintenance will contribute to the longevity and efficiency of the entire MARVEL system, providing a stable and sustainable power source.

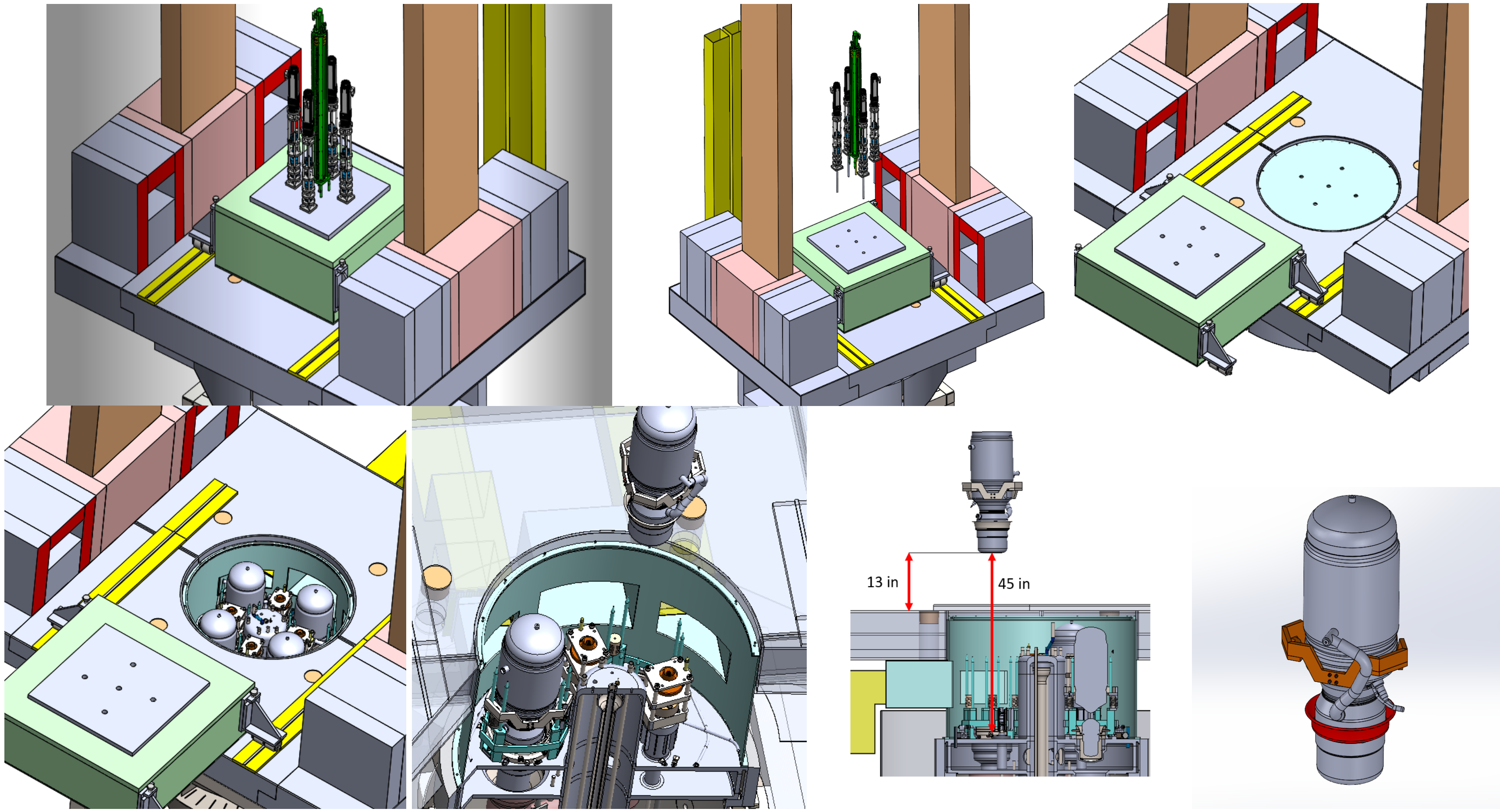

Figure 76. Schematic of Stirling Engine Replacement Sequence.

13. DECONTAMINATION AND DECOMMISSIONING

When radiation levels are low enough for safe access (dependent on power history), MARVEL sub-systems will be decommissioned and the NaK primary coolant will be drained (either by pumping or by vacuum evacuation). NaK treatment will convert the sodium and potassium to their respective hydroxide forms using chemical reactions with high temperature steam and an appropriate cover gas, likely nitrogen. Conversion of the sodium and potassium to their respective carbonate forms via chemical reaction with carbon dioxide will follow. The resultant aqueous carbonate solution will then be solidified for disposal Residual NaK will be deactivated by introducing in an inert gas purge, followed by a water wash which will generate an alkali solution of potassium and sodium hydroxide. Deactivation of the NaK allows disposal at an off-site mixed waste Subtitle D or Subtitle C disposal facility, which are readily available. NaK removal is required prior to defueling the reactor.

Originally, the IHX was intended to contain lead-bismuth eutectic (LBE) which would be a solid at decommissioning temperatures. Changing the secondary coolant to GaInSn eutectic eliminated Polonium-210 concerns from activation of the LBE and simplified removal of the Stirling engines and IHXs. Therefore, after draining of the GaInSn eutectic, the Stirling engines and IHX can be removed without significant physical or health and safety concerns. The vessel head can then be removed, and the reactor can be defueled.

Preliminary criticality and radiation shielding evaluations for various transfer and storage configurations of the 36 MARVEL microreactor fuel elements (spare elements would be contact handled and easily transferred) indicate that various transfer casks currently in use at INL are suitable for the transport of MARVEL used fuel. The Advanced Test Reactor transfer cask, HFEF-5 transfer cask, and the High Load Charger are used for the transfer of irradiated fuel between INL facilities, and these casks could be used to transfer irradiated MARVEL microreactor fuel between INL facilities. A more detailed analysis will select the best cask and other specific evaluations, such as a cask-specific criticality analysis, will be needed to prepare a transport plan but no significant obstacles were identified.

The transport of MARVEL used fuel to RSWF or to INTEC would likely be an out-of-commerce shipment consistent with a tailored, specific transport plan or functional equivalent. SNF transportation of this type is performed periodically within INL and does not pose a significant risk to the disposition of the MARVEL used fuel. Storage at any of the INL facilities constitutes long-term interim storage. Final disposal options for many used fuels, including TRIGA used fuel, were identified as part of the Yucca Mountain Repository project. The project assumes that MARVEL fuel will be permanently stored at a national geologic repository when identified and constructed.

After the core is de-fueled, the nuclear instruments will be disconnected and disposed of or stored for re-use. Power to in-pit systems will be disconnected, and reactivity control systems will be removed and disposed of separate from the reactor vessel. The activated BeO can also be removed and managed under TREAT's beryllium management program for use in subsequent demonstrations.

Following de-fueling and after removing instrumentation, the reactor vessel will be removed and grouted if needed for disposal at the ICDF. Size reduction, as identified in the EA, remains a back-up disposition technique that would require contamination controls such as tents and active ventilation. If the entire vessel is disposed of intact (adopted as the planning basis for the project), a transfer vessel may be required. or a bottom-loading cask may be used. The reactor pit shielding can be removed for storage if needed, but it is assumed it will remain in place for use in subsequent demonstrations.

Further analysis and trade studies are needed to identify the details of the decommissioning. However, the conclusion that all radioactive waste, other than the reactor fuel, generated in this phase will be NRC Class A LLW or MLLW and has current disposition paths in DOE or commercial facilities is valid. Further detail is needed prior to D&D but the final MARVEL design adequately provides enough flexibility to support execution of the identified D&D options.

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Appendix A
Principal Design Criteria

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Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
Section I-Overall Requirements						
1	Quality Standards and Records.	Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 1 is required under: • 10 CFR 830.121, which requires a quality management program. • 420.1C, Attachment 3, Section 3.a.(7), "Quality Assurance." [Note Section numbering error in Attachment 3, Section 3 of reference document. Numbering in this table column reflects correction.] • DOE O 414.1D, which establishes requirements for quality assurance.	MARVEL design, construction and operation complies with all applicable quality requirements as outlined in program description document (PDD)-13000, "Quality Assurance Program Description."[41] MARVEL SSCs designated in Section 3.2 as SR and NSR-AR are treated with an appropriate graded approach. The generally recognized codes and standards used are documented in the "MARVEL Code of Record." MARVEL design, construction and operation complies with applicable quality requirements as outlined in PDD-13000. All MARVEL records are identified and managed within the requirements of PDD-13000 as implemented by the MARVEL program. Retention of MARVEL program and lab generated records is on the Idaho National Laboratory (INL) records management system.
2	Design Bases for Protection Against Natural Phenomena	Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 2 is required by: • DOE 420.1C which requires NPH mitigation in Chapter IV. • DOE-STD-1020 sets the specific requirements for the analysis.	The requirements for natural phenomenon hazard (NPH) design from DOE O 420.1C and DOE-STD-1020-2016, "Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities," were followed for the design and construction of the MARVEL as applicable and appropriate. Consistent with the requirements of DOE O 420.1C and DOE-STD-1020-2016, NPH design criteria were established by performing a conservative estimate of the potential unmitigated consequences as a result of NPH induced failure of MARVEL SSCs. Corresponding NPH criteria were established based upon the potential for the specific NPH to create a postulated accident event and the corresponding consequence of the identified event. Based on the evaluation in ECAR-5127, "Evaluation of the MARVEL Reactor Inhalation Dose Consequences," MARVEL SR and NSR-AR SSCs are categorized using the criteria in DOE-STD-1020-2016 as NPH design category (NDC)-2 and seismic design category (SDC)-2.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
3	Fire Protection	Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, or components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 3 is required by DOE 420.1C Attachment 3, Section 3.a.(6) "Protection Against Fire" which invokes requirements set by DOE-STD-1066 fire protection.	Reliability and survivability of identified MARVEL SSCs designated in Section 3.2 as SR and NSR-AR from fires were considered in the MARVEL design efforts as applicable and appropriate. Consistent with the requirements of DOE STD 1066-2016, "Fire Protection," an evaluation of the potential for failure of SR-SSCs and NSR-AR SSCs was performed in the fire hazards analysis and appropriate preventive and mitigative measures to ensure survivability were included into the MARVEL design as applicable and appropriate. MARVEL construction materials were selected for compatibility with NaK to minimize energetic NaK reactions in areas where NaK leaks are present as well as minimizing to the maximum extent practicable, locations where NaK and water may be proximately located. The MARVEL contains appropriate fire detection and suppression systems to identify, detect, alarm, and mitigate both NaK and combustible material fires. Specific evaluation and selection of appropriate NaK fire mitigation strategies were performed based upon the varying physical locations and associated hazards. Fire suppression systems are designed and evaluated in such a manner as to minimize their potential adverse impact on overall safety and reliability. Design solutions have been selected which most effectively ensure the prevention or ability to mitigate potential fire scenarios while ensuring that reactor safety and overall risk is reduced.
4	Environmental and Dynamic Effects Design Bases	Structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents, including the effects of liquid sodium and its aerosols and oxidation products. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. Chemical consequences of accidents, such as sodium leakage, shall be appropriately considered for the design of structures, systems, and components important to safety, which must be protected.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	Structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, anticipated operational occurrences, and postulated accidents, including the effects of NaK and its aerosols and oxidation products. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	Criterion 4 is required by: • DOE 420.1C Attachment 3, Section 3.a.(3), "Environmental Qualification." • DOE-STD-1020 invokes requirements on external projectile hazards.	Appropriate environmental conditions applicable to MARVEL SSCs designated in Section 3.2 as SR and NSR-AR under all normal operation, maintenance, testing, anticipated event, and postulated accident conditions, were identified in the design process, as well as criteria for evaluation and demonstration of ability of SSCs to perform their safety functions. MARVEL SSC failures and associated impacts on safety systems were considered to ensure that lower classified or other similarly classified systems would not fail in such a way that the ability to perform safety functions are compromised. Pipe whip is not considered a MARVEL concern compared to existing LWRs due to the much lower operating pressure. Environmental conditions due to NaK leakage or approved NaK operating conditions were included and evaluated in the MARVEL preliminary design.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
5	Sharing of Structures, Systems, and Components	Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 5 is required by DOE 420.1C Attachment 3, Section 3.a.(5), "Support System and Interface Design."	MARVEL is a single unit microreactor located in the TREAT facility in the north high-bay equipment pit. Shared SSCs between the TREAT facility and MARVEL were evaluated in the preliminary design to ensure a postulated accident in one reactor will not inhibit the SR and NSR-AR SSC safety functions in the other.
Section II—Multiple Barriers						
10	Reactor Design	The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Application to Criterion 10 is strengthened by: • DOE O 420.1C Attachment 3, Section 3.a.(1) "Conservative Design Margins." The hierarchy of controls imposed by DOE O 420.1C denotes the boundary closest to the radionuclide should be considered which is typically the fuel.	MARVEL SSCs necessary for reactivity control, heat removal, and radioactive material confinement were designed in conjunction with analyses to ensure that fuel, clad, and PCB temperature limits were clearly established and that the design is capable of keeping the reactor within the established limits for all normal operations, anticipated events, and postulated accident conditions.
11	Reactor Inherent Protection	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.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Beyond design basis event considerations also support this GDC by making sure that the potential for small error that leads to a "runaway" event is avoided in the design.	The MARVEL reactor is designed to limit reactivity transients by means of negative inherent reactivity feedback (IRF). The IRF safety function, via geometric and physics changes, is to promote a system performance that provides a negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted and the reactor is brought to new stable state before fuel, clad, and PCB temperature limits are challenged, or before core damage occurs during anticipated events and postulated accident conditions.
12	Suppression of Reactor Power Oscillations	The reactor core; associated structures; and associated coolant, control, and protection systems shall be designed to ensure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	N/A	The MARVEL reactor core is designed for axial and radial stability. The reactor core and associated coolant, control, and protection systems ensure that power and hydraulic oscillations that can result in conditions exceeding fuel, clad, and PCB temperature limits are not possible.
13	Instrumentation and Control	Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the primary coolant boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 13 is required by DOE 420.1C Attachment 3 Section 3.b.(6) which requires specific special treatments for instrumentation and controls as well as DOE human machine interface (HMI) requirements.	MARVEL instrumentation and control (I&C) hardware is designed to monitor variables and systems over the anticipated ranges for normal operation, anticipated events, and postulated accident conditions. A reactor instrumentation and control system is provided to ensure that MARVEL operations and controls are performed in a manner which keeps plant parameters and systems within their prescribed operating ranges. The only instrument that is SR is the pressure indication, which verifies the pressure in the guard vessel and reactor barrel can perform their safety function to prevent the core from being uncovered in the case of a large break of the primary coolant loop. The same I&C hardware used for normal operations is also used for post-accident monitoring. The limitations imposed by the interlocks and software will keep the operations within the prescribed operating ranges, either through inherent physical means or defense in depth measures.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
14	Primary Coolant Boundary	The primary coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 14 is required by DOE 420.1C Attachment 3 Section 3.b.(2) which requires designs to meet current applicable codes and standards but allows for selection of appropriate codes and standards based upon coolant boundary function and risk as opposed to an *a priori* decision absent technical details.	MARVEL PCB SSCs, described in detail in Addendum Chapter 5, Reactor Coolant System, have been designed to ensure that they are capable of withstanding the normal operating conditions, anticipated events, and postulated accident conditions as applicable and appropriate.
15	Primary Coolant System Design	The primary coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 15 as identified by necessary safety functions is required by DOE O 420.1C which imposes special treatments as necessary on the coolant design to provide “conservative design margins” and DID as necessary (Attachment 2 Chapter 1 Section 3.b.(2)).	The design of the MARVEL primary coolant system (PCS) system, described in detail in Addendum Chapter 5, Reactor Coolant System, considers appropriate conservatism and factors of safety to ensure that margins are maintained for normal operations, anticipated events, and postulated accident conditions.
16	Containment Design	A reactor containment consisting of a low-leakage, pressure retaining structure surrounding the reactor and its primary cooling system shall be provided to control the release of radioactivity to the environment and to ensure that the reactor containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The containment leakage shall be restricted to be less than that needed to meet the acceptable onsite and offsite dose consequence limits, as specified in 10 CFR 50.34 for postulated accidents.	Rewritten as new PDC to reflect the overall MARVEL functional confinement strategy.	A reactor functional confinement strategy, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactive and hazardous material to the environment, and to ensure that the functional confinement barrier design conditions are not exceeded for as long as postulated accident conditions require.	Criterion 16 as applicable to FSFs is required by DOE O 420.1C as a special treatment and analysis of DID (Attachment 3 Section 3.a.(1)).	The MARVEL functional confinement strategy is derived from a performance-based perspective in that the confinement barrier performance requirements are derived from the MARVEL accident analysis and not prescriptively required such as for a typical LWR pressure-retaining containment structure. The evaluated MARVEL design has the following strategies for limiting the release of radionuclides: • Fuel, cladding, and primary coolant (NaK), • PCB including reactor barrel and piping (downcomers), • Guard vessel, top confinement structure, and • T-REXC SSCs

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RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
17	Electric Power Systems	Electric power systems shall be provided when required to permit functioning of structures, systems, and components. The safety function for each power system shall be to provide sufficient capacity and capability to ensure that (1) that the design limits for the fission product barriers are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of postulated accidents. The electric power systems shall include an onsite power system and an additional power system. The onsite electric power system shall have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. An additional power system shall have sufficient independence and testability to perform its safety function. If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 17 is required by • DOE O 420.1C invokes special treatment for electrical power to follow IEEE codes and standards (Attachment 3 Section 3.b.(5). • DOE O 420.1C requirements for safety SSC classification will require elevation of the electrical system classification if it is necessary for other safety systems to perform their safety function (Attachment 3 Section 3.a.5.(a).	RG 1.232 provides some clarification on this topic by stating, "In this context, important to safety functions refer to the broader, potentially non-safety related functions such as post-accident monitoring, control room habitability, emergency lighting, radiation monitoring, communications and/or any others that may be deemed appropriate for the given design." MARVEL system passive design and construction is such that normal electrical power is provided to all systems where such capability is required for appropriate system functions. Normal power to the TREAT facility is discussed in detail in SAR-420, Chapter 8. Normal power is site commercial power, supplied over a pole line from the Materials and Fuels Complex (MFC)-768 Power Plant 13.8-kV system. Electrical power does not intrinsically perform a FSF in the MARVEL design. MARVEL is designed with passive SR SSCs for safe shutdown, core flow/heat removal, and confinement boundary integrity. Electrical power is not relied upon to meet design limits for the fission product barriers as a result of anticipated events or postulated accidents. The availability of electrical power sources does not affect the ability to achieve and maintain SR functions. NSR-AR electrical system SSCs have been identified in Addendum Chapter 15 to provide power to NSR-AR instruments and monitoring panels for monitoring safe plant shutdown from the control room and for post-accident monitoring.
18	Inspection and Testing of Electric Power Systems	Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 18 satisfied by DOE O 420.1C Attachment 2 Chapter 1 Section 3.b.(4)(b) which requires "facilities must be designed to facilitate inspections, testing, maintenance, repair, and replacement of safety SSCs as part of a reliability, maintainability, and availability program with the objective of maintaining the facility in a safe state."	There are no SR electric power SSCs identified in the Addendum Chapter 15 Accident Analyses, required to meet this PDC. NSR-AR SSCs have been evaluated for meeting this Criterion as discussed in greater detail in Addendum Chapter 8, Electric Power Systems.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
19	Control Room	A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent, as defined in § 50.2 for the duration of the accident. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions. Adequate protection against sodium aerosols shall be provided to permit access and occupancy of the control room under accident conditions. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.	Revised to reflect that the MARVEL design employs NaK as the primary coolant and to revise the control room dose limit to reflect the Evaluation Guidelines in Addendum Chapter 15.	A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of the DOE Evaluation Guidelines in Addendum Chapter 15. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions. Adequate protection against NaK aerosols shall be provided to permit access and occupancy of the control room under accident conditions. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.	Criterion 19 is not required by guidance in DOE O 420.1C. In the event that manual scram is required to shutdown the reactor then DOE O 420.1C provides direction to use appropriate ANSI/ANS standards on time response and design criteria to ensure those actions can be fulfilled.	The MARVEL control room will be collocated with the TREAT control room. There are no MARVEL credited SR operator actions under postulated accident conditions. Shutdown and monitoring functions of the control room are not required by the accident analyses in Addendum Chapter 15 for accident mitigation. The accident analyses in Addendum Chapter 15, identifies that manual scram by the control room operator may be required for long term shutdown of the reactor under postulated accident conditions, and manual scram SSCs are therefore SR. The control room has a manual scram button to allow the operator to cut the power to the electromagnets and scram the reactor. This action can achieve cold shutdown, which bounds the hot shutdown. Radiological and non-radiological consequences to workers are limited by the distance from the TREAT building to the TREAT control room, and by evacuation of the TREAT building during transient and MARVEL operations. Radiological and non-radiological consequences to workers are well within the evaluation guidelines in Addendum Chapter 15 at the control room and no measures are required for ensuring habitability.
Section III—Reactivity Control						
20	Protection System Functions	The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 20 is required by DOE 420.1C Attachment 2 Chapter 1 Section 3.b.(2)(f) 2 which requires “maintaining processes in safe status” and “providing preventative/mitigative controls for accidents….”, which imposes a requirement for a system design that automatically (through passive or active means) controls heat generation or maintains cooling to ensure that radionuclide retention is maintained.	MARVEL is provided with a reactor protection system (RPS) to ensure reactor shutdown under anticipated events and postulated accident conditions. Design and performance requirements for the RPS have been developed to meet the requirements for safety systems from associated standards based upon the safety functions identified in the hazard and accident analysis. The use of standards cited by and referenced in DOE O-420.1C are used on a graded basis, based on the function of the SSC and importance in the accident analysis. Addendum Chapter 15 takes no credit for all automatic systems and sensing of accidents, with the exception of the seismic sensor. The seismic sensor is SR and is an input to the RPS. When the P-wave reaches the seismic sensor, it will release a relay which will release the electromagnets and shutdown the reactor. There are several automatic and accident sensing trips that serve as DID and are called computer trips.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
21	Protection System Reliability and Testability	The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 21 is required by DOE 420.1C which invokes the single failure criterion for Safety Class Systems (Attachment 3, Section 3.a.(2)).	Consistent with the requirements of DOE O 420.1C and supporting standards, MARVEL RPS SR, and NSR-AR SSCs, as identified in the hazard and accident analysis, include features for high reliability by including requirements and criteria for redundancy and independence. Provisions for ensuring the ability to monitor the status of the RPS and perform appropriate surveillance tests and monitoring functions to assure RPS functionality are incorporated into the plant design and operating philosophy as applicable and appropriate. There are certain aspects of PDC-21 that are not applicable. The MARVEL system is not designed to have any in-service testing or testing with the reactor in operation. The RPS can be tested independently. Removal of any component from the RPS will automatically cause the RPS to perform its safety function because the electromagnets will not be able to be powered. The RPS has been designed with single failure built into the design by analysis and adding redundant and independent features. No single failure in the RPS will result in a loss of the protection function. The MARVEL RPS complies with the single failure criterion. There are no provisions for taking a channel out of service with the reactor operating. No in-service testing or maintenance is planned. The RPS will fail into a safe state upon loss of power or disconnection. The RPS SSCs are designed so that the loss of power will automatically result in a shutdown of the reactor. Commercial power reactors need detailed reliability analysis to demonstrate that the protection systems can reliably mitigate the accidents described in the accident analyses. The RPS for MARVEL is not required to mitigate any accidents described in Addendum Chapter 15. The reliability of the MARVEL RPS is achieved by use of IEEE-379, IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems. The MARVEL RPS does not need a formal, detailed reliability analysis, so IEEE-352 is not applicable. This SFR-DC was written for commercial power stations that are expected to operate for long periods of time without shutdown. MARVEL is planned to operate only for a few days at a time, and to be regularly shutdown. This criterion explicitly allows for exception to testing and calibration during power operation are allowed where this capability cannot be provided without adversely affecting the safety or operability. The MARVEL safety related I&C system can be testing and calibrated when MARVEL is shutdown and maintain sufficient reliability.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
22	Protection System Independence	The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 22 is required by: • DOE-STD-1020 which requires safety SSCs to meet NPH requirements. • DOE O 420.1C which requires DID and Environmental Qualification (Attachment 2 Chapter 1 Section 3.b.(2) and Attachment 3 Section 3.a.(3)).	Consistent with PDC-22, MARVEL RPS SR and NSR-AR SSCs identified in the Addendum Chapter 15 hazard and accident analysis are subject to the appropriate requirements for NPH qualification. No RPS SSCs have been identified in Addendum Chapter 15 that are subject to PDC-22 requirements for functional and component diversity. This is because all the postulated accidents rely on passive means to mitigate the accident until a manual shutdown can be performed by an operator to result in a cold shutdown.
23	Protection System Failure Modes	The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, sodium and sodium reaction products, pressure, steam, water, and radiation) are experienced.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, NaK and NaK reaction products, pressure, steam, water, and radiation) are experienced.	Criterion 23 is required by DOE 420.1C GDC 4 which requires safe failure modes for SSCs (Attachment 3 Section 3.a.(4)).	MARVEL RPS SR and NSR-AR SSCs, identified in the hazard and accident analysis, are subject to the appropriate requirements for either safe failure mechanisms. The RPS is designed to fail with the result being a reactor shutdown. Likewise, the reactivity control system (RCS) is designed to result in a safe state with loss of energy.
24	Separation of Protection and Control Systems	The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 24 is required by DOE 420.1C GDC 5 which requires system reliability in GDC 2 (Attachment 3 Section 3.a.(5)).	MARVEL RPS SR and NSR-AR SSCs, identified in the hazard and accident analysis, are subject to the appropriate requirements for redundancy and diversity in sensors as well as the manner in which they are connected precludes the ability for a single failure within either system from adversely affecting the performance of the other system. MARVEL RPS SR and NSR-AR SSCs, identified in the Addendum Chapter 15 hazard and accident analysis, are subject to the appropriate requirements for interconnection such that failure of lower safety classified systems cannot have an adverse impact on the ability of the RPS to perform its designated safety functions. The RPS will be separated from the control system to the extent that failure of any single component or channel, or failure or removal from service of any single RPS component or channel which is common to the control system or RPS leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. The forcing mechanisms to control the CDs and CIA rod are independent from the RPS system which releases the electromagnetic clutch and separates the control systems from the protection systems. This clutch defaults to the disconnection with no power. No postulated failure has been found to challenge the ability of the SSCs from performing their safety functions.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
25	Protection System Requirements for Reactivity Control Malfunctions	The protection system shall be designed to ensure that specified acceptable fuel design limits are not exceeded during any anticipated operational occurrence accounting for a single malfunction of the reactivity control systems.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 25 is required by DOE O 420.1C GDC 1 which requires the application of a conservative design margin (Attachment 3 Section 3.a.(1)).	MARVEL RPS SR and NSR-AR SSCs, identified in the hazard and accident analysis, are subject to the appropriate requirements to ensure that any single malfunction will not impair the ability of the system to perform its associated safety function, which will result in ensuring that no design limits for the fission product barriers are challenged. The design of the interlock relays, the hard stops and the procedural requirements on only one CD above the critical banked position were designed for the purpose of restricting a single malfunction of the reactivity control system. The interlock relays force only one CD to be moved at once. The procedures limit only one CD to be moved beyond critical so that the reactivity between critical and the hard stop are the total excess reactivity that can be achieved with a malfunction. The hard stop restricts the reactivity to a level below the TOP transient limit in Addendum Chapter 15. Additionally, there are DID measures from the control computer to enforce operations and reactivity limits with automatic shutdowns if those limits are exceeded.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
26	Reactivity Control Systems	A minimum of two reactivity control systems or means shall provide: (1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the design limits for the fission product barriers are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences. (2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the design limits for the fission product barriers are not exceeded. (3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident. (4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	Rewritten as new PDC to reflect the overall MARVEL passive reactivity control strategy.	A minimum of two independent reactivity control systems or means of different design principles shall be provided. 1) One of the means shall be capable of reliably controlling reactivity changes to ensure that under conditions of normal operation, including anticipated events, and with appropriate margin for malfunctions, design limits for the fission product barriers are not exceeded. 2) The second means shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes to ensure design limits for the fission product barriers are not exceeded. 3) One of the means shall be capable of holding the reactor core subcritical under cold conditions. 4) The means shall be designed to have a combined capability to ensure that under postulated accident conditions, excess reactivity does not cause the design limits for the fission product barriers to be exceeded and that the reactor core is subcritical with appropriate margin	Wording in SFR-DC 26 revised to reflect DOE O 5480.30 for DOE reactors.	The importance of reactivity control is that it is the means to control the generation of heat in the reactor. Imbalances between the heat generation and the heat removal in the reactor core lead to changes in core temperatures. The FSF of controlling reactivity is intended to control normal plant operation and to prevent abnormal plant conditions from escalating into a more significant accident. Reactivity control also helps facilitate any response to a postulated accident, should one occur, by shutting down the nuclear reaction and reducing the heat generation within the plant that other installed systems (e.g., CDs) would be required to mitigate. The evaluated MARVEL micro-reactor design has the following strategies for reactivity control: • IRF • CDs • CIA rod The first and primary means of limiting fuel temperature during postulated accident conditions is by the control of the reactivity of the reactor through the passive insertion of negative reactivity. IRF is not a physical SSC but relies on core system SSCs to provide system performance related to geometric and physics changes in order to provide negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted and the reactor is brought to new stable state before fuel, clad, and PCB temperature limits are challenged and core damage occurs. The CDs release following a signal from the RPS to provide insertion of negative reactivity to shutdown the reactor and maintain in shutdown condition. The reactor trip system monitors reactor process variables and sends a reactor trip signal when a process variable exceeds a limit setpoint. The CD system is designed to limit both the rate and magnitude of reactivity insertion that the system can achieve so as to minimize the effect of an unintended reactivity insertion. The CD system consists of four independent mechanical assemblies evenly spaced within the radial neutron reflector around Marvel's core. A single CD can bring the reactor subcritical in a hot operation condition. This provides excellent redundant shutdown capability as there are four independently controlled CDs. The CIA rod provides DID reactor shutdown capability in the beyond design basis scenario that common mode failure prohibits insertion of more than one CD. The CIA rod alone can bring the reactor subcritical in all credible accident scenarios at a hot operation condition. The CIA rod plus one CD allows for shutdown of the reactor in all conditions.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
28	Reactivity Limits	The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the primary coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 28 is required by DOE O 420.1C which requires conservative design margins (Attachment 3 Section 3.a.(1)).	The following design and administrative controls limit reactivity: 1) The interlock relays limit movement to 1 CD at a time (design control), 2) The hard stop limits the max reactivity of a CD (design control), 3) The operations must only move one CD above critical (administrative control), and 4) The gray rod limits total excess down to give margin for errors in calculations (design control).
29	Protection Against Anticipated Operational Occurrences	The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 29 is required by DOE O 420.1C which invokes reliability and environmental qualification requirements. Additionally, safety analysis and classification requirements will ensure that reliability of systems that prevent accidents is sufficient to address anticipated events (Attachment 3 Section 3.a.(2) and (3)).	MARVEL RPS and reactivity control system SR and NSR-AR SSCs, identified in the hazard and accident analysis, are high reliability systems which are designed to minimize plant upset conditions and ensure performance of safety functions during anticipated events and postulated accident conditions. The anticipated events are listed in Addendum Chapter 15. Each of the initiating events are presented below along with how PDC-29 was considered for that phenomena and that apply to this system.
			Section IV—Fluid Systems			
30	Quality of Primary Coolant Boundary	Components that are part of the primary coolant boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of primary coolant leakage.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 30 is required by DOE O 420.1C (Attachment 3 Section 3.a.(7)) and DOE O 414.1D as a QA programmatic requirement. DOE O 420.1C also provides flexibility in allowing the specific quality requirements to be graded consistent with their safety and mission impact as opposed to an a priori decision that all of the coolant boundary is of high safety consequence.	The MARVEL PCB has appropriate design and procurement requirements which are verified by adherence to the MARVEL quality assurance program. The potential for leakage from the PCB within the MARVEL is extremely limited, however the need for provisions for detecting the leakage and to the extent practicable the location of the leakage was evaluated in the preliminary design. I&C instrumentation includes NaK leak detectors that are provided to 1) signal triggered when in contact with NaK, 2) detect a NaK leak into the guard vessel, and 3) detect if the NaK that has leaked into the guard vessel has reduced the NaK inventory in the primary to uncover the core. The pressure difference between the guard vessel and the reactor barrel is monitored to keep the core from being uncovered in a LOCA event. A discrepancy in the guard vessel pressure indicates a leak in the guard vessel, a leak in the reactor barrel or a faulty pressure sensor. If there is a leak and the pressures change drastically, the operator will be expected to press the manual scram button. The pressure sensors are qualified to NQA-1 standards as safety related or through commercial grade dedication to ensure fabrication and testing meet high quality standards.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
31	Fracture Prevention of Primary Coolant Boundary	The primary coolant boundary shall be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady–state, and transient stresses, and (4) size of flaws.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 31 is required by DOE O 420.1C GDC 1 which requires conservative design margins as well as performance analyses that demonstrate safety SSC performance under appropriate Environmental Qualification conditions (Attachment 3 Section 3.a.(1)).	The performance of the MARVEL PCB under stressed condition is ensured by selection of materials which behave in a ductile manner over the range of MARVEL operating temperatures. The MARVEL PCB design material selection is focused on ensuring that the selected materials can perform their identified design functions under all anticipated events and postulated accident conditions and are subject to the appropriate environmental qualification requirements. The PCB is a metal weldment made from 316H stainless steel for high temperature reactors designed in accordance with ASME Section III Division 5. Finite element analysis (FEA) of the PCB has confirmed the design limits of ASME BPVC Section III Division 5 paragraph HBB-3222.1 have not been exceeded; and (2) load-controlled stress limits and strain and deformation limits for Level A, B, C, and D Service Loadings per paragraphs HBB-3223, HBB-3224, HBB-3225, and Nonmandatory Appendix HBB-T will be managed by the operating conditions of the reactor. The FEA analysis conservatively envelopes the unprotected transient overpower (UTOP) conditions for primary stress in the design limit due to the low operating pressure of the reactor. In addition, the FEA analysis accounts for secondary stresses along with the combined primary membrane plus bending stress intensity to show that the design limits of paragraph HBB-3222.1 have not been exceeded. PCB 316H is expected to have essentially an unlimited lifetime in NaK (no corrosion), given the anticipated purity of the given NaK at 1 ppm oxygen level (Vol V Sodium and NaK Handbook). At present it is planned to control the oxygen content in NaK through (i) a barrel vacuum-purge prior to NaK fill and (ii) use of argon cover gas after NaK fill.
32	Inspection of Primary Coolant Boundary	Components that are part of the primary coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 32 is required by DOE O 420.1C Attachment 2 Chapter 1 that requires the ability to periodically inspect and test SSCs along with the requirements of a QA program.	A combination of boundary monitoring, inspections and functional testing will be performed to ensure adequate performance of the MARVEL PCB as described in Addendum Chapter 5, Reactor Coolant System.
33	Primary Coolant Inventory Maintenance	A system to maintain primary coolant inventory for protection against small breaks in the primary coolant boundary shall be provided as necessary to ensure that specified acceptable fuel design limits are not exceeded as a result of primary coolant inventory loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that the system safety function can be accomplished using the piping, pumps, and valves used to maintain primary coolant inventory during normal reactor operation.	SFR-DC 33 is not applicable as MARVEL will not require NaK inventory maintenance.	N/A	Not applicable due to the MARVEL passive core flow and decay heat removal design.	N/A

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
34	Residual Heat Removal	A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the primary coolant boundary are not exceeded. Suitable redundancy in components and features and suitable interconnections leak detection, and isolation capabilities, shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	Rewritten as new PDC to reflect the overall passive MARVEL core flow and heat removal strategy.	SSCs responsible for core flow/heat removal shall passively transfer fission product decay heat and other residual heat from the reactor core at a rate such that fuel and vessel temperature limits and the design conditions of the PCB are not exceeded, and safe shutdown is achieved and maintained during normal operation, anticipated events, and postulated accident conditions.	Criterion 34 is required by DOE O 420.1C which requires special treatments of DID and meeting system reliability such as single failure criterion (Attachment 2 Chapter 1 Section 3.b.(2).	The evaluated MARVEL micro-reactor design has two strategies for heat removal: • Natural circulation and active heat removal via the Stirling engines during normal operations and shutdown • Passive conduction through barrel and CDs and natural draft heat removal from the outside surface of the guard vessel to the surrounding ambient air environment. The FSF of removing heat serves two critical objectives: 1) removal of the generated heat during all normal operations and shutdown conditions and postulated accident conditions to assure that equipment would operate within the environmental envelope for which it is designed and qualified, and 2) to prevent a postulated accident from progressing into a more severe event category and, as such, would serve to mitigate the potential for releases of radioactivity from the facility. Core flow SSCs provide structural, mechanical, and geographic spacing to ensure natural circulation through the fuel assemblies at reactor operating and elevated transient temperatures and ensure passive conduction heat transfer to the passive ambient air heat rejection system is possible.
35	Emergency Core Cooling	A system to assure sufficient core cooling during postulated accidents and to remove residual heat following postulated accidents shall be provided. The system safety function shall be to transfer heat from the reactor core during and following postulated accidents such that fuel and clad damage that could interfere with continued effective core cooling is prevented. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	An active emergency core cooling system is not required due to the MARVEL passive core flow and decay heat removal design. Therefore, SFR-DC 35 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL passive core flow and decay heat removal design.	N/A
36	Inspection of Emergency Core Cooling System	A system that provides emergency core cooling shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	An active emergency core cooling system is not required due to the MARVEL passive core flow and decay heat removal design. Therefore, SFR-DC 36 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL passive core flow and decay heat removal design.	N/A
37	Testing of Emergency Core Cooling System	A system that provides emergency core cooling shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of any associated systems and interfaces necessary to transfer decay heat to the ultimate heat sink.	An active emergency core cooling system is not required due to the MARVEL passive core flow and decay heat removal design. Therefore, SFR-DC 37 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL passive core flow and decay heat removal design.	N/A

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
38	Containment Heat Removal	A system to remove heat from the reactor containment shall be provided as necessary to maintain the containment pressure and temperature within acceptable limits following postulated accidents. Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 38 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
39	Inspection of Containment Heat Removal System	The containment heat removal system shall be designed to permit appropriate periodic inspection of important components to ensure the integrity and capability of the system.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 39 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
40	Testing of Containment Heat Removal System	The containment heat removal system shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including the operation of associated systems.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 40 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
41	Containment Atmosphere Cleanup	Systems to control fission products and other substances that may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents and to control the concentration of other substances in the containment atmosphere following postulated accidents to ensure that containment integrity and other safety functions are maintained. Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to ensure that its safety function can be accomplished, assuming a single failure.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 41 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
42	Inspection of Containment Atmosphere Cleanup Systems	The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 42 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
43	Testing of Containment Atmosphere Cleanup Systems	The containment atmosphere cleanup systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including the operation of associated systems.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 43 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
44	Structural and Equipment Cooling	A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided, as necessary, to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 44 is required by DOE O 420.1C which requires environmental qualification of SSCs to meet this criterion (Attachment 3 Section 3.a.(3).	The passive heat removal system safety function is to maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and anticipated events and postulated accident conditions. Passive heat removal system adequacy is document in the thermal-hydraulic analyses in Addendum Chapter 15, and redundancy, assuming a single failure, is not required.
45	Inspection of Structural and Equipment Cooling Systems	The structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to ensure the integrity and capability of the systems.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 45 is required by DOE O 420.1C which requires that designs must permit periodic testing and maintenance (Attachment 2 Chapter 1 Section 3.b.(2)).	The MARVEL system is not designed to have any in-service testing or testing with the reactor in operation.
46	Testing of Structural and Equipment Cooling Systems	The structural and equipment cooling systems shall be designed to permit appropriate periodic functional testing to ensure (1) the structural and leaktight integrity of their components, (2) the operability and performance of the system components, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring the systems into operation for reactor shutdown and postulated accidents, including the operation of associated systems.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 46 is required by DOE O 420.1C which requires that designs must permit periodic testing and maintenance (Attachment 2 Chapter 1 Section 3.b.2)).	The MARVEL system is not designed to have any in-service testing or testing with the reactor in operation.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
			Section V—Reactor Containment			
50	Containment Design Basis	The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 50 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
51	Fracture Prevention of Containment Pressure Boundary	The boundary of the reactor containment structure shall be designed with sufficient margin to ensure that, under operating, maintenance, testing, and postulated accident conditions, (1) its materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 51 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
52	Capability for Containment Leakage Rate Testing	The reactor containment structure and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted to demonstrate resistance at containment design pressure.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 52 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
53	Provisions for Containment Testing and Inspection	The reactor containment structure shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations that have resilient seals and expansion bellows.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 53 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
54	Piping Systems Penetrating Containment	Piping systems penetrating the reactor containment structure shall be provided with leak detection, isolation, and containment capabilities that have redundancy, reliability, and performance capabilities necessary to perform the containment safety function and that reflect the importance to safety of preventing radioactivity releases from containment through these piping systems. Such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 54 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
55	Primary Coolant Boundary Penetrating Containment	Each line that is part of the primary coolant boundary and that penetrates the reactor containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to ensure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 55 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
56	Containment Isolation	Each line that connects directly to the containment atmosphere and penetrates the reactor containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 56 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A
57	Closed System Isolation Valves	Each line that penetrates the reactor containment structure and is neither part of the primary coolant boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve unless it can be demonstrated that the containment safety function can be met without an isolation valve and assuming failure of a single active component. The isolation valve, if required, shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.	As discussed in Criterion 16, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker. Therefore, SFR-DC 57 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL functional confinement design.	N/A

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
Section VI-Fuel and Reactivity Control						
60	Control of Releases of Radioactive Materials to the Environment	The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 60 is required by both 10 CFR 835 and DOE O 458.1 which require a radiation protection program that minimizes radioactivity release to the environment and controls worker exposures. 40 CFR 61 requirements on emissions from NESHAPS ensures normal emissions to the public are well within limits.	MARVEL systems have been designed to ensure appropriate protection to public and the workers with provisions to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated events. Implementation of this PDC is discussed in further detail in Addendum Chapter 11, Radioactive Waste management. The MARVEL reactor is a NaK cooled reactor. During routine operations of MARVEL, the NaK will become activated through neutron interactions. The NaK may also become contaminated with fission products and fissile material if there is a fuel rod leak or failure. Emissions from NaK system leaks are not postulated during normal operations. The MARVEL reactor has an inert cover gas in the reactor barrel to prevent NaK oxidation. The cover gas will activate through neutron interactions and potentially contain fission products and small amounts may leak out of the barrel. The air in the pit will also be activated through neutron interactions. All of the gaseous radioactive waste produced by the MARVEL reactor will be removed from the TREAT building though T-REXC SSCs with any effluent discharged to the environment. A radiation detection system will monitor the radioactivity of the gas exhausted from the MARVEL reactor area ventilation system. Consistent with the MARVEL Environment Assessment, radioactive material releases from MARVEL effluent releases are anticipated to be small. As such, MARVEL does not need to hold up gaseous effluents.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
61	Fuel Storage and Handling and Radioactivity Control	The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage cooling under accident conditions.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 61 is required by DOE O 458.1 (Section 4.h). Operational risks are addressed through implementation of the 10 CFR 835 compliant Radiation Protection Program. Accident and abnormal risks are addressed through existing safety analysis and emergency management requirements.	MARVEL fuel handling systems will be designed to ensure appropriate protection to public and the workers with provisions to ensure criticality safety, adequate radiation shielding and retention of radioactive materials within approved confinement boundaries, and to perform these functions under normal and postulated accident conditions. MARVEL fuel handling systems will include provisions in the design for providing assurance of the ability of these systems to perform their functions through appropriate inspections and tests. MARVEL fuel handling systems will be designed to ensure that appropriate radiation protection exists to ensure that the requirements of 10 CFR 835 Subpart K are satisfied. MARVEL fuel handling systems will be designed in such a fashion that it complies with appropriate requirements from various sources associated with ensuring appropriate radiological material confinement under both normal and postulated accident conditions to comply with radiation protection standards, environmental monitoring and permitting standards and nuclear safety accident requirements. MARVEL fuel handling systems will include passive decay heat removal capabilities to ensure reliability under normal operating and upset conditions. There is no fresh or irradiated fuel storage planned for MARVEL outside of the reactor barrel.
62	Prevention of Criticality in Fuel Storage and Handling	Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 62 is required by 10 CFR 830 and DOE O 420.1 (Attachment 2 Chapter 3) which require a criticality safety program that meets the requirements of this GDC.	The requirements of DOE O 420.1C and associated American Nuclear Society consensus standards will be complied with in ensuring that fissionable material handling processes ensure prevention of inadvertent criticality. Prevention of inadvertent criticality outside of the reactor is discussed in Addendum Section 9.4. MARVEL fuel elements will be stored at the TREAT facility in shipping/storage packages with assigned CSI values. There will be a cumulative criticality safety index (CSI) limit of 50 for the TREAT facility. The handling of MARVEL fuel elements will include removal from shipping/storage packages, configured into sub-assemblies, and loading of sub-assemblies into the reactor. The criticality analysis, Criticality Safety Evaluation for the MARVEL Reactor Fuel Storage and Handling, does not include reactor loading operations. When dry, it would require more than 50 MARVEL fuel elements to go critical. This is substantially more fuel elements that what will be shipped to INL. Therefore, in the absence of moderation there are no credible criticality accident scenarios with MARVEL fuel elements. When arranged in an optimally moderated array, 18 MARVEL fuel elements can achieve criticality. Handling in the TREAT facility will be limited 8 MARVEL fuel elements at one time.
63	Monitoring Fuel and Waste Storage	Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 63 is required by DOE O 435.1 which sets requirements on radioactive waste storage.	There is no fuel storage planned for MARVEL outside of the reactor barrel.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
64	Monitoring Radioactivity Releases	Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for primary system sodium and cover gas cleanup and processing, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for primary system NaK and cover gas cleanup and processing, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated events, and from postulated accidents.	Criterion 64 is required by 10 CFR 835 and DOE 458.1 which require radiation protection programs that monitor all radioactive releases as well as existing environmental regulations.	Radiation monitoring of effluent paths and radiation containing atmospheres will be provided to ensure the ability to detect and respond to the potential for release of radioactive materials. However, as discussed in PDC-16, the MARVEL does not have a reactor containment structure meeting the atmosphere monitoring requirements in this SFR-DC.
Section VII—Additional SFR-DC						
70	Intermediate Coolant System	If an intermediate cooling system is provided, then the intermediate coolant system shall be designed with sufficient margin to assure that (1) the design conditions of the intermediate coolant boundary are not exceeded during normal operations, including anticipated occupational occurrences, and (2) the integrity of the primary coolant boundary is maintained during postulated accidents.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 70 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	Reactor heat is transferred to the power generation equipment through natural convection of the primary coolant fluid (NaK). The PCS provides the geometric and structural support for the containment and flow of NaK. The SCS contains the IHX component, which transfers heat from the NaK to the secondary cooling fluid that interfaces directly with the Stirling engines. The SCS is capable of removing at least 80 kWth of heat from the reactor barrel during all operating conditions. The IHX provides the boundary between the primary and secondary fluids. The secondary coolant is eGa-In-Sn at atmospheric pressure. The design follows a pipe-in-pipe arrangement, where the eGa-In-Sn in the outer annulus pulls heat from the primary coolant, is driven up by buoyancy forces and sinks through the center downcomer pipe once the heat is removed. The SCS is filled with an inert argon blanket gas as well to prevent reaction of any NaK which might leak from the PCS. To ensure that the SCS can accommodate the NaK leakage from a large LOCA without allowing the reactor core to be uncovered, the pressure of the argon gas in the SCS must be kept within 10 psig of the pressure in the primary coolant system. To allow the SCS pressure to be maintained within this band, an active control system is provided which will monitor the pressures in the primary and SCS gas spaces. This system will adjust SCS pressure as required using an electronically controlled back-pressure regulator and an electronic supply flow controller.
71	Primary Coolant and Cover Gas Purity Control	Systems shall be provided as necessary to maintain the purity of primary coolant sodium and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	Systems shall be provided as necessary to maintain the purity of primary coolant NaK and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations, and (4) air or moisture ingress as a result of a leak of cover gas.	Criterion 71 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	The MARVEL design provides cover gas of desired constituent concentrations to blanket both PCS and SCS coolants. The PCS cover gas minimizes NaK coolant contact and reaction with oxygen and water. Argon cover gas supplied to the reactor systems must be of sufficient purity to avoid corrosion and reaction with NaK. The inert gas system (IGS) is designed to limit the impurity levels in the argon cover gas to those specified below: Water: 6 ppm, Oxygen: 5 ppm, Hydrogen: 2 ppm, Nitrogen: 15 ppm, Carbon: 5 ppm, Krypton: 0.3 ppm, Xenon: 0.02 ppm, and Neon: 6 ppm. To minimize the potential for impurity formation within the primary coolant system that can impact the thermal properties of the coolant, NaK supplied to the primary coolant system will be of oxygen concentration of 1 ppm or less.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
72	Sodium Heating Systems	Heating systems shall be provided for systems and components that are important to safety, and that contain or could be required to contain sodium. These heating systems and their controls shall be appropriately designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing sodium are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of sodium aerosol or vapor could prevent accomplishing a safety function, the temperature control and the relevant corrective measures associated with that line shall be considered important to safety.	The MARVEL design employs NaK as the primary coolant and therefore heating systems are not required per SFR-DC 72. Therefore, SFR-DC 72 is not applicable to the MARVEL design.	N/A	Not applicable due to the MARVEL use of NaK as the primary coolant.	N/A
73	NaK Leakage Detection and Reaction Prevention and Mitigation	Means to detect and identify sodium leakage as practical and to limit and control the extent of sodium-air and sodium-concrete reactions and to mitigate the effects of fires resulting from these sodium-air and sodium-concrete reactions shall be provided to ensure that the safety functions of structures, systems, and components important to safety are maintained. Systems from which sodium leakage constitutes a significant safety hazard shall include measures for protection, such as inerted enclosures or guard vessels.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	Means to detect and identify NaK leakage as practical and to limit and control the extent of NaK-air and NaK-concrete reactions and to mitigate the effects of fires resulting from these NaK-air and NaK-concrete reactions shall be provided to ensure that the safety functions of structures, systems, and components important to safety are maintained. Systems from which NaK leakage constitutes a significant safety hazard shall include measures for protection, such as inerted enclosures or guard vessels.	Criterion 73 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	The guard vessel serves as the secondary confinement boundary for the primary coolant and contains the PCS. The pressure difference between the guard vessel and the reactor barrel is the SR SSC required to keep the core from being uncovered in a LOCA event. If there is a leak and the pressures change drastically, the operator will be expected to press the manual scram button. The postulated reason for this requirement is if there is a leak in reactor barrel or guard vessel, but a LOCA has not occurred. This is possible if the guard vessel leaks or there is a small leak in the reactor barrel in the off-gas region. Operations must shutdown the reactor since the pressure will no longer be able to perform its function. Leak detectors are provided to detect NaK within the guard vessel. Several probes located in a cup at the bottom of the guard vessel so that multiple sensors should be flagged as well the NaK should be directed to that catch basin to minimize the required NaK to generate a signal.
74	NaK/Water Reaction Prevention/ Mitigation	Structures, systems, and components containing sodium shall be designed and located to avoid contact between sodium and water and to limit the adverse effects of chemical reactions between sodium and water on the capability of any structure, system, or component to perform any of its intended safety functions. If steam-water is used for energy conversion, to prevent loss of any plant safety function, the sodium-steam generator system shall be designed to detect and contain sodium-water reactions and limit the effects of the energy and reaction products released by such reactions, including mitigation of the effects of any resulting fire involving sodium.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	Structures, systems, and components containing NaK shall be designed and located to avoid contact between NaK and water and to limit the adverse effects of chemical reactions between NaK and water on the capability of any structure, system, or component to perform any of its intended safety functions.	Criterion 74 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	The guard vessel serves as the secondary confinement boundary for the primary coolant and contains the primary coolant system. If the NaK leak detectors indicate that there is a leak, then a leak must have developed between the guard vessel and the reactor barrel. The guard vessel is analyzed and fabricated per ASME Section 3 Division 5, class B vessel code. In the event of a LOCA the guard vessel prevents the core from being uncovered by providing a controlled, inert environment for the coolant to flow into, as well as preventing adverse reactions between pit SSCs (concrete) or air and maintaining the PCS passive decay heat removal capability.
75	Quality of the Intermediate Coolant Boundary	Components that are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 75 is required by DOE O 420.1C (Attachment 3 Section 3.a.(7)) and DOE O 414.1D as a QA programmatic requirement.	It is expected that 316H in the SCS (secondary coolant subsystem) will experience minimal corrosion due to its interaction with eGa-In-Sn. 316H is expected to have essentially an unlimited lifetime in NaK (no corrosion), given the anticipated purity of the given NaK at 1 ppm oxygen level (Vol V Sodium and NaK Handbook). At present it is planned to control the oxygen content in NaK through (i) a barrel vacuum-purge prior to NaK fill and (ii) use of argon cover gas after NaK fill.

Table A-10. Applicability of the RG 1.232 SFR-DC to the MARVEL reactor.

RG 1.232 SFR-DC	RG 1.232 Title	RG 1.232 SFR-DC Criterion Content	Applicability to MARVEL	MARVEL PDC	Applicable DOE Requirements	Implementation in the MARVEL Design
76	Fracture Prevention of the Intermediate Coolant Boundary	The intermediate coolant boundary shall be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 76 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	Consistent with the PCB, the performance of the MARVEL SCS under stressed condition is ensured by selection of materials which behave in a ductile manner over the range of MARVEL operating temperatures. Additionally, design margins for the SCS are such that the boundary is capable of withstanding normal operations and postulated accident conditions in order to ensure that the safety function of keeping the SCS intact is preserved. The SCS is described in detail in Addendum Chapter 5, Reactor Coolant System.
77	Inspection of the Intermediate Coolant Boundary	Components that are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leaktight integrity commensurate with the system's importance to safety, and (2) an appropriate material surveillance program for the intermediate coolant boundary.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 77 is required by DOE O 420.1C Attachment 2 that requires the ability to periodically inspect and test SSCs along with the requirements of a QA program. (Attachment 2 Chapter 1 Section 3.b.4.b and Attachment 3 Section 3.a.(7).	The SCS components are designed for at least 2 years of operation without maintenance. SCS components will be inaccessible throughout the operating life of the reactor.
78	Primary Coolant System Interfaces	When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically incompatible with the primary coolant, the interface location shall be designed to ensure that the primary coolant is separated from the chemically incompatible fluid by two redundant, passive barriers. When the primary coolant system interfaces with a structure, system, or component containing fluid that is chemically compatible with the primary coolant, then the interface location may be a single passive barrier provided that the following conditions are met: (1) postulated leakage at the interface location does not result in failure of the intended safety functions of structures, systems or components important to safety or result in exceeding the fuel design limits (2) the fluid contained in the structure, system, or component is maintained at a higher pressure than the primary coolant during normal operation, anticipated operational occurrences, shutdown, and accident conditions.	Adopted as PDC without revision for MARVEL.	Same as SFC-DC.	Criterion 78 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	The PCB provides multiple barriers completely separating the PCB from the TREAT pit and atmosphere. The inert PCS cover gas minimizes NaK coolant contact and reaction with oxygen and water/moisture. The secondary coolant boundary (SCB) provides a single passive physical barrier between primary and secondary coolants, which are chemically compatible with each other. The SCS headspace is swept with an inert blanket gas to limit reaction of any NaK which might leak from the PCS; this swept gas is confined and filtered of any chemical reaction products of NaK with oxygen or moisture. If there is a cover gas leak it will be detected with either smoke detectors from the NaK vapors in the upper confinement ventilation or there will be a loss in pressure from the pressure indicators.
79	Cover Gas Inventory Maintenance	A system to maintain cover gas inventory shall be provided as necessary to ensure that the primary coolant sodium design limits are not exceeded as a result of cover gas loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the primary coolant boundary.	Revised to reflect that the MARVEL design employs NaK as the primary coolant.	A system to maintain cover gas inventory shall be provided as necessary to ensure that the primary coolant NaK design limits are not exceeded as a result of cover gas loss due to leakage from the primary coolant boundary and rupture of small piping or other small components that are part of the primary coolant boundary.	Criterion 79 is required by DOE O 420.1C GDC 1 which requires conservative design margins (Attachment 3 Section 3.a.(1)).	Cover gas storage is required for initial filling and for cover gas loss during the lifespan of the MARVEL reactor. The inert cover gas system is capable of storing 2500 L of argon and provides the method to fill the primary coolant system and secondary confinement structure with cover gas.

Appendix B
MARVEL SSC Classification

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Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

<table>
<tr><th>System</th><th>Subsystem</th><th>Major Components</th><th>MARVEL Safety Designation (Criterion)</th><th>DOE Safety SSC Classification</th><th>Safety Functions</th><th>Functional Requirements</th></tr>
<tr><td rowspan="11">Fuel and Core System (FCS)</td><td rowspan="7">Fuel System (FS)</td><td rowspan="4">Fuel/Cladding/Fuel Assembly</td><td rowspan="4">SR (SR1, 4)</td><td>SS</td><td>Reactivity Control – Passive IRF</td><td>- Provide system performance related to geometric and physics changes in order to provide net negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted and the reactor is brought to new stable state before fuel, cladding, and PCB temperature limits are challenged, or before core damage occurs during anticipated events and postulated accident conditions.</td></tr>
<tr><td>SS</td><td>Heat Removal - Passive heat rejection</td><td>- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and accident conditions.</td></tr>
<tr><td>SS</td><td>Core Flow – Natural circulation and coolable geometry</td><td>- Provide structural, mechanical, and geometric spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible.
- Provide design provisions to ensure cladding failure due to chemical interactions are not credible.
- Provide design provisions to ensure major core flow blockages are not credible.
- Maintain core coolable geometry in a SDC-2 seismic event.</td></tr>
<tr><td>SS</td><td>Confinement of Radioactive and Hazardous Material Release - Fission product barriers including fuel matrix and Cladding</td><td>- Fuel design provides for retention of radionuclides within its matrix.
- Cladding design provides a barrier for gaseous fission products.
- Provide fuel and cladding structure design to remain within temperature limits to maintain core coolable geometry.</td></tr>
<tr><td rowspan="2">Radial Be Core Reflector (metal) Inserts</td><td>SR (SR1, 4)</td><td>SS</td><td>Core Flow – Natural circulation and coolable geometry</td><td>- Provide structural, mechanical, and geometric spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible.
- Design provisions to ensure major core flow blockages are not credible.
- Maintain core coolable geometry in a SDC-2 seismic event.</td></tr>
<tr><td>SR (SR1, 4)</td><td>SS</td><td>Heat Removal - Passive heat rejection</td><td>- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and postulated accident conditions.</td></tr>
<tr><td>Neutron Source</td><td>NSR</td><td>Non-Safety</td><td>N/A</td><td>- N/A</td></tr>
<tr><td rowspan="3">Core Structures System (CSS)</td><td rowspan="3">Lower Grid Plate Structures</td><td>SR (SR1, 4)</td><td>SS</td><td>Core Flow – Natural circulation and coolable geometry</td><td>- Provide structural, mechanical, and geometric spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible.
- Provide design provisions to ensure major core flow blockages are not credible.
- Maintain core coolable geometry in a SDC-2 seismic event.</td></tr>
<tr><td>SR (SR1, 4)</td><td>SS</td><td>Heat Removal - Passive heat rejection</td><td>- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and postulated accident conditions.</td></tr>
<tr><td>SR (SR1)</td><td>SS</td><td>Reactivity Control – Passive IRF</td><td>- Contribute to the net negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted to bring reactor to a new stable state before fuel, cladding, and vessel temperature limits are challenged, or before core damage occurs.</td></tr>
</table>

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Upper Grid Plate Structures	SR (SR1, 4)	SS	Core Flow – Natural circulation and coolable geometry	- Provide structural, mechanical, and geometric spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible. - Provide design provisions to ensure major core flow blockages are not credible. - Maintain core coolable geometry in a SDC-2 seismic event.
			SR (SR1, 4)	SS	Heat Removal - Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and postulated accident conditions.
			SR (SR1)	SS	Reactivity Control – Passive IRF	- Contribute to the net negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted to bring reactor to a new stable state before fuel, cladding, and vessel temperature limits are challenged, or before core damage occurs.
	Stationary Core Reflector System (SCR)	Stationary BeO Core Reflector Plates (Outside Reactor Barrel)	SR (SR1, 4)	SS	Heat Removal - Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and postulated accident conditions.
			SR (SR1)	SS	Reactivity Control – Passive IRF	- Provide system performance related to geometric and physics changes in order to provide net negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted and the reactor is brought to new stable state before fuel, cladding, and vessel temperature limits are challenged, or before core damage occurs during anticipated events and postulated accident conditions.
			SR (SR1)	SS	Reactivity Control – CD Insertion	- Maintain structural performance of CDs, guide structures, and core under operating and transient conditions to ensure unobstructed insertion path and reactor shutdown.
MARVEL Reactor Structure (MRS)	Primary Coolant System (PCS)	NaK	NSR-AR (NSR-AR2)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including NaK	- Provide design provisions to minimize likelihood of containment failure and radionuclide release.
			SR (SR1, 4)	SS	Heat Removal – Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and SBEs.
			SR (SR1, 4)	SS	Core Flow – Natural circulation and coolable geometry	- Provide structural, mechanical, and geometric spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible. - Provide design provisions to ensure cladding failure due to chemical interactions are not credible. - Provide design provisions to ensure major core flow blockages are not credible. - Maintain core coolable geometry in a SDC-2 seismic event.
			SR (SR-1)	SS	Reactivity Control – Passive IRF	- Provide negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted to bring reactor to a new stable state before fuel, cladding, and vessel temperature limits are challenged, or before core damage occurs.
		Reactor Vessel, Upper Vessel Head, Distribution Block, Downcomers, and all PCB Penetrations	SR (SR1, 4)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including PCS SSCs	- Provide confinement barrier to ensure primary NaK and any leaked fission or activation products remain within vessel and oxygen remains outside.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
			SR (SR1, 4)	SS	Core Flow – Natural circulation and coolable geometry	- Provide structural, mechanical, and geometric spacing to ensure natural circulation through fuel assemblies at reactor operating and elevated transient temperatures and to ensure conduction heat transfer to the passive ambient air heat rejection system is possible. - Provide design provisions to ensure major core flow blockages are not credible. - Maintain core coolable geometry in a SDC-2 seismic event.
			SR (SR1, 4)	SS	Heat Removal – Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and SBEs.
			SR (SR-1)	SS	Reactivity Control – Passive IRF	- Provide system performance related to geometric and physics changes in order to provide net negative reactivity insertion as a function of temperature increase such that the any accidental positive reactivity insertion is passively counteracted and the reactor is brought to new stable state before fuel, cladding, and vessel temperature limits are challenged, or before core damage occurs during anticipated events and postulated accident conditions.
		Insulation	SR (SR1, 4)	SS	Core Flow – Natural circulation and coolable geometry	- Maintain natural circulation ability during all normal operations and shutdown conditions and SBEs.
	Primary Coolant Management System (PCMS)	NaK Storage Tank, Piping (Removable)	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Pressure relief valve	SR (SR4)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including PCMS SSCs	- Prevent overpressurization and failure of PCB SSCs during NaK fill.
	Guard Vessel System (GVS)	Guard Vessel	SR (SR1, 4)	SS	Heat Removal – Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and SBEs.
			SR (SR1, 4)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including GVS SSCs	- Prevent the core from being uncovered during a postulated LOCA by controlling the void space inside the guard vessel. - Prevent NaK-air, NaK water, NaK-concrete, and NaK organics interactions. - Provide a confinement barrier to ensure primary NaK and any leaked fission or activation products remain within PCB and oxygen remains outside. - Reduce probability of large NaK leaks due to pipe design under normal and transient operating conditions.
	Upper Confinement Susbsystem (UCS)	Upper Confinement Structure	NSR-AR (NSR-AR2)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including GVS SSCs	- Provide design provisions to minimize likelihood of containment failure and radionuclide release. - Prevent NaK-air, NaK water, NaK-concrete, and NaK organics interactions.
	Inert Gas System (IGS)	IGS up to double isolation valves on patch panel (ASME Section III boundary)	NSR-AR (NSR-AR2)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including IGS SSCs	- Provide design provisions to minimize likelihood of containment failure and radionuclide release.
		Pressure relief valves	SR (SR4)	SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including PCS and GVS SSCs	- Prevent overpressurization of PCS and GVS.
		Remainder of IGS	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Reactor Support Frame (RSF)	Support Frame, including Alumina Ceramic Plate	SR (SR1, 4)	SS	Core Flow – Natural circulation and coolable geometry	- Provide design provisions to ensure major core flow blockages are not credible. - Maintain core coolable geometry in a SDC-2 seismic event.
			SR (SR1, 4)	SS	Heat Removal – Passive heat rejection during seismic event	- Maintain core coolable geometry in a SDC-2 seismic event.
			SR (SR1, 4)	SS	Heat Removal – Passive heat rejection during normal operations and accident conditions	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and SBEs.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	Reflector Support System (RSS)	Reflector Straps, Upper Reflector Support Plate, Lower Reflector Support Plate, Compression Springs	SR (SR1, 4)	SS	Heat Removal – Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and postulated accident conditions.
			SR (SR1)	SS	Reactivity Control – CD Insertion	- Structural performance of CDs, guide structures, and stationary core reflectors under operating and transient conditions to ensure unobstructed insertion path and reactor shutdown.
		Zirc Debris Shield	SR (SR3)	SS	Reactivity Control – CD Insertion	- Structural performance of CDs, guide structures, and stationary core reflectors under operating and transient conditions to ensure unobstructed insertion path and reactor shutdown.
	Secondary Coolant System (SCS)	IHXs	SR (SR1, 4)	SS	Heat Removal – Active and passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and SBEs.
		IHX Liner with Flange	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		eGa-In-Sn	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Secondary Output Structure (SOS)	Mounting Brackets, Vibration Isolators (Frame)	SR (SR1, 4)	SS	Heat Removal – Active and passive heat rejection	- Provide structural support of the Stirling Engines.
		Vibration Isolators	SR (SR1, 4)	SS	Heat Removal – Active and passive heat rejection	- Reduce translation of Stirling Engine vibration to other reactor components reducing their likelihood of failure.
	Secondary Support Structure (SSS)	Steel Frame Above PCS Distribution Block	SR (SR1, 4)	SS	Reactivity Control – CD Insertion	- Prevent failure of the SCS boundary that could prohibit the ability of the SR CDs from performing their intended safety function.
		Guide Pins	SR (SR1, 4)	SS	Reactivity Control – CD Insertion	- Prevent failure of the SCS boundary that could prohibit the ability of the SR CD from performing their intended safety function.
	Secondary Cover Gas System (SCGS)	Exhaust Ductwork, Back Pressure Regulator, HEPA Filter, Bellows	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Actuated Valves	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Secondary Coolant Management System (SCMS)	Purification Skid, Piping (removable), Valves [If Used]	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Vacuum pump [If Used]	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Regulator [If Used]	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Upper Shield System (USS)	Upper shield system	SR (SR1, 4)	SS	Control Direct Radiation Exposure - Shielding	- Ensure that large shielding components are designed such that the design shielding rates are met under normal operations and potential accident conditions.
	Reactor Shielding System (SHLD)	Radial Gamma and Neutron Shields Outside Guard Vessel	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Axial Gamma and Neutron Shields Above Core Reflectors	SR (SR1, 4)	SS	Control Direct Radiation Exposure - Shielding	- Ensure that large shielding components are designed such that the design shielding rates are met under normal operations and potential accident conditions.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
Reactivity Control System (RCS)	Drum Forcing System (DFS)	Control Drum Motors, Motor Controllers, Motor Resolvers, Current Sensors	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Control Drum EM Clutch	SR (SR1)	SS	Reactivity Control – CD Insertion	- Separate shaft from drive (scram) to ensure CD insertion under normal operating and transient conditions to ensure reactor shutdown.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CD Torsion Spring	SR (SR1)	SS	Reactivity Control – CD Insertion	- Provide stored energy for drum shutdown (scram) rotation.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CD Rotary Damper	SR (SR4)	SS	Reactivity Control – CD Insertion	- Reduce impact to hard stops.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CIA Motor, Motor Controller, Motor Resolver, Motor Gear	NSR-AR (NSR-A3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Ball Screws & Nuts	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CIA Electromagnet	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	Drum Structures	Control Drum Cage & Rails, Cage Platforms	SR (SR1)	SS	Reactivity Control – CD Insertion	- Structural support of drum motors, switches.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	Drum Structures System (DSS)					- Ensure CD insertion under normal operating and transient conditions to ensure reactor shutdown.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Control Drum Shaft and Bearings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Connect control drum to drive system.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
						- Ensure CD insertion under normal operating and transient conditions to ensure reactor shutdown.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Control Drum Hard Stops	SR (SR1)	SS	Reactivity Control – CD Insertion	- Limit CD movement to ensure that available excess reactivity insertion does not challenge fuel and temperature limits when inserted instantaneously.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Control Drum (Rotary) Seal & Standoff	SR (SR1)	SS	Reactivity Control – CD Insertion	- Limit leakage of argon (from Guard Vessel)

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
						- Minimize seal friction.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Control Drum Lock	SR (SR1)	SS	Reactivity Control – CD Insertion	- Provide a physical lock of drum in shutdown position.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Axial Expansion Springs	SR (SR1)	SS	Reactivity Control – CD Insertion	- Accommodate axial expansion of drums. Keep individual BeO plates compressed.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Couplings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Accommodate misalignment between drive and drum shaft.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Upper Alignment Bearings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Align control drums, Allow rotary motion, Minimize friction.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Lower Support Bearings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Support control drums, Align control drums, Allow rotary motion, Minimize friction.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CIA Cage Standoffs, Rails, Platforms	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	Drum Neutronics System (DNS)	Poison Plates	SR (SR1)	SS	Reactivity Control – CD Insertion	- Ensure negative reactivity insertion under normal operating and transient conditions to ensure reactor shutdown.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		BeO Plates	SR (SR1)	SS	Reactivity Control – CD Insertion	- Support and position poison to ensure negative reactivity insertion under normal operating and transient conditions to ensure reactor shutdown.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CIA Rod (B_4C) and Drive Shaft	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CIA Gray Rod (Hafnium) including lock nut, and spacer	SR (SR3)	SS	Reactivity Control – Gray Rod Insertion	- Inadvertent removal could exceed allowable excess reactivity assumed in safety analyses.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
						- Provide negative reactivity for excess reactivity management installed at the beginning of life and stays installed stationary in the reactor throughout the reactor life.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	Drum Position Measurement System	Control Drum Position Indicator & Gear	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	(DPMS)	Control Drum In Limit Switch, Out Limit Switch	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		CIA Position Indicator, In Limit Switch, Out Limit Switch	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
Instrumentation and Control System (ICS)	Interlocks	Control Drum and CIA Motor Relays	SR (SR1, 3)	SS	Reactivity Control – CD/CIA Insertion	- Prevent simultaneous uncontrolled withdrawal of more than one CD as a result of equipment or operator error. - Prevent uncontrolled withdrawal of the CIA rod as a result of equipment or operator error.
	Human Machine Interface (HMI)	HMI Screen	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Analog Pressure Indication	SR (SR1)	SS	Confinement – PCB pressure indication	- Provide reactor pressure indication under normal operating and transient conditions.
		LED Lights	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Control System	I/O Modules	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Chassis	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Computer	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		UPSs	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		DC Power Supply Unit	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Reactor Protection System (RPS)	Scram Button	SR (SR1)	SS	Reactivity Control – CD/CIA Insertion	- Shut down the reactor and maintain it in a safe shutdown condition by manual operator scram.
		DC Power Supply Unit	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Key Switch	NSR-AR (NSR-AR3	Non-Safety	N/A	- N/A
		Seismic Sensor	SR (SR1)	SS	Reactivity Control – CD/CIA Insertion	- Sense a seismic event and provide RPS actuation signal to shutdown reactor by insertion of the CDs.
		Scram Circuit (breakers, relays, latch coils)	SR (SR1)	SS	Reactivity Control – CD/CIA Insertion	- Receive input signal and initiate a reactor shutdown by insertion of the CDs. - Upon loss of off-site power (LOOP), initiate a reactor shutdown by insertion of the CDs.
	Reactor Instrumentation System (RIS)	Neutron detectors and supporting equipment	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Thermocouples	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Leak Detectors	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
		Pressure Sensors	SR (SR3)	SS	Confinement – PCB pressure indication	- Sense pressure differential between primary and guard vessel.

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
Power Generation	Electrical Production	QB80 Engine	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
System (PGS)	System (EPS)	Water Line Connection and Pipes	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Qenergy Engine Control Units (ECUs)	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Qenergy Computer/HMI	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	Engine Cooling	Stirling Engine Automatic Stop System	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
	System (ECS)	Compact Heat Exchangers	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Water Piping/tubing	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Glycol Piping/tubing	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Heat Rejection Units (HRUs)	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Flow/Temp Sensor	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Resistance Temperature Detector, Flow Meter	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Pumps	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Fill Tanks (water and glycol)	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		HRU fan	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Check Valve	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Engine Stall Circuit	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Pressure relief valve	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Humidity sensor	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Accelerometer	NSR	Non-Safety	N/A	- N/A

Table B-11. MARVEL SSC classification summary (Ref. ECAR-6440).

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
		Drain	NSR	Non-Safety	N/A	- N/A

Appendix C
T-REXC SSC Classification

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Table C-12. T-REXC SSC classification summary (Ref. SDS-119).

Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Function(s)	Functional Requirement(s)
Fuel System	Be oxide reflectors	SR (SR1)	SS	Reactivity control – passive IRF	- Provide sufficient negative reactivity to shut down the reactor, and maintain it in a safe shutdown condition, following a normal shutdown, manual scram, or RPS seismic trip.
		SR	SS	Heat removal – passive heat rejection	- Support heat removal from the core under normal and shutdown conditions. - Provide for passive heat rejection to the ultimate heat sink during all normal operations, shutdown, and accident conditions by preserving natural conduction, radiation, and convection capabilities in the reactor system.
		SR	SS	Reactivity control – CD insertion	- Maintain structural performance of CDs, guide structures, and core under operating and transient conditions to ensure unobstructed insertion path and reactor shutdown. - Capable of withstanding material stresses (e.g., creep, swelling) imposed by the operating environment and thermal cycles of the reactor and maintain its capability of rotating into a shutdown position when a SCRAM is initiated. - Designed to the seismic criteria of IBC-2015 using the response coefficients in Table 3-1 of DOE-STD-1020-2016.
		NSR-AR (NSR-AR2)	SS	Control direct radiation exposure – shielding	- Ensure that large shielding components are designed such that the design shielding rates are met under normal operations and potential accident conditions, which might otherwise result in significant loss of shielding.
Confinement	Upper confinement ventilation	NSR-AR (NSR-AR2)	SS	Confinement of radioactive and hazardous material release – fission product barrier	- Provide design provisions to minimize likelihood of confinement failure and radionuclide release.
	Decay heat removal ventilation	SR (SR1, 4)	SS	Heat removal – passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and SBEs.
Shielding	Radial gamma and neutron shields outside guard vessel	NSR-AR (NSR-AR2)	SS	Control direct radiation exposure – shielding	- Personnel radiation protection and protection of non-SR SSCs. - Line the pit walls and protect the pit concrete from radiation-induced degradation.
	Axial gamma and neutron shields above core reflectors	SR (SR4)	SS	Control direct radiation exposure – shielding	- Ensure that large shielding components are designed such that the design shielding rates are met under normal operations and potential accident conditions, which might otherwise result in significant loss of shielding.
Electrical	Electrical panels	NSR	Non-Safety	N/A	- N/A
I&C	I&Cs, including a panel installed into the TREAT Control Room (MFC-724)	NSR	Non-Safety	N/A	- N/A
	Seismic sensors	SR (SR1)	SS		- Sense a seismic event and provide PPS actuation signal to shutdown reactor by passive insertion of control elements.
Compressed Air	Plant air system	NSR	Non-Safety	N/A	- N/A
Radiation Monitoring	CAMs, RAMs	NSR-AR (NSR-AR2)	SS	Control direct radiation exposure – monitoring	- Monitor radiation fields and any released contamination emitted from the microreactor.
Fire Systems	MARVEL NaK fire detection	NSR-AR (NSR-AR2)	SS	Confinement of radioactive and hazardous material release – fire detection	- Detect fires at or near the microreactor and send alarm signals.
	Fire barrier (If installed)	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
Process Gases	Equipment pad and gas manifold outside the building	NSR	Non-Safety	N/A	- N/A
Generator	Standby electrical power system	NSR	Non-Safety	N/A	- N/A
Control Room	Post-accident monitoring	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Instrumentation power	NSR-AR (NSR-AR3)	Non-Safety	N/A	- N/A
	Manual scram	SR	SS	Reactivity control – CD insertion	- Provide manual scram capability to shut down the reactor and maintain it in a safe shutdown condition by manual operator scram.

Appendix D
List of MARVEL 90% Final Design Deliverables

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Appendix D

List of MARVEL 90% Final Design Deliverables

Safety Design Strategy: SDS-119, Revision 1, "*Safety Design Strategy for the Microreactor Applications Research Validation and Evaluation (MARVEL) Project and the TREAT Microreactor Experiment Cell (T-RExC)*"

Hazard Analysis: ECAR-6440, Revision 1, "*MARVEL Hazard Evaluation*"

Requirements:

- FOR-868, Rev. 0, "*Microreactor Applications Research Validation and Evaluation (MARVEL) Project.*"
- TFR-2574, Rev. 0, "*MARVEL Instrumentation and Control System (ICS)*."
- TFR-2575, Rev. 0, "*MARVEL Power Generation System (PGS)*."
- TFR-2576, Rev. 0, "*MARVEL Reactor Structure (MRS).*"
- TFR-2577, Rev. 0, "*MARVEL Fuel and Core System (FCS)*."
- TFR-2578, Rev. 0, "*MARVEL Reactivity Control System (RCS)*."
- FOR-684, Rev. 0, "*Transient Reactor Test (TREAT) Facility Micro-Reactor Experiment Cell (T-REXC*)."

Code of Record: COR-0011, Rev 1, "*MARVEL Code of Record*"

Drawings: See Drawing Tree in Figure

Specifications:

- SPC-3184, Rev 3, "*MARVEL Primary Coolant System and Reflector Support Structure Build to Print Fabrication Specification*"
- SPC-3185, Rev 4, "*Guard Vessel Build to Print Fabrication Specification*"
- SPC-3186, Rev 6, "*Reactor Support Frame, Outer Shield, and Secondary Support Structure Build to Print Fabrication Specification*"
- SPC-30690, Rev 0, "*MARVEL Reactor B_4C Poison Plates*"
- SPC-70525, Rev 0, "*Beryllium Metal Procurement Specification*"
- SPC-3178, Rev 5, "*Beryllium Oxide Procurement Specification*"
- SPC-2999, Rev 2, "*MARVEL Reactor Fuel Element Specification*"
- SPC-70734, Rev 0, "*NaK and Fill System*"

- SPC-70773, Rev 0, "*MARVEL Central Insurance Absorber Gray Rod Build to Print Specification*"
- SPC-70774, Rev 0, "*MARVEL Black Rod Boron Carbide Pellet Build to Print Fabrication Specification*"
- SPC-70692, Rev 0, "*MARVEL Boron Carbide Powder*"
- SPC-TBD, Rev X, *"MARVEL Bearings for Use in Control Drums and Control Drum Actuator Assembly"*
- SPC-70208, Rev 1, *"PCAT Software Requirements"*
- SPC-70388, Rev 1, "*PCAT Calibration Requirements*"
- SPC-2918, Rev 0, *"MARVEL Test Loop Flowmeters"*
- SPC-70469, Rev 1, *"MARVEL Primary Coolant Apparatus Test (PCAT) Subcontractor Design Requirements"*
- SPC-70731, Rev 0, *"MARVEL Reactor Project ASME BPVC Section III Division 5 Design Specification"*

Commercial Grade Dedication Plans:

- CGI-1238, Rev 1, *"Stainless Steel: 316/316L, 316H, and Nitronic 60 for Use in MARVEL*"
- CGI-1224, Rev 5, *"Metallic and Oxide Beryllium Commercial Grade Dedication Plan"*
- CGI-1272, Rev 0, *"MARVEL Vibration Isolators"*
- CGI-1295, Rev 0, *"MARVEL NaK"*
- CGI-1285, Rev 0, *"Zircaloy-4 Debris Shields for Use in MARVEL"*
- CGI-1288, Rev 0, "*Central Insurance Absorber (Hafnium) Rod for Use in MARVEL*"
- CGI-1229, Rev 0, "*MARVEL Control Drum Actuator Clutch and Spring*"
- CGI-1223, Rev 0, "*Seismic Sensor*"
- CGI-1298, Rev 0, *"Axial Compression Springs for Use in MARVEL"*
- CGI-1308, Rev 0, *"Control Drum Actuator Telescoping Coupling"*
- CGI-1297, Rev 0, "*Flanged Sleeve Bearing, Upper Alignment Bearing, and Lower Support Bearing for Use in MARVEL*"
- CGI-1299, Rev 0, *"Axial Gamma Shielding Blocks for Use in MARVEL"*
- CGI-1305, Rev 0, *"ABAQUS v 2021hf6"*
- CGI-1310, Rev 0, *"MCNP v6.2"*

- CGI-1306, Rev 0, *"MOOSE Build 08.01.2023"*

- CGI-1309, Rev 0, *"RELAP5-3D v 4.4.2"*

- CGI-1307, Rev 0, *"STAR CCM+ v. 2210.0001"*

- CGI-1233, Rev 1, *"MARVEL PCAT Test Control"*

- CGI-1232, Rev 0, *"ASME Section III, Division 5 Design Tool Software"*

- CGI-1291, Rev 0, *"BNi-5 Braze Powder for Nuclear Pressure Boundaries"*

- CGI-1287, Rev 0, "*Procurement of Engineering Services including Hazard Analysis and Shielding Design for MARVEL and T-REXC within the TREAT Reactor Building (MFC-720)"*

- CGI-TBD, Rev 0, *"Insulation for Use in MARVEL"*

Engineering Calculation and Analysis Reports:

- ECAR-6585, Rev.0, *"Core Support Structure Analysis."*

- ECAR-6594, Rev.0, CFD Analysis of the PCS

- ECAR-6564, Rev.0, PCS Pressure Vessel Stress Documentation

- ECAR-6580, Rev.0, ASME Section III, Division 5 Analysis of PCS

- ECAR-6502, Rev.0, External Pressure Charts for 316 SS in the Creep Range

- ECAR-6574, Rev.0, Guard Vessel System FEA and ASME Analysis

- ECAR-6586, Rev.0, INERT GAS SYSTEM (NaK Cover Gas Pressure Calcs & Dissolved Oxygen Control In GaInSn

- ECAR-6584, Rev.0, MARVEL Support Frame Analysis

- ECAR-6589, Rev.0, Reflector Support Structure Analysis

- ECAR-6576 Rev.0, Secondary Support Structure Analysis

- ECAR-6583, Rev.0, Outer Shield Stress and Performance Analysis

- ECAR-6588, Rev.0, Chemical Compatibility of MARVEL Components

- ECAR-7210, Rev.0, Steady State Thermomechanical Analysis of Marvel Fuel Bundle Reactor

- ECAR-6601, Rev.0, MARVEL Project Seismic Accelerations

- ECAR-6573, Rev.0, PGS Stirling Engine Radiation Lifetime Estimate

- ECAR-5127, Rev.0, Evaluation of Marvel Reactor Inhalation Dose Consequences

- ECAR-6150, Rev.0, Crit Safety Evaluation for Marvel Reactor Fuel Storage and Handling

- ECAR-6076, MARVEL Reactor End of Life Enveloping Radiological Source Term
- ECAR 7447, Rev.0, MARVEL Upper Shield and Integrated Doses
- ECAR 7448, Rev.0, MARVEL Post Shutdown Dose Analysis in the Upper Confinement
- ECAR-7228, Rev.0, MARVEL Control Drum Actuator Stress Analysis
- ECAR-6099, Rev.0, Neutronics Analyses for the MARVEL Preliminary Documented Safety Analysis
- ECAR-6581, Rev.0, Hydraulic Assessment of Ring-type Spacer
- ECAR-6332, Rev.0, RELAP5-3D Thermal-Hydraulic Analysis
- ECAR-XXX, Rev.0, Fuel-to-Fuel Gap Tolerance Analysis
- ECAR-6598, Rev.0, Calorimetry Calc
- ECAR 6654, Rev.0, Verification and Validation of RELAP5-3D for the design and safety analysis of MARVEL microreactor
- ECAR 6126, Rev.0, Gallium Corrosion
- ECAR-7300, Rev.0, MCNP V&V
- INL/RPT-22-68555 Rev. 0, MARVEL Reactor Fuel Performance Report
- PLN-6874, Rev.0, MARVEL Reactivity Control System Assembly and Checkout (Phase I) and Functional Testing (Phase II)
- PLN-6772, Rev.0, MARVEL Engine Prototype Test Plan
- ECAR-7428, Rev.0, MARVEL Upper Confinement Structure Airflow Analysis

Risk & Opportunity Matrix: See Table 1

Current Cost Estimate: 1G93-D, Revision 5, MARVEL Demonstration Project

Current Construction Schedule: The construction schedule will be appended to the final design report after it is updated following the final design review.

Project Execution Plan: PLN-6384 Rev 1, MARVEL Project Execution Plan

Security & Vulnerability Assessment: Recorded in CCN 254268

Software Quality Assurance Plan: PLN-6908

Test Plans:

- PLN-6874: MARVEL Reactivity Control System Assembly and Checkout (Phase I) and Functional Testing (Phase II)

- PLN-6772: MARVEL Engine Prototype Test Plan

- PLN-6573: MARVEL PCAT Test Plan

Engineering Change Forms:

- EC-1755 R0 TREAT MARVEL REACTOR STRUCTURE (MRS) SYSTEM
- EC-1756 R2 TREAT MARVEL REACTIVITY CONTROL SYSTEM (RCS)
- EC-1757 R0 TREAT MARVEL POWER GENERATING SYSTEM (PGS)
- EC-1758 R0 TREAT MARVEL INSTRUMENTATION AND CONTROLS SYSTEM (ICS)
- EC-1759 R2 TREAT MARVEL FUEL AND CORE SYSTEM (FCS)

Final Design Review Comments and Resolutions: TBD

Engineering Verification Matrices: VM-118 MARVEL Design Verification Matrix