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MARVEL Hazard Evaluation ECAR-6440

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Changing the World's Energy Future

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Idaho National Laboratory

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9. Objective / Purpose

This hazard evaluation supports the Microreactor Applications Research, Validation and Evaluation (MARVEL) Project's design effort and the development of the preliminary documented safety analysis (PDSA) (addendum to safety analysis report [SAR]-420, "Transient Reactor Test [TREAT] Facility FSAR").

The hazard evaluation process for the MARVEL project for compliance with the requirements in 10 CFR 830, "Nuclear Safety Management," 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," and supporting documents.

The LMP process is 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.

10. If revision, please state the reason and list sections and/or page being affected.

Entire document changed. Updated Table 11 and Table 12 to clarify MARVEL and DOE SSC classification criteria and results and other project information as needed.

11. Conclusion / Recommendations

This qualitative hazard evaluation 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 this hazard evaluation is 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.

CONTENTS

FIGURES

TABLES

ACRONYMS/ABBREVIATIONS

A	anticipated
ACGIH	American Conference of Governmental Industrial Hygienists
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
APET	accident progression event tree
Ar	argon
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical
BDBA	beyond deign basis accident
Be	beryllium
BeO	beryllium oxide
BEU	beyond extremely unlikely
CD	control drum
CED	committed effective dose
CFR	*Code of Federal Regulations*
CIA	central insurance absorber
CSS	core structures system
CZP	cold zero power
DBA	design basis accident
DBE	design basis event
D&D	decontamination and decommissioning
DFS	drum forcing system
DHR	decay heat removal
DID	defense-in-depth
DNS	drum neutronics system
DOE	Department of Energy
DPMS	drum position measurement system
DSA	documented safety analysis
DSS	drum structures system
e	eutectic
EAB	exclusion area boundary
ECS	engine cooling system
EG	evaluation guideline
EM	electromagnetic
EPS	Electrical Production System
ES	event sequence
ESF	event sequence family
EU	extremely unlikely
F/CS	filtration/cooling system
FCS	Fuel Core System
FMEA	failure modes and effects analysis

FS	fuel system
FSAR	final safety analysis report
FSF	fundamental safety function
Ga	Gallium
GVS	guard vessel system
H	high
HC	hazard category
He	helium
HFE	human failure event
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
IE	initiating event
IGS	inert gas system
IHX	intermediate heat exchanger
In	Indium
INL	Idaho National Laboratory
IRF	inherent reactivity feedback
KRUSTY	Kilopower Reactor Using Stirling TechnologY
kW	kilowatt
kWe	kilowatt-electric
kW_{th}	kilowatt-thermal
L	low
LBE	licensing basis event
LMP	Licensing Modernization Project
LOCA	loss of coolant accident
LOF	loss of flow
LOHS	loss of heat sink
LOOP	loss of offsite power
LOP	loss of power
LPZ	low population zone
LWR	light water reactor
M	moderate
MARVEL	Microreactor Applications Research, Validation and Evaluation Project
MFC	Materials and Fuels Complex
MJ	megajoules
MLD	master logic diagram
MRS	MARVEL reactor structure system
MSR	Molten Salt Reactor
NaK	sodium-potassium alloy

NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPH	natural phenomenon hazard
NRAD	Neutron Radiography Reactor
NRC	Nuclear Regulatory Commission
NSR	nonsafety-related
NSR-AR	nonsafety-related with augmented requirements
NUREG	Nuclear Regulatory Commission Regulation
OBE	operating basis earthquake
OSHA	Occupational Safety and Health Administration
PAC	protective action criteria
PC	performance category
PCB	PCS boundary
PCMS	primary coolant management system
PCS	primary coolant system
PGS	power generation system
Po	polonium
PRA	probabilistic risk assessment
PrHA	process hazards analysis
PRISM	power reactor inherently safe module
RCS	reactivity control system
RG	regulatory guide
RIA	reactivity insertion accident
RPP	Radiation Protection Program
RPS	reactor protection system
RSAC	Radiological Safety Analysis Computer Program
RSF	reactor support frame
RSS	reflector support system
SAR	safety analysis report
SBE	safety basis event
SC	safety-class
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
SHLD	reactor shielding system
Sn	tin
SNAP	systems for nuclear, auxiliary power
SOS	secondary output structure
SR	safety-related
SS	safety-significant
SSC	structures, systems, and components
SSE	safe shutdown earthquake

SSS	secondary support structure
TED	total effective dose
TLV	threshold limiting value
TOP	transient overpower
TREAT	Transient Reactor Test (TREAT) facility
TRIGA	Training, Research, Isotope, General Atomics
TS	technical specifications
U	unlikely
U-ZrH	uranium zirconium hydride
VTR	Versatile Test Reactor

PROJECT ROLES AND RESPONSIBILITIES

Project Role	Name	Organization	Pages Covered (if applicable)
Performer	Doug Gerstner	H374	See DCR 709066
Checker[a]	Troy Reiss	H374	See DCR 709066
Independent Reviewer[b]	Dr Carlo Parisi	C130	See DCR 709066
CUI Reviewer[c]	Troy Reiss	H374	See DCR 709066
Manager[d]	Jason Andrus	U750	See DCR 709066
Requestor[e]	Yasir Arafat	C120	See DCR 709066
Nuclear Safety[f]	Dr Amanda Foley	H374	See DCR 709066
Document Owner	Jim Parry	U023	See DCR 709066

Responsibilities:

a. Confirmation of completeness, mathematical accuracy, and correctness of data and appropriateness of assumptions.

b. Concurrence of method or approach. See definition, LWP-10106.

c. Concurrence with the document's markings in accordance with LWP-11202.

d. Concurrence of procedure compliance. Concurrence with method/approach and conclusion.

e. Authorizes the commencement of work of the engineering deliverable.

f. Concurrence with the document's assumptions and input information. See definition of Acceptance, LWP-10200.

NOTE: *Delete or mark "N/A" for project roles not engaged. Include ALL personnel and their roles listed above in the DCR system. The list of the roles above is not all inclusive. If needed, the list can be extended or reduced.*

1 INTRODUCTION

1.1 Methodology

The hazards evaluation and accident analysis process for the Microreactor Applications Research Validation and Evaluation (MARVEL) project (Figure 1) for compliance with 10 *Code of Federal Regulations* (CFR) 830, "Nuclear Safety Management," Subpart B, "Safety Basis Requirements,"[1] follows the process identified in safety design strategy (SDS)-119, "Safety Design Strategy for the Microreactor Applications Research Validation and Evaluation Project (MARVEL)."[2] As discussed in SDS-119, the documented safety analysis (DSA) for the MARVEL Project is in the form of an addendum to the existing Transient Reactor Test (TREAT) facility final safety analysis report (FSAR) [Safety Analysis Report (SAR)-420].[3]

The following briefly summarizes the major tasks in Figure 1 that are implemented in this document:

- Task 1: The MARVEL safety-in-design summary is provided to support the hazard evaluation and design basis accident (DBA) analysis (Sections 1.2 and 1.3).
- Task 2: 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 (Sections 2.1, 2.2, and 2.3).
- Task 3: Fundamental safety function (FSFs) necessary to keep the IEs identified in Task 2 from progressing to end states that could result in core damage and release of radioactive or hazardous material, are identified (Section 2.4).
- Task 4: 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 (Section 2.5).
- Task 5: 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 (Section 2.6).
- Task 6: 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 (Section 2.7).

The following major tasks in Figure 1, supported by this hazard evaluation, are implemented outside of this document:

- Task 7: For each defined DBA, a deterministic transient safety analysis will be performed to 1) demonstrate compliance with EGs, 2) establish safety margins, and 3) define SSC performance requirements and operational limits. The DBA analysis will be documented in the MARVEL SAR-420 Addendum Chapter 15, Accident Analyses.
- Task 8: Derivation of the MARVEL technical specifications (TS) will be documented in the MARVEL SAR-420 Addendum Chapter 16, Derivation of Technical Specifications.
- Tasks 9, 11, 12: Documentation of the results of the analyses will be found in the MARVEL SAR-420 Addendum submitted for approval by DOE, and development of the MARVEL TS document.
- Task 10: The process in Figure 1 is iterative and will be repeated as necessary.

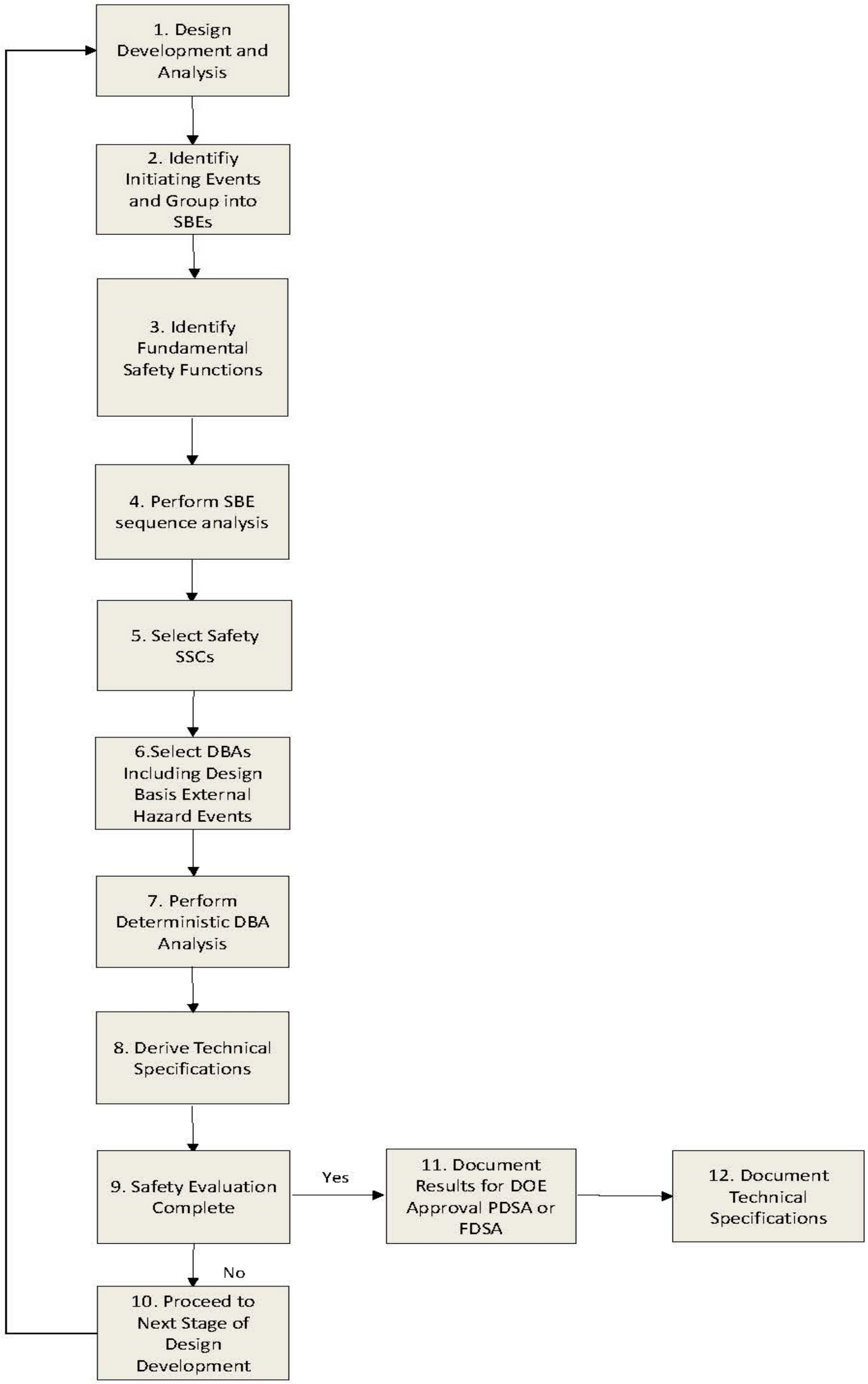


Figure 1. MARVEL hazards evaluation and accident analysis general process flow.

1.2 MARVEL Safety-In-Design Summary

MARVEL takes advantage of well-established uranium zirconium hydride (U-ZrH) fuel, sodium-potassium eutectic (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 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 control drums (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 central insurance absorber (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 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 barrier. 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 primary coolant system (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 guard vessel (fifth barrier) and any associated piping, including the downcomers, ensures primary NaK and any leaked fission or activation products remain within PCB and oxygen remains outside. Altogether, these barriers provide defense-in-depth (DID) to the release of radionuclides to the environment.

The MARVEL design is capable of accommodating various DBA and beyond DBA (BDBA) basis accident initiators 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 feedback 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 CD actuation) passive design requirements will be imposed on the design in the MARVEL TSs.

Refer to INL/RPT-23-74280, “MARVEL 90% Final Design Report,”[4] for a detailed description of MARVEL SSCs.

2 HAZARD EVALUATION

The approach (Figure 1) in the MARVEL hazard evaluation is a qualitative process that ensures a wide variety of possible challenges are considered, while ultimately focusing the analysis on the events of highest importance. The MARVEL hazard evaluation consists of the following tasks as outlined in the following sections:

- Hazard Identification (Section 2.1)
- Hazard Categorization (Section 2.2)
- Initiating Event Analysis (Section 2.3)
- Identification of Fundamental Safety Functions (Section 2.4)
- Event Sequence Analysis (Section 2.5)
- Selection of Safety SSCs (Section 2.6)
- Selection of Design Basis Accidents (Section 2.7).

2.1 Hazard Identification

Hazards that are normally associated with a small reactor facility can result from postulated failure conditions in one or more of the reactor systems or from operational errors. The principal safety functions to protect against potential hazards are adequate cooling, reactivity control, and continued integrity of radioactive material confinement boundaries. All three may be related to a degree, depending upon the details of a given accident.

Hazards to workers include exposure to direct radiation or airborne radioactive material. SSCs serving a safety function in protecting the facility worker from radiological hazards include confinement, shielding and monitoring systems. In addition to nuclear hazards, the possibility of NaK chemical reactions or fires, and exposure to liquid metal secondary coolants [e.g., or eutectic (e) Gallium (Ga)-Indium (In)-Tin (Sn)] also exists. Hazardous materials (radiological and chemical) shall be minimized to those necessary to accomplish the mission.

The radionuclide inventory for the MARVEL core for use in this analysis is found in ECAR-6076, “MARVEL Reactor End of Life Enveloping Radiological Source Term,”[5] and is a reactor run calculated for seven million megajoules (MJ), which is two years of 24/7 operation at ~111 kW_{th}.

The reactor will contain approximately 120 kg of NaK. The primary hazards associated with NaK are fires, explosions, and release of caustic fumes. When exposed to water, NaK reacts violently, producing fire, small explosions, release of caustic fumes, and spattering of hot, reactive particles of NaK and combustion compounds. Argon will be used as the inert cover gas. Neutron reflector material will consist of beryllium oxide (BeO) and beryllium (Be) metal. Molten liquid metal secondary coolants (e.g., eGa-In-Sn) will be used as a secondary coolant. The non-routine material and energy hazard sources that have the potential to result in an uncontrolled release of radioactive and/or hazardous materials or other effects due to MARVEL operations are summarized in Table 1. These non-routine material and energy hazard sources could affect the offsite public, workers, or environment.

Table 2 identifies standard industrial hazards that may be present for MARVEL. Standard industrial hazards are hazards that are routinely encountered in general industry and construction; for these, national consensus codes and/or standards, such as Occupational Safety and Health Administration (OSHA) standards, exist to guide safe design and operation. No special analysis is required for these occupational hazards unless they are possible initiators for an uncontrolled release of radioactive or hazardous material. This hazard analysis includes events associated with initiators of this type.

Direct radiation hazards associated with planned work and operational activities are managed through the INL Radiation Protection Program (RPP), which includes training and analysis of all radiation work to ensure worker protection per 10 CFR 835, "Occupational Radiation Protection,"[6] regulations. Table 3 identifies standard radiation hazards associated with MARVEL and its operations and RPP program features that prevent or protect against them.

Table 1. Summary of non-routine material (1) and energy hazard (2) sources.

Hazard	Hazard Source(s)	Concern
Fissionable materials (1)	Fissionable materials	Potential for inadvertent nuclear criticality
Hazardous materials (1)	Hazardous material (e.g., NaK, eGa-In-Sn, Be)	Potential for hazardous material exposure and release
Radioactive materials (1)	Radioactive materials (core fuel, Ar-41, tritium, neutron source)	Potential for radioactive material release or direct radiation
Electrical energy (2)	Electrical equipment (PGS, plant electrical systems)	Potential initiator of a fire causing a release of radioactive or hazardous material
Fire, explosion, flammable materials (thermal chemical energy) (2)	Flammable materials and ignition sources in facility; range fires; and transient combustible materials	Potential for fire or explosion causing building damage and a release of radioactive or hazardous material
Kinetic energy (2)	Rotational energy from motors, moving equipment, vehicle impact	Potential to cause a loss of confinement resulting in a material release causing a release of radioactive or hazardous material
Potential energy (2)	Suspended loads, Heavy Load Drops	Potential for impact damage causing release of radioactive or hazardous material
Pressure (2)	Compressed gasses (Helium (He), Argon (Ar)), pressurized systems (reactor barrel, intermediate heat exchanger (IHX), He within Stirling Engines)	Potential to cause a loss of material boundaries causing a release of radioactive or hazardous material
Natural phenomena (2)	Earthquake, severe weather (e.g., wind, flood, lightning, etc.)	Potential initiator of a radioactive or hazardous material release

Table 2. Standard industrial hazards regulated by DOE-prescribed OSH standards.

Hazard	Applicable to Facility (Yes/No)	DOE-Prescribed Program and OSH Standards
High voltage (≥600 V)	Yes	29 CFR 1910 Subpart S; National Electric Code [National Fire Protection Association (NFPA) 70]
Low voltage (<600 V)	Yes	29 CFR 1910 Subpart S; National Electric Code (NFPA 70)
Volatile flammable or reactive gases or liquids (NaK fire/explosion hazard)	Yes	29 CFR 1910 Subpart H, 0.144, 0.1200; 29 CFR 1926.152
Explosive materials (NaK fire/explosion hazard)	Yes	29 CFR 1910.109; DOE Explosives Safety Manual (DOE Manual 440.1-1)
Cryogenic systems	No	None of the DOE-prescribed standards clearly address cryogenics
High temperature (≥125°F at contact or 203°F)	Yes	American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, ANSI/ASME Standard B31
High pressure (≥15 psig for gas or vapor or ≥200 psig for liquids)	Yes	ASME Boiler and Pressure Vessel Code, ANSI/ASME Standard B31
Low pressure	Yes	ASME Boiler and Pressure Vessel Code, ANSI/ASME Standard B31
Inert and low-oxygen atmospheres (Confined spaces in Pit and MARVEL upper confinement)	Yes	29 CFR 1910.119, .120, .1200; 29 CFR 1926.651
Toxic materials	Yes	29 CFR 1910.119, .120, .1200, Subpart Z; 29 CFR 1926.353; ACGIH TLVs
Nonionizing radiation	Yes	29 CFR 1910.97; ACGIH TLVs
High intensity magnetic fields	No	ACGIH TLVs
High noise levels	Yes	29 CFR 1910.95, .1200; 29 CFR 1926.52; ACGIH TLVs
Mechanical and moving equipment dangers	Yes	29 CFR 1910.147, .211 through 222; 29 CFR 1910 Subparts O, P, Q; 29 CFR 1926 Subpart W
Working at heights	Yes	29 CFR 1910.25, .28; 29 CFR 1926.951, .451
Excavation	No	29 CFR 1926 Subpart P
Material handling dangers	Yes	29 CFR 1910.120, .176 through .182; 29 CFR 1926.953; DOE-STD-1090-2007 Hoisting and Rigging
Material transportation	No	Hazardous Material Transportation Program, DOE O 460.1B and 460.2A
Pesticide use	No	29 CFR 1910.1200
Temperature extremes (high and low temperatures during activities)	No	29 CFR 1910.120, .1200; ACGIH TLVs
Inadequate illumination	No	29 CFR 1910.37, .68, .120, .177 through .179, .219, .303; 29 CFR 1926.26
Construction	Yes	29 CFR 1926

Table 2. Standard industrial hazards regulated by DOE-prescribed OSH standards.

Hazard	Applicable to Facility (Yes/No)	DOE-Prescribed Program and OSH Standards
Ionizing radiation	Yes	Radiation Protection Program, 10 CFR 835
Reactive materials: alkali metal and corrosives	Yes	10 CFR 851
Structural or natural phenomena	Yes	DOE O 420.1C, DOE G 420.1-2, 29 CFR 1910.119 Subpart E
Fire	Yes	Fire Protection Program, DOE O 420.1C
Biological agents	No	None of the DOE-prescribed standards clearly address biological agents
Other	No	29 CFR 1903.1 (General Duty Clause)

Table 3. Radiological hazards regulated and mitigated by 10 CFR 835.

Facility Specific Hazards/Issue	Mitigating Program Features
Direct radiation – Entry to areas with potentially high radiation	The use of written procedures, radiological work permits, in-process surveys, supplemental dosimetry, radiological postings, access controls.
Direct radiation – Degradation of shielding	The use of written procedures, radiological work permits, in-process surveys, supplemental dosimetry, radiological postings, access controls, design reviews, area monitoring.
Direct radiation – Mishandling of reactor and components	The use of written procedures, radiological work permits, approved procedures, in-process surveys, supplemental dosimetry, area monitoring.
Direct radiation – Contamination of equipment (e.g., primary or other piping)	The use of written procedures, radiological work permits, in-process surveys, facility routine surveys, supplemental dosimetry, housekeeping, contamination control.

2.2 MARVEL Facility Hazard Categorization

10 CFR 830 Subpart B paragraph 202(b)(3) requires that the hazard categorization for a DOE nuclear facility be performed consistent with DOE-STD-1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports,"[7] which has been superseded with DOE-STD-1027-2018, "Hazard Categorization of DOE Nuclear Facilities."[8] DOE-STD-1027-2018 is an update of DOE-STD-1027-92. Both are still applicable and consistent for MARVEL.

DOE-STD-1027-2018, deemed an appropriate mechanism for meeting and implementing the requirements of DOE-STD-1027-92, identifies that reactors with steady-state powers 20 MWth and greater are considered Category A reactors and that Category B reactors are reactors that are not classified as Category A reactors. Category B reactors are considered to be hazard category (HC)-2 facilities. TREAT is a Category B reactor and is classified as a HC-2 nuclear reactor facility. Given the anticipated thermal power, or power being produced by the core, as <85 kWth (nominal HFP), MARVEL is also a Category B reactor. Consistent with the hazard category interpretations in DOE-STD-1027-2018, given that TREAT and MARVEL are both Category B reactors, the TREAT facility remains overall a HC-2 facility.

As required by DOE-STD-1027-2018, a hazard analysis is performed in this document as part of final hazard categorization to determine the effects of available energy sources and radioactive material release mechanisms. Based on the evaluation in ECAR-5127, "Evaluation of the MARVEL Reactor Inhalation Dose Consequences."[9] and on the criteria in DOE-STD-1020-2016, "Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities,"[10] the MARVEL reactor and support safety systems are categorized as seismic design category (SDC)-2.

2.3 Initiating Event Analysis

A systematic approach was used to identify IEs that challenge MARVEL plant operation and require successful mitigation to prevent radionuclide release. Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,"[11] Nuclear Regulatory Commission Regulation (NUREG)-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,"[12] RG 1.206, Combined License Applications for Nuclear Power Plants (LWR Edition)," Part I: Standard Format and Content of Combined License Applications,"[13] and NUREG-1537, "Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,"[14] were first reviewed for identification of potential IE's applicable to MARVEL. In addition, available design and licensing/safety basis documentation for the following light water reactor (LWR) and non-LWR reactor designs were reviewed for applicability to MARVEL and an assessment of the failure modes and effects of systems that are unique to the MARVEL design:

1. Microreactor designs:
    - Kilopower Reactor Using Stirling TechnologY (KRUSTY)
    - Systems for Nuclear, Auxiliary Power (SNAP) – 10A
    - eVinci
    - OKLO Aurora.
2. Sodium-cooled reactors:
    - Power Reactor Inherently Safe Module (PRISM)
    - Versatile Test Reactor (VTR).
3. Small modular reactors:
    - NuScale.
4. INL small light water test reactors:
    - Advanced Test Reactor Critical (ATRC)
    - Neutron Radiography Reactor (NRAD)
    - Advanced Test Reactor (ATR).
5. Other reactor designs:
    - AP-1000
    - Molten Salt Reactor (MSR) Case Study
    - X-energy Xe-100
    - Terrapower
    - Kairos
    - Megapower
    - High Temperature, Gas-Cooled Pebble Bed Reactor.

In addition to the identification of generic IEs, a detailed, systematic review of important MARVEL systems was performed using a failure modes and effects analysis (FMEA) technique to identify IEs unique to the MARVEL design [see Appendix A].

The IE analysis includes internal events that occur while MARVEL is shut down and while at-power, and external NPH events (e.g., fires, flooding, seismic events) that could challenge MARVEL plant operations and require successful mitigation to prevent radionuclide or hazardous material release. The MARVEL external events analysis includes events from TREAT operations that could impact MARVEL SSCs and result in a release of radioactive or hazardous material.

Human error induced initiating events are also considered. Most human failure events (HFEs) that disrupt normal plant operations will result in a general transient and reactor trip. These events are assumed to be subsumed in the general transient occurrence data. HFEs are considered for the other general categories as well.

For this analysis, the IEs are grouped for similar core response and success criteria into the following major categories:

- Internal Hazard Events (Section 2.3.1)
- External Hazard Events (Section 2.3.2).

2.3.1 Internal Hazard Events. The internal hazard IE group includes equipment- and human-induced events that disrupt normal plant operations. Based on the guidance in NUREG-0800, NUREG 1537, and RG 1.206, MARVEL internal IEs are grouped into the following categories:

1) Shutdowns
2) General Transients
3) Increase in Heat Removal by the Secondary System
4) Decrease in Heat Removal by the Secondary System
5) Decrease in Primary Coolant System Flow Rate
6) Loss of Power
7) Reactivity and Power Distribution Anomalies
8) Core and Local Faults
9) Decrease in Reactor Coolant Inventory
10) Increase in Reactor Coolant Inventory.

2.3.1.1 Shutdowns—Shutdowns and general transients are separated based on the MARVEL response necessary to preserve reactor safety. Shutdowns and general transients present general challenges to normal operation. The shutdown category of IEs includes those events that involve planned or unplanned reactor scrams and shutdowns to ensure reactor safety, but still require DHR once shutdown conditions are achieved. Owing to the low power (and thermal neutron flux) of MARVEL, Xenon effects are negligible and not included in this category. MARVEL specific IEs for this group include:

- Manual shutdowns: Purposeful, controlled descents from critical power, including both planned shutdowns and unplanned shutdowns
- Test scrams: Purposeful reactor scrams initiated to demonstrate RPS actuation or plant response to prescribed conditions

- Spurious reactor trips: Incidental scrams caused by spurious RPS signals
- Unintended rotation inward of the CDs.

Reactor scrams may pose cooldown issues from normal operating temperatures, assuming that the PGS Stirling engines continue to run after scram, resulting in thermal stresses to the PCB SSCs. A specific cooldown rate is required for compliance with the ASME code. An automatic shutdown of the Stirling engines is followed by operations turning the Stirlings back on slowly, in a manner assumed to cooldown the primary coolant below an acceptable ramp to prevent thermal stresses and meet ASME Code limits. Evaluation of the reactor structure may be required prior to the next reactor cycle.

2.3.1.2 General Transients—Transients are the changes of the plant's parameters resulting from anticipated or unanticipated changes in one or more parameters of the plant. Transients may be initiated by changes in the operation, behavior, or performance of equipment, leading to changes in parameters such as reactor power level, coolant temperature or flow rate, generator load or more. General transients include events or component failures that have no impact or an indirect impact on safety systems; however, they require immediate action (typically scram) to prevent further degradation or challenges to plant systems. With these IEs, both reactivity control (scram) and DHR are required. MARVEL specific IEs for this group include:

- Minor core blockages (e.g., flow disruption between neighboring pins) from loose parts or debris from cladding leak or failure due to 1) design error, 2) internal pressure build up (fission gas, hydrogen release), 3) pellet cracking, 4) fuel swelling, 5) weld failure, 6) excessive temperatures, 7) fission product cladding interactions.
- Small reactivity changes (e.g., miss-positioning of a single CD, failure to seat a fuel pin properly).
- Slower reactivity events, such as due to in-cycle reactivity changes from fuel deformation, or more immediate changes such as from thermal expansion-induced rod bowing and the accompanying changes when temperature comes back down.
- Beryllium material or mass loading error or reflector structural failure leads to excess reactivity higher or lower than expected. Reactor unable to achieve criticality, or higher or lower than expected heat output.
- Loss of a support system from internal facility fire or flooding.
- Neutron source failure leads to inability to adequately monitor initial criticality.
- Coolant system minor leaks.
- Excessive Stirling engine vibration.
- Stirling engine depressurization
- Inadvertent actuation of the heaters.

A startup sequence on electric power from CZP condition has been simulated in ECAR-6332, "RELAP5-3D Thermal-Hydraulic Analysis of Marvel Microreactor - Final Design,"[15] Section 4.2.5, assuming that the four 3.525-kW heaters are instantaneously turned on. The startup from CZP state using first 14.1-KWth electric heaters power and then nuclear power shows that all the safety criteria for the fuel are respected. An accident involving the heaters was not performed in ECAR-6332. As such, a control is then placed that the heaters are disabled after startup.

2.3.1.3 Increase in Heat Removal by the Secondary System—The MARVEL does possess a secondary coolant system (SCS) for heat removal, though it is very different from those present for LWRs. The primary effect of an increase in heat removal by the SCS is to remove more heat from the PCS that is being generated by fission, which causes a reduction in fuel temperature and an increase in reactivity Such a reactivity insertion due to overcooling of the MARVEL primary system by the power conversion unit is bounded by the Reactivity and Power Distribution Anomalies group.

2.3.1.4 Decrease in Heat Removal by the Secondary System— Undercooling or decrease in heat removal faults involve the loss of the capability to remove reactor heat. There are two types of undercooling initiators: 1) loss of active heat removal from the reactor core to the ultimate heat sink (through the PGS Stirling engines to ambient air) during normal operations and shutdown but allowing for the capability to passively move decay heat from the reactor core to the ultimate heat sink (conduction and convection through reactor from core to ambient air in the TREAT Pit), and 2) total loss of both active and passive heat removal paths (e.g., seismic event). MARVEL specific IEs for this group include:

- Degradation or loss of PGS Stirling engines (single and cascading), failure of engine control units, heat exchangers or electronics due to vibration, mechanical failure, engine stall, radiation damage, heat damage, physical damage, or material, structural, or seismic failures.
- SCS pipe leaks/breaks due to corrosion caused by exposure to liquid metal secondary coolants (e.g., eGa-In-Sn).
- Stirling engine heat exchanger (Hx) tube failure and flow of gas into SCS.
- Loss of an intermediate heat exchanger (IHX) (tube failure, flow blockage, oxide buildup) and loss of heat transfer through the PGS.
- DHR system blockage from failure of the pit shielding structures, loose parts or debris, facility fires, internal flooding of TREAT pit, or external NPH.
- Blockage of passive heat removal pathway due to failure of shielding in a seismic event.
- Secondary cooling system (SCS) SSC material or structural failure.
- PGS water line connection and pipes failure and leak.
- Heavy load drop over reactor results in damage to DHR SSCs.
- Heavy load drop over reactor results in damage to Stirling engines or SCS SSCs.

2.3.1.5 Decrease in Primary Coolant System Flow Rate—There are no operating pumps or active components in the MARVEL design producing forced flow. Therefore, a decrease in flow only considers faults that cause a reduction in natural circulation through the core. MARVEL specific IEs for this group include:

- Core blockage (partial or total) due to loose parts or debris.
- Core blockage due to distortion, bowing, or bulging of fuel pins or CDs.
- Core blockage from cladding failure due to corrosion of steel, chemical interactions, formation of zirconium oxide, formation of uranium oxide, inferior end of life strength or premature cladding failure (fretting not considered due to low fluid velocities).
- Internal cladding pressure build up (fission gas, hydrogen release), pellet cracking, fuel swelling, weld failure, excessive temperatures, fission product cladding interactions (fretting not considered due to low fluid velocities).
- Failure in a SCS IHX and leakage of secondary coolant into the PCS NaK resulting in flow reduction and loss of natural circulation.

- PCB leaks.
- Fuel/cladding material loading error, design error, or fuel assembly design error or structural failure, or bowing of fuel pins, or mechanical defects (straightness, tolerance) leads to blockage or insufficient heat transfer from fuel to primary coolant and decrease in PCS natural convection flow rate (fretting not considered due to low fluid velocities).
- Fuel assembly structure failure during seismic event.
- Be reflector design error or structural failure leads to insufficient heat transfer within the core and decrease in PCS natural convection flow rate.
- Upper or lower grid plate design error or manufacturing error or structural failure results in failure to maintain the primary coolant pressure drop across the core, fuel rod separation, decrease in PCS natural convection flow rate, and insufficient heat transfer within core.
- Misplacement/movement of stationary reflectors and insufficient heat transfer within core.
- NaK material loading error.
- Fuel system (FS), PCS or reflector or support SSC material or structural failures or seismic event.
- Low PCS pressure, NaK boiling under accident conditions.
- IHX material or design error, structural failure.

2.3.1.6 Loss of Power—MARVEL systems are anticipated, and therefore assumed, to be designed to be fail safe in the event of a loss of power, and it is assumed that sufficient margin will be available in the design to assure safe plant shutdown upon loss of power without needing a separate electrical power source. A loss of power (LOP) event considers interruptions of normal power to the electrical buses, which will result in reactor scram. The following IEs have similar mitigation requirements and were grouped together to form the loss of power IE group:

- TREAT facility related LOP
- Grid-related loss of offsite power (LOOP)
- Switchyard-centered
- Weather-related
- Seismic events
- TREAT facility fire or internal MARVEL system fire.

2.3.1.7 Reactivity and Power Distribution Anomalies—MARVEL does not use control rods for reactivity control, instead incorporating slow-moving CDs for this purpose. Four reactivity insertion IEs in this category are identified:

- Reactivity insertion (small)
- Reactivity insertion (moderate)
- Reactivity insertion (large)
- Reactivity insertion (extreme).

Small reactivity insertions are events defined as a reactivity insertion significant enough that the plant should be shut down, but within the capability of the design to tolerate without fuel damage despite failure to scram. These reactivity insertions are covered under general transients and include mispositioning of a single CD.

Moderate or large reactivity insertions are the result of positive reactivity insertions during operation, which can be introduced by means of:

- Control drum malfunction (medium or large reactivity insertion) (CD gets stuck and fails to insert due to swelling, distortion, bowing, or bulging due to gas buildup).
- CD motor, CD structure, neutronics, or position measurement SSC electronics failures, heat damage, or radiation damage, physical damage, or material, structural, or seismic failures.
- CIA motor, structure, neutronics, or position measurement SSC electronics failures, heat or radiation damage, physical damage, or material, structural, or seismic failures.
- Drum control system, MARVEL control system, RPS (including scram circuit, seismic sensor, neutron detector, temperature/pressure/leak sensors, heaters, and seals) SSC electronics failures, heat damage, or radiation damage, physical damage, or material, structural, or seismic failures.
- Fuel/cladding material loading error, manufacturing error, design error, or fuel assembly design error or structural failure results in less net negative temperature coefficient than expected.
- Fuel pin Uranium mass loading error leads to greater or less excess reactivity or heat generation than expected.
- Cladding leak or failure from corrosion leads to mechanical defects (pin holes). Release of loose material to coolant.
- Seismic event leads to fuel assembly structural failure.
- Upper or lower grid plate manufacturing error or structural failure.
- Misplacement/movement of stationary reflectors.
- NaK voiding (such as gas entrainment).
- Overcooling of the primary system by the PGS.
- Flooding of the core from PGS water line connection or pipe failure and leak during maintenance.
- NaK material loading error, or structural failure.
- CD support SSC material or structural failures.
- Heavy load drop over reactor results in damage to CD or CIA structural SSCs above reactor.
- Heavy load drop over reactor results in core compaction and reactivity insertion.

Conservatively, a CD motor malfunction event involves the rotation speed of the drum at the maximum speed that the motors are capable of rotating, higher than necessary for properly compensating for fuel depletion.

There are physical drum limits or stops that limit excess reactivity. The safety-related (SR) CD stops limit CD movement to ensure that available excess reactivity insertion does not challenge fuel and temperature limits when inserted instantaneously. The safety-related CD relays prevent simultaneous uncontrolled withdrawal of more than one CD as a result of equipment or operator error. Operators will be restricted on approach to criticality to avoid creating a situation where a single drum could have excessive reactivity. The reactor will be started with all 4 drums moved approximately equally to achieve criticality, versus achieving criticality with some drums at the fully inserted position. This limits the excess reactivity from any single drum.

Generally, larger reactivity insertion rates at the highest operating power are the most limiting events in the reactivity anomalies category. The fission heat generation rate is the key driver of the system response and strongly influences the fuel temperature, which is the primary safety metric of interest. The fuel temperature is highest at full operating power. Therefore, low power events, such as an uncontrolled control drum assembly withdrawal from a subcritical or low-power startup condition, are bounded by full power events. Applicable events for at-power uncontrolled reactivity insertions involve malfunctions in the rotation of the CDs, such as from bowing or bulging due to helium gas generation, or operator error. NaK voiding from argon bubble gas or air (from Gap and Plenum Fill Gas) entrainment into the PCS could occur and result in a power surge. The reactivity is increased because NaK is a slight neutron poison. Therefore, when a gas void displaces some NaK, the local neutron absorption goes down and the reactivity goes up as long as the NaK is displaced.

Extreme reactivity insertions are events defined as non-credible, non-mechanistic, reactivity insertions beyond the nominal reactivity worth for withdrawal of the four CDs resulting in a core disruptive event and may involve fuel, coolant, or material relocation.

Since there are no active components or pumps, overcooling from pump overspeed is not considered.

2.3.1.8 Core and Local Faults—Core faults include stochastic fuel cladding failures, and core flow reduction events such as those caused by loose parts, foreign material, and assembly bowing or deformation. Core faults that can lead to core flow reduction are grouped into three categories: minor, moderate, and major blockages, based on the necessary plant response.

Minor core blockage is a blockage of minimal size such that any potential fuel damage is expected to be extremely localized and not a challenge to core safety. IEs in this category include small blockages from loose material or stochastic cladding failure. A local blockage would result in a decrease of mass flow/heat transfer in the affected sub-channel(s) with local fuel overheating and consequent power reduction. the operator could detect by an unplanned reactor power decrease and new core T_{outlet}/T_{inlet} steady state values. If detected, these conditions could warrant a scram or plant shutdown. Therefore, minor core blockages are included under General Transients due to similar plant response.

Moderate core blockages can result in a loss of flow to a small portion of the core, such as a group of fuel assemblies. Major core blockages (e.g., seismic event) include near complete core blockage or substantial flow diversion. Moderate and major core blockages are covered under the Decrease in Primary Coolant System Flow Rate plant response.

The local faults considered in the MARVEL are sub-categorized into: (1) Increased heat generation local faults, and (2) Reduced heat removal local faults. The specific local faults that are described in the increased heat generation sub-category are enrichment error (placing an assembly with a higher enrichment than desired into a wrong loading location, leading to greater heat generation than expected) and oversized fuel. The reduced heat removal local faults include flow blockages, SCS heat exchanger secondary coolant leaks into primary (NaK), as well as fuel element bond defects, and are covered under the Core Faults category above.

The MARVEL core utilizes fuel with the same enrichment in every position: as such, an enrichment error due to fuel misloading are extremely unlikely. Enrichment errors are minimized by the quality assurance program applied to the fuel manufacturing; nonetheless, the response to enrichment errors would be more benign due to the MARVEL low core power densities.

2.1.3.9 Decrease in Reactor Coolant Inventory—Decrease in Reactor Coolant Inventory events could result from PCB penetration leaks/breaks/seal ruptures, cover gas line breaks or valve failures, or opening of drain valves resulting in reduction of coolant inventory and release of radionuclides. A PCB leak considers a breach of the reactor barrel and may result in a loss of primary NaK inventory. Reactor barrel rupture could occur due to cracks at welds rupturing under pressure or load. Seismic events may cause welds to fail, resulting in loss of coolant pathway. Other parts of the system may be more vulnerable to seismic or similar events.

The following scenarios cover the potential leakage paths for primary NaK coolant from the MARVEL reactor system. The PCB will be fully welded below the maximum NaK coolant level. Since there are no mechanical fittings or connections, leakage (except for cover gas scenario) would be due to failure of either welds or base materials. MARVEL specific IE for this group include:

- Undetected weld flaw (either penetrating flaw or defect which weakens the weld).
- Undetected flaw in base material (either penetrating flaw or defect which weakens the material).
- Fatigue crack due to repeating stress (thermal striping also a possible initiator).
- Creep failure.
- PCB penetration leaks/breaks/seal ruptures or support SSC failure.
- PCMS, inert gas system (IGS) NaK system leak or failure.
- Overpressure, overstress or overtemperature of PCB and SCB SSCs.
- Corrosion.
- Heavy load drops over reactor result in impacts to core barrel, PCS piping or GV and leak of primary coolant.
- Heavy load drops results in Stirling engine heat exchanger (HX) failure leading to a high-pressure gas release in the SCS, failure of the SCB to PCB, and over-pressurization and failure of the primary barrel and guard vessel.
- Heavy load drops results in high temperature heat extraction HX failure leading to a high-pressure gas release in the SCS, failure of the SCS boundary to PCB, and over-pressurization and failure of the primary barrel and guard vessel, SCS bellows, and Stirling engine bolts.
- Heavy load drops results in PGS pipe break and pipe whip (pipe break containing 1000 psi helium or PGS 50 bar helium) with a high energy line break damage to the area containing the primary boundary, guard vessel, control drum and control rod drives, as well as the ducting for the ultimate heat sink.
- Operator error introduced by setting PCS and GVS pre-load pressures incorrectly. Operator error introduced by violation of heat-up and cool-rate limits.

Chemical compatibility of reactor components in contact with the primary and secondary coolants has been evaluated for NaK in ECAR-6588, “Chemical Compatibility of MARVEL Components,”[16] and for eGa-In-Sn alloy in ECAR-6126, “Gallium Based Corrosion on Stainless Steel for MARVEL.”[17] Gallium is well known for its corrosion.

The reactor design avoids the core from being uncovered in the above scenarios by a combination of design, monitoring, and administrative controls. The set of monitoring and administrative controls needed to ensure that required reliability will be developed in the MARVEL PDSA.

The potential for leaks of the cooling water system onto the primary containment head may also occur. However, there will not be break/rupture of the PCB because water splashing will not cause a primary stress and it will "self-relax". But if there is a water leak, an inspection will be required.

Potential PCS leak locations (see Figure 2) are therefore as follows:

1. PCS Leak Inside Guard Vessel – This path involves any leakage from the reactor barrel, distribution plenum or PCS piping which occurs within the Guard Vessel. Potential leakage volume would be limited by the design of the Guard Vessel, which encompasses the sides and bottom portions of the PCS, the elevation of the leakage opening, and the initial gas pressures in the primary vessel and guard vessel. The core will remain covered.

2. PCS-IHX Leak – This path involves a failure of the IHX wall between the NaK primary coolant and the secondary coolant due to 1) corrosion, or 2) Stirling engine Hx tube high energy break and impingement of high-velocity He gas on and rupture of the IHX/PCS boundary. The impingement has been determined to be of insufficient force to result in failure of both the liner and IHX wall and leakage of NaK into the SCS. Therefore, the leak is assumed to occur due to NaK corrosion of the IHX wall at worst location at the lowest point in the IHX, resulting in NaK leakage into the space between the IHX wall and IHX liner. Leakage into the upper confinement is prevented by the IHX liner flange. The core will remain covered.

3. PCS-Upper Confinement (UC) Leak – This path involves leakage via the top plate of the Distribution Plenum (see Figure 3) which also extends radially beyond the PCS to form the top of the guard vessel), the Closure Head (Figure 4), heater tubes [not shown], or PCS-CIA rod. Leakage from the CIA Rod and heaters are prevented by the double wall design. Although unlikely, leakage into the upper confinement could occur through weld failures in top plate. Potential leakage volume is limited to NaK vapors and/or droplets. This could lead to a NaK-air interaction, smoke, and release through the UC ventilation system. The core will remain covered.

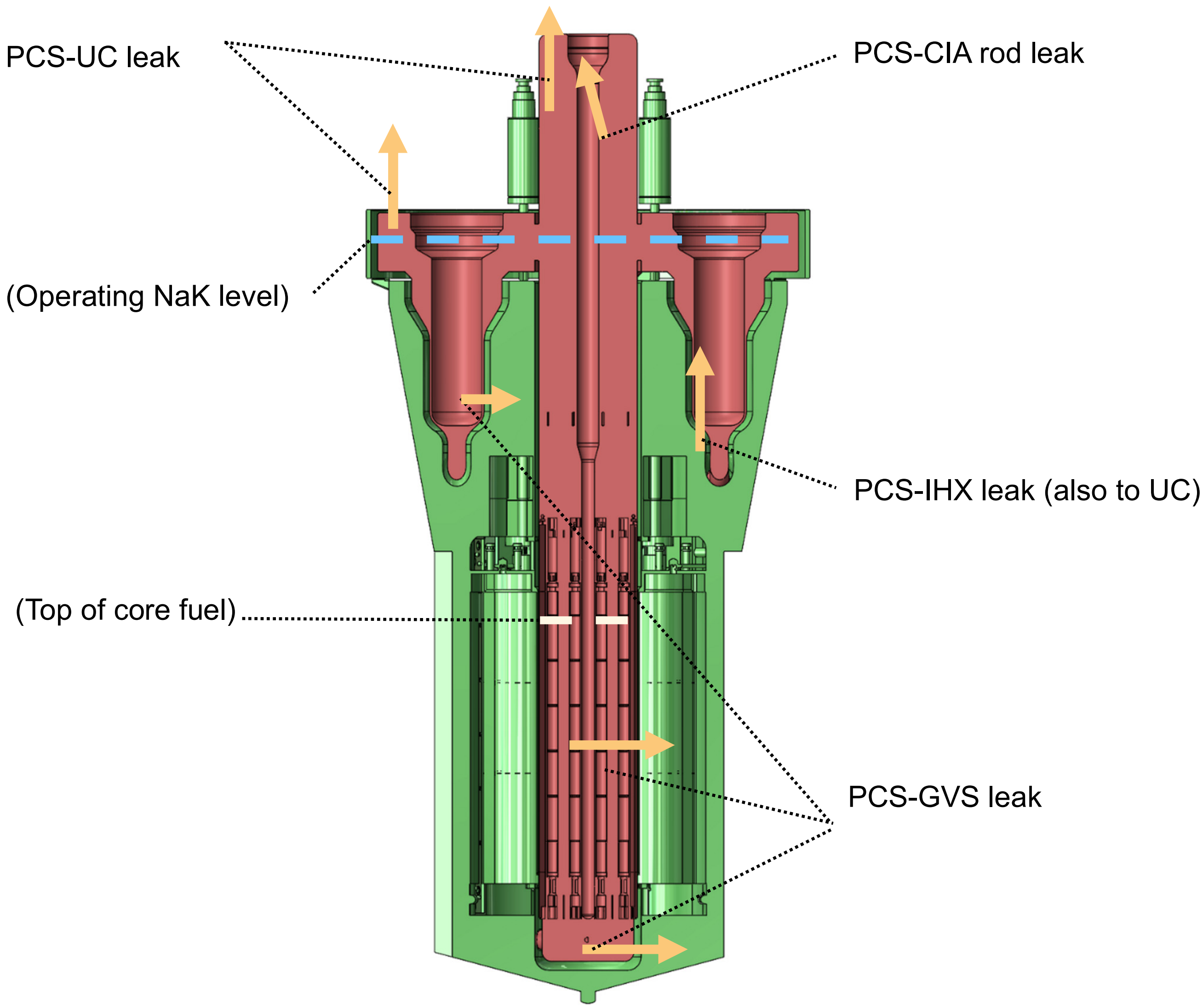


Figure 2. Potential PCS leak locations.

Figure 3. Distribution plenum upper surface highlighted in blue.

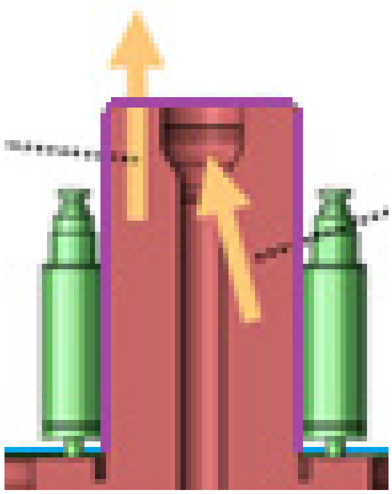


Figure 4. Closure head outlined in purple.[a]

2.1.3.10 Increase in Reactor Coolant Inventory—This is not applicable to the MARVEL design.

2.3.2 External Hazard Events. The external hazard events group includes equipment- and human-induced events and NPH events external to the reactor which disrupt normal plant operations. The external groups are as follows:

1) Seismic Events
2) External Floods, Fires, High Winds/Tornadoes, Extreme Temperatures, and Lightning
3) Radioactive or Hazardous Material Release or Direct Radiation Exposure from a Subsystem or Component
4) TREAT Facility Fires
5) TREAT Facility Flooding
6) TREAT Crane or Equipment Impacts to MARVEL Equipment.

The results of the grouping are summarized in the subsections below. As discussed in SAR-420 Chapter 15, external events include plane crash, vehicle crash, and adjacent building fire/explosion. As concluded in SAR-420, the frequency of an aircraft crash into the TREAT facility is less than 10^{-6} events/year and will not be considered as an accident initiator in the MARVEL safety analysis.

2.3.2.1 Seismic Events—The MARVEL reactor and support safety systems are categorized as SDC-2, and the other facility handling systems are categorized as SDC-2 or less, per the criteria in DOE-STD-1020-2016.

As defined by the NRC, and operating basis earthquake (OBE) is an earthquake "that could be expected to affect the site of a nuclear reactor, but for which the plant's power production equipment is designed to remain functional without undue risk to public health and safety." As defined by the NRC, a safe shutdown earthquake (SSE) is "the maximum earthquake potential

a. Heater tubes are not shown in this diagram. They are positioned radially between the outside diameter of the CIA rod and the inside diameter of the closure head.

for which certain structures, systems, and components, important to safety, are designed to sustain and remain functional."

The SSE for MARVEL is defined such that its occurrence frequency is less than or equal to one per 2,500 years. For TREAT, a reasonable value was selected for the magnitude of the SSE. That value was based on an earthquake of low magnitude with an annual 99.99% probability of not being exceeded. The value is equivalent to a 50-year annual service-life probability of 1×10^{-4}. SAR-420 Figure 2-6 indicates that the SSE acceleration level of 0.22 g gives an annual probability of between 10^{-3} and 10^{-4}. Similar to TREAT, a seismic early warning trip signal is included for the MARVEL design. This trip signal on the p-wave would allow the reactor to trip before the more damaging s-waves generated by an earthquake are expected to arrive at the site. It is assumed that the seconds of advance warning can allow the reactor to receive the trip signal and insert the CDs.

MARVEL core and internals SSCs are designed to the SDC-2 seismic event; therefore, core damage as a result of a seismically induced IE at the level of the SSE is not considered credible and is considered as beyond the design bases.

The TREAT building and cranes have been analyzed to performance category (PC)-2 seismic criteria (considered equivalent to SDC-2); therefore, TREAT building and crane SSCs impacting MARVEL SSCs as a result of a seismic event are not analyzed. In addition, MARVEL SSCs have been analyzed to SDC-2 seismic criteria; therefore, MARVEL equipment and SSCs impacting TREAT SSCs during a seismic event are not analyzed.

A BDBA is evaluated for a seismic event greater than the SDC-2 level that TREAT building structures and cranes are evaluated to withstand. The BDBA seismic event results in failure of TREAT structures and crane and system impact to the MARVEL reactor in the TREAT pit. The impact results in core rearrangement/compaction, and an extreme reactivity insertion leading to an energetic core disassembly.

2.3.2.2 External Floods, Fires, High Winds/Tornadoes, and Lightning— SAR-420 Section 3.3 discusses in detail the TREAT facility responses to the following NPH events:

- Straight-line winds
- Tornados
- Missile protection
- Extreme temperatures
- Snow loads
- Floods
- Lightning
- Range fire.

As discussed in detail in SAR-420 Section 3.3, TREAT SSCs are considered to have adequate protection from the above NPH events. The MARVEL system will be located in the TREAT facility in the north high-bay equipment pit. Therefore, the MARVEL system is considered to also have an adequate level of protection against the above NPHs.

2.3.2.3 Radioactive or Hazardous Material Release, or Direct Radiation Exposure, from a System, Subsystem or Component—This section addresses events that could result in a radioactive release from a component or system other than the reactor coolant system. MARVEL does not refuel since the initial fuel load is designed to last the entire operating life of the core. Fuel handling during initial loading does not pose any safety challenge associated with radionuclide release since no fission products or high-activity radionuclides are present before operation commences. MARVEL operations have the potential to impact TREAT operations and SSCs as a result of radioactive and hazardous materials or direct radiation, and vice versa. Hazardous materials include NaK, liquid metal secondary coolants (e.g., Pb or eGa-In-Sn), and Be. MARVEL specific IEs for this group include:

- Radioactive or hazardous material release due to fuel/cladding, MARVEL PCS or SCS SSC structural failure, or cover gas system breach.
- Radioactive or hazardous material release from drops/impacts (failure of lifting hardware or operator error) of fresh fuel or Be reflector materials during MARVEL handing operations during initial core loading.
- Radioactive or hazardous material release from contaminated NaK spills during fuel or PCB loading or unloading operations or PCB breaches.
- Radioactive or hazardous material release from drops/impacts (failure of lifting hardware or operator error) of used fuel or casks, Stirling engines, IHXs, or contaminated components such as CDs during PCB repair/replacement/maintenance/unloading operations.
- Stirling engine helium tube rupture leads to high energy gas release that would cause activated secondary coolant and cover gas release to upper confinement.
- Direct radiation exposure during MARVEL or TREAT reactor operations or from used fuel or contaminated components during repair/replacement/maintenance/unloading operations, or failure of pit shielding structure.
- Direct radiation exposure due to reflector material loading error, design error , or structural failure.
- Radioactive or hazardous material release from system impacts on barrel or equipment from cranes, failure of pit shielding structure or other heavy loads, or vehicles.
- Radioactive material or direct radiation release from inadvertent criticality outside of barrel.
- IHX failure and leak of contaminated NaK outside of reactor confinement.
- Radioactive material release from helium gas generation due to neutron absorption by B_4C.
- Release of Ar-41 and Polonium (Po)-210 as a result of normal operations.
- Reactor shielding SSC material or structural failure due to impact from drop of heavy load over the reactor.
- Heavy load drops results in failure of cover gas SSCs.

TREAT crane operations over the MARVEL reactor will be limited by administrative control to reduce the time-at-risk over the reactor (i.e., the crane will not be “parked” over the reactor when not being used). TREAT equipment/crane drops are evaluated for MARVEL that involve the potential for the release of radioactive or hazardous materials.

2.3.2.4 Facility Fires—Facility fires that may occur external to the reactor may prompt operator action to shut down the reactor until the fire has been suppressed and the reactor can return to normal operation. MARVEL specific IEs for this group include:

- TREAT facility fire or internal MARVEL system fire, resulting in a loss of a support system initiating reactor shutdown either in direct response based on a loss of equipment or initiated by operators (covered under general transients IE).
- TREAT facility fire or internal MARVEL system fire, resulting in a LOP and initiating a reactor shutdown (covered under LOP IE).
- NaK spill and fire during fuel loading or unloading.
- IHX failure and leak of NaK outside of reactor confinement.
- Engine cooling system (ECS) glycol fire.
- PCB (including GV) penetration leaks/breaks/seal ruptures and NaK leak and fire.
- PCS, PCMS, IGS, or SCS SSC material or structural failure and NaK leak and fire.

PCB breaches may result in 1) NaK leakage outside of confinement, 2) NaK fires, 3) adverse NaK-concrete interactions in the TREAT pit, and 4) potential for loss of DHR functions. NaK spills are not considered an IE but could result from another IE and equipment failure, and may occur during operations, loading, or unloading.

PCB breaches due to weld or other failures in the barrel downcomers are of the greatest concern to reactor safety, given the ability to 1) disable heat removal pathways and potentially impact other reactor equipment, 2) result in a NaK fire in the pit, or 3) result in adverse NaK-concrete interactions. However, if a downcomer breach were to occur, the guard vessel is a credited safety SSC to prevent these interactions from occurring.

Consequences from PCB breaches could include the release of radionuclides from contaminated NaK, and non-radiological consequences to collocated workers or the public. NaK spills and fires are included in the external hazards analysis and accident analysis to quantify consequences and identify preventive/mitigative SSCs and controls.

MARVEL barrel and piping SSCs are designed to the SDC-2 seismic event; therefore, a seismic spill and fire as a result of a seismically induced loss of coolant event is not analyzed.

A NaK fire may also occur due IHX corrosion and failure, and NaK leakage to the MARVEL upper confinement structure.

2.3.2.5 Facility Flooding—Flooding within the TREAT facility due to TREAT building fire suppression system activation or other water leaks or line breaks may result in:

- Loss of a support system initiating reactor shutdown either in direct response based on a loss of equipment or initiated by operators (covered under General Transients IE).
- Flooding of MARVEL pit and degradation of passive decay removal from the pit (Covered under Decrease in Heat Removal IE).

2.4 Fundamental Safety Functions

The MARVEL FSFs (See Figure 5) are defined as high-level important safety functions that if satisfied, will provide reasonable assurance of adequate protection of the public, worker, and environment. Consistent with the definition in Nuclear Energy Institute (NEI)-18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,"[18] the MARVEL FSFs include control of reactivity and heat generation in the core, control of heat removal, and confinement of radioactive material. The FSFs are necessary to keep the IEs identified in Section 2.3 from progressing to end states that could result in fuel, cladding, or PCB damage and release of radioactive or hazardous material.

2.4.1 Reactivity Control. The FSF of controlling reactivity is to control reactivity and thereby control heat generation rate, to prevent abnormal conditions from escalating into a more significant event. Reactivity control also helps facilitate any response to an accident, should one occur, by shutting down the nuclear reaction and reducing the heat generation within the plant that other installed systems would be required to mitigate.

The importance of reactivity control for MARVEL 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 leads to changes in core temperatures. As such, the first means of limiting core temperatures is by the control of the reactivity of the reactor through the insertion of negative reactivity.

The evaluated MARVEL micro-reactor design has the following strategies for reactivity control:

1. CDs
2. IRF
3. Manual Scram
4. CIA rod.

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. The CDs release following a signal from the RPS to provide insertion of negative reactivity to shut down the reactor and maintain it in shutdown condition.

The MARVEL RPS is composed of 1) the reactor trip system which monitors reactor process variables and sends a reactor trip signal when a process variable exceeds a limit setpoint, or as a result of a seismic trip, and 2) the portion of the reactivity control system that implements a shutdown command by rapidly inserting all CDs by means of passive return mechanisms associated with each drum.

As shown in ECAR-6332[15] a single CD can bring the reactor subcritical at HFP conditions with the other 3 CDs at their hard stop limits and the CIA rod fully withdrawn. This provides excellent redundant shutdown capability as there are four independently controlled CDs. With successful RPS trip and reactor shutdown by the CDs, the reactivity control FSF is met.

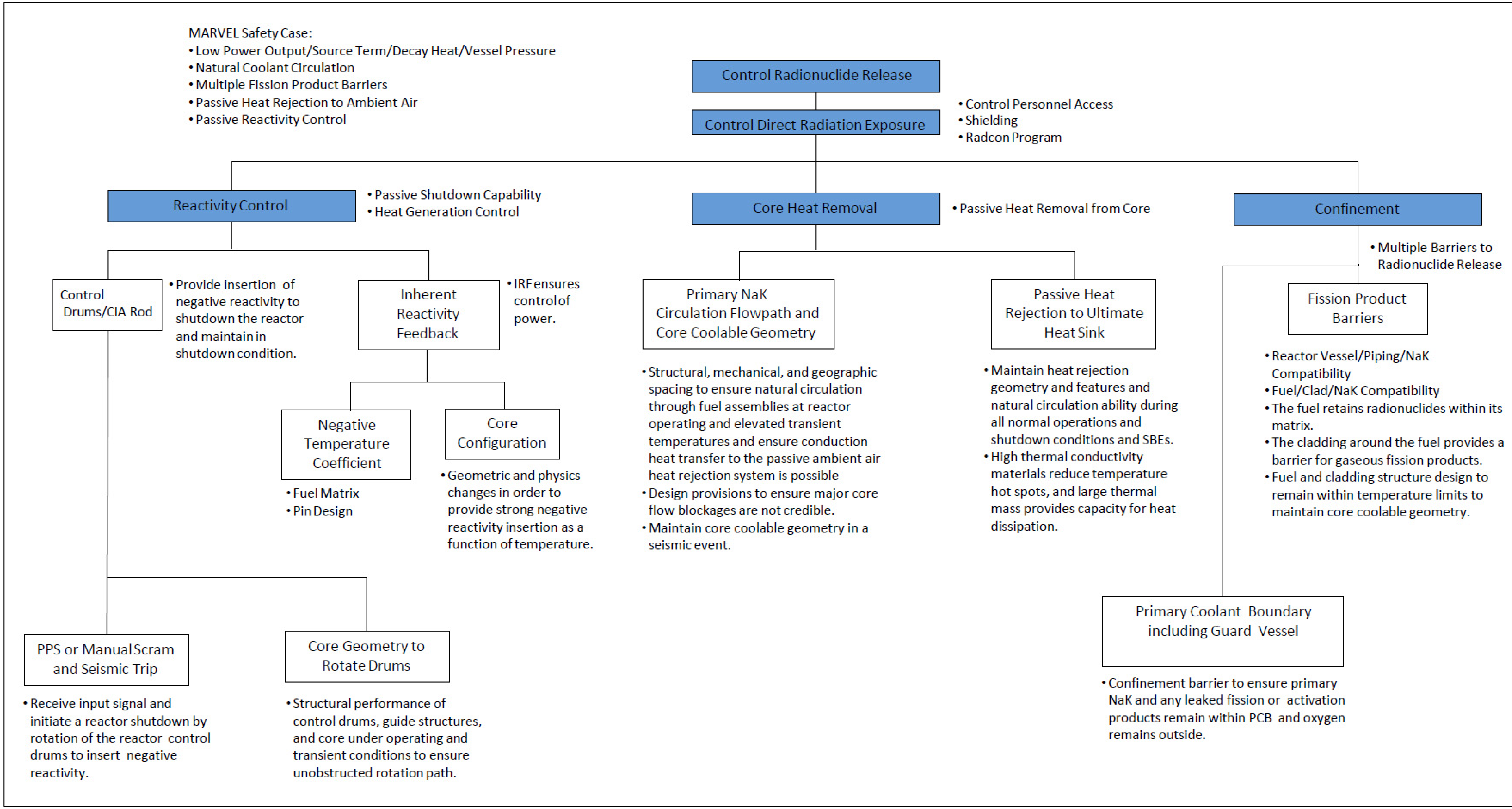


Figure 5. MARVEL fundamental safety functions.

If RPS trip and insertion of the CDs is unsuccessful, the reactor can be brought to a safe stable state by IRF. 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 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, and core damage occurs.

The fuel and core system (FCS) is 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, cladding, or PCB temperature limits are challenged, and core damage occurs. The reactivity coefficients of the MARVEL reactor system are required to have a net negative IRF. The UZrH fuel form was selected because of its demonstrated strong negative reactivity feedback with temperature increases in historical Training, Research, Isotope, General Atomics (TRIGA) research reactors.

The fuel prompt reactivity coefficient is large and negative. The fuel reactivity feedback dominates the IRF effects. There are positive reactivity feedback effects from the metallic beryllium and beryllium oxide components along with the sodium-potassium eutectic (NaK) coolant. These effects are significantly smaller than the negative feedback from the fuel. However, these positive feedback effects will need to be taken into account during operations as they will provide delayed reactivity increases as the bulk coolant and external neutron reflectors heat up. Overall, the calculated reactivity balance demonstrates the MARVEL system will be stable and inherently safe with no risk of an uncontrolled reactivity increase feedback loop. Even though these features suppress core power to match the heat removal rate, operators must manually initiate the insertion of the CDs (manual scram) to reach subcriticality.

The CIA rod is an annular rod composed of boron carbide withdrawn vertically from the core. On a scram gravity inserts it into the core. Another feature of the CIA rod is the ability to incorporate a hafnium burnable absorber rod (gray rod). Early in core life the gray rod is inserted to reduce excess reactivity. Later in core life, the rod is withdrawn to compensate for fuel burnup. 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 to supplement the use of the CDs.

2.4.2 Core Flow/Heat Removal. The evaluated MARVEL micro-reactor design has the following strategies for heat removal:

1. Natural circulation and active heat removal via the PGS (Stirling engines) during normal operations and shutdown.
2. Passive conduction to large thermal masses, provided by structures and shielding, and connection to surrounding air.

The FSF of removing heat serves two critical objectives: 1) removal of the generated heat during all normal operations and shutdown conditions to assure that equipment would operate within the environmental envelope for which it is designed and qualified, and 2) to prevent an event 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 geometric 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 DHR system is possible. The MARVEL is designed to maintain passive heat rejection geometry and features and natural circulation ability during all normal operations and shutdown conditions and events.

Decay heat removal following successful shutdown may be through the normal operation engineered pathway through the active PGS Stirling engine heat exchanger to the ultimate heat sink, or passively thermally connected via conduction and radiation to large thermal masses, provided by structures and shielding, and convection to surrounding air. The Passive DHR system is designed to be capable of removing heat from the core at only decay heat levels, and to maintain a coolable geometry. This means fuel temperatures can remain below operating limits relying purely on passive conduction, convection, and radiation.

2.4.3 Confinement of Radioactive Material. The evaluated MARVEL micro-reactor design has the following strategies for limiting the release of radionuclides:

1. Fission products barriers, including fuel and cladding
2. PCB, including reactor barrel and piping (downcomers), and
3. Guard vessel.

The FSF of limiting the release of radioactive materials represents the ultimate objective of protecting the public from exposure to radiation. For many events, the previous two FSFs address the avoidance of precursor conditions that would challenge or exacerbate the release of radioactive materials. The plant design features included to limit potential releases to the environment, whether they be physical barriers or systems, present the final in-plant opportunity to assure that public health and safety are protected. The MARVEL confinement strategy is derived from a performance-based perspective in that the performance requirements are derived from the accident analysis and not prescriptively identified by the general design criteria.

The MARVEL fuel retains many radionuclides within its matrix. The cladding around the fuel provides a barrier for gaseous fission products (i.e., xenon, krypton). Damage to the fuel cladding releases radionuclides, and possibly air from gap and plenum fill gas to the primary NaK coolant. The air may react with reflector graphite and the NaK coolant. However, due to small volume of air in the gap and interactions are considered negligible.

The NaK coolant acts as a third radionuclide barrier by retaining fission products by plate-out, chemical solubility, or adsorption mechanisms. The PCB design, which includes the reactor and guard vessels and any associated piping, including the downcomers, ensures primary NaK and any leaked fission or activation products remain within PCB and oxygen remains outside. The PCB is required to remain intact and could be degraded as a result of increased bulk coolant temperature.

2.4.4 Direct Radiation Exposure Control. The TREAT facility design will provide for meeting these criteria by the following:

1. Shielding and containment of radioactivity and radiation sources
2. Ventilation operation
3. Radiation and radioactivity monitoring instrumentation.

The MARVEL reactor requires shielding for neutron, gamma, and beta radiation. Various combinations of boron, concrete, hydrogenous material, and steel, used as described in the following section for shielding against beta, gamma, and neutron radiations. Alpha particles and beta particles are largely associated with the reactor fuel; the containment vessel or fuel cladding provide adequate shielding against these radiations under planned operations.

When possible, equipment and components that require maintenance or testing are located in low- radiation areas. To minimize exposure, the facility design takes into account personnel and equipment traffic patterns. Also incorporated into the facility design are accessibility and space requirements for equipment removal or maintenance.

Before the start of MARVEL operations, the Reactor Building (MFC-720) and the TREAT exclusion zone defined by the fence around the Reactor Building (see SAR-420 Figure 12-1) will be evacuated. Post operation radiation levels are shown on remote readouts in the control room. Before normal reentry of the Reactor Building, the readings on the continuous air monitors and radiation area monitors are monitored to verify that an abnormal condition is not present. After these prerequisites have been met and before general reentry of personnel is permitted, a health physics technician surveys the Reactor Building with a portable radiation monitor, providing a backup check on the fixed monitoring equipment. These procedures, supplemented by radiation area postings, alert operating personnel to areas where residual levels do not allow free access.

A source term will be developed to account for the neutron activation of the reactor structure and surrounding materials. The shielding is designed to minimize activation of materials outside of the reactor area to minimize the dose rate to workers in the TREAT building when occupied. This minimizes the impact to the TREAT transient testing mission and also complies with ALARA principles.

2.5 Event Sequence Analysis

2.5.1 Methodology. Consistent with the process in NEI-18-04, SBEs are defined from the entire collection of MARVEL event sequences considered in the design and safety of the plant. SBEs are defined in terms of event sequences comprised of an initiating event, the plant response (SSCs performing the FSFs) to the initiating event, which includes a sequence of successes and failures of mitigating systems, and a well-defined end state. Generally, for each internal or external IE in Section 2.3.1 or 2.3.2, respectively, an SBE sequence is developed with the varying functionality of SSCs performing the FSFs in Section 2.4.

A simplified accident progression event tree is shown in Figure 6 to assist in the qualitative SBE sequence analysis. It IS NOT a probabilistic risk assessment (PRA), but a simple qualitative tool from AIChE, “Guidelines for Hazard Evaluation Procedures, with Worked Examples,”[19] used in this SBE sequence analysis. Figure 6 event sequences may or may not require some modifications for the analysis of External Ies from Section 2.3.2, but the overall process remains the same considering the FSFs of Confinement and Control of Direct Radiation Exposure.

The event tree shown in Figure 6 is also similar to the structure of generic micro-reactor event tree found in SAND2020-4609, “Technical and Licensing Considerations for Micro-Reactors.”[20] The process used is largely qualitative as suggested by BNL-212380-2019-INRE, “Regulatory Review of Micro-Reactors – Initial Considerations, Brookhaven National Laboratory.”[21]

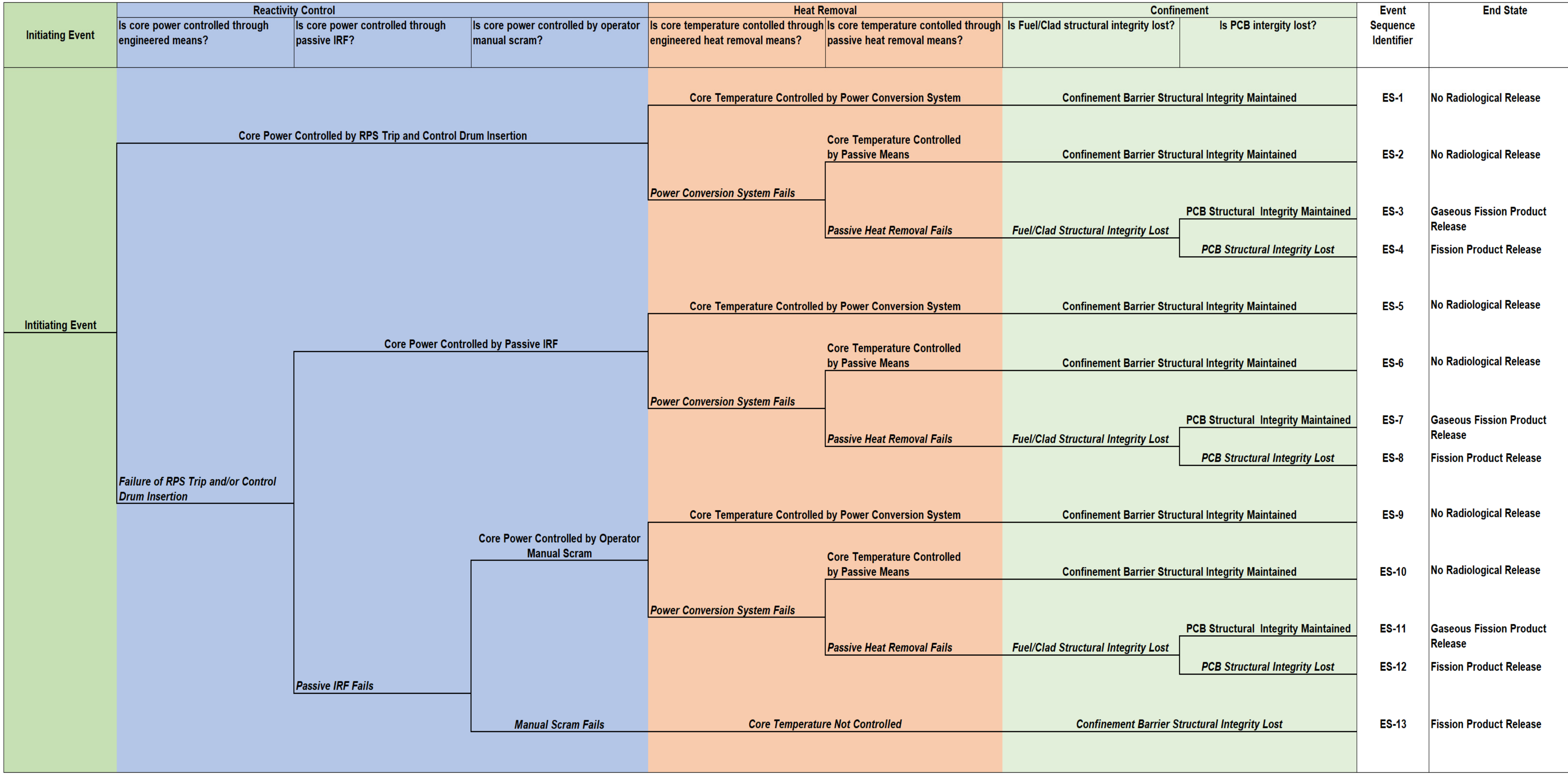


Figure 6. MARVEL SBE accident progression event tree.

2.5.1.1 Event Sequence Analysis Methodology—The SBE sequence analysis is performed in the following steps:

1. Identify internal or external IE in Section 2.3.1 or 2.3.2 respectively and assign a qualitative frequency category from Table 4 (see results in Table 5). IEs are assigned a qualitative frequency category based on review of safety basis documentation in Section 2.3 and based on engineering judgement. Due to the large uncertainties, the qualitative frequency assignments tend to error on the overconservative side of the categories given.

2. Develop the SBE sequences using Figure 6 starting with the IE and the reactor response assuming successful performance of active SSCs performing the reactivity control, decay heat removal, and confinement FSFs identified in Section 2.4 and Figure 5.

3. Determine the SBE sequence end state based on the success of the FSFs using the qualitative success criteria in Table 6.

4. Determine the overall SBE sequence qualitative frequency and consequences using Table 4 and Table 7 respectively.

5. Determine the associated risk bins for the public, collocated worker and facility worker using Figure 7, Figure 8, and Figure 9 respectively, assuming successes of the SSCs to perform their intended safety functions.

6. Develop the SBE sequences using Figure 6 starting with the IE and the reactor response assuming unsuccessful performance of the various active and passive SSCs performing the reactivity control, decay heat removal, and confinement FSFs.

7. Determine the SBE sequence end states using the success criteria in Table 6.

8. Determine the overall SBE sequence qualitative frequency and consequences using Table 5 and Table 7 respectively.

9. Determine associated risk bins for the public, collocated worker and facility worker using Figure 7, Figure 8, and Figure 9 respectively, assuming successes and failures of the various SSCs to perform their intended safety functions.

10. Identify candidate safety SSCs from the SBE sequence analysis.

Table 4. Qualitative frequency categories.

Frequency	Description	Likelihood Range (per year)
Anticipated (A)	Events that may occur during the lifetime of the facility (incidents that commonly occur).	Likelihood $>10^{-2}$
Unlikely (U)	Events that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this likelihood class include Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.	$10^{-2}>$ likelihood $>10^{-4}$
Extremely Unlikely (EU)	Events that will probably not occur during the lifetime of the facility.	$10^{-4} >$ likelihood $>10^{-6}$
Beyond Extremely Unlikely (BEU)	All other accidents.	Likelihood $<10^{-6}$

Ref: MCP-18121, "Safety Analysis Process."[22]

Table 5. Qualitative IE frequency assignments.

IE Group	Qualitative IE Frequency
INTERNAL EVENTS	
Shutdowns:	
- Manual shutdowns, Test Scrams, Spurious trips, Unintended rotation inward of the CDs, Xenon buildup.	- Anticipated
General Transients:	
- Minor core blockages (e.g., flow disruption between neighboring pins, random cladding failures)	- Anticipated
- Small reactivity changes (e.g., miss-positioning of a single CD through operator error or spurious trip while at power cause enough flux tilt to increase fuel temperature in a fuel pin)	- Anticipated
- Small coolant system NaK leaks	- Anticipated
- Loss of a support system from internal fire or flood	- Anticipated
- Excessive Stirling engine vibration	- Anticipated

Table 5. Qualitative IE frequency assignments.

IE Group	Qualitative IE Frequency
Decrease in Heat Removal by Secondary System [Loss of Heat Sink (LOHS)]:	
- Loss of a single PGS Stirling engine	- Anticipated
- Loss of multiple PGS Stirling engines	- Unlikely
- Small SCS pipe leaks/breaks	- Anticipated
- SCS, SSS, SOS, SCGS, or SCMS material or structural SSC failures	- Unlikely
- Loss of an IHX	- Unlikely
- Passive DHR system clogging from failure of the pit shielding structures, loose parts or debris, facility fires, internal flooding of TREAT pit, or external NPH	- Unlikely
- Stirling engine Hx failure and flow of gas into SCS	- Unlikely
- Blockage of passive heat removal pathway due to failure of shielding in a seismic event	- Extremely Unlikely (g>SSE)
- PGS Water Line Connection or pipe failure and leak	- Anticipated
Decrease in primary coolant system flow rate [Loss of Flow (LOF)]:	
- Core blockage (partial or total) due to debris	- Unlikely
- Core blockage due to distortion, bowing, or bulging of fuel pins or CDs	- Unlikely
- Core blockage from fuel/cladding pressure buildup or failure	- Unlikely
- Core blockage from fuel assembly, Be reflector, grid plate failure	- Unlikely
- Core blockage (partial or total) as a result of leakage of secondary coolant into the PCS due to corrosion induced failure of the IHX boundary (IHX wall and liner)	- Extremely Unlikely
- Low PCS pressure, NaK boiling	- Unlikely
- Reactor PCB leaks	- Extremely Unlikely

Table 5. Qualitative IE frequency assignments.

IE Group	Qualitative IE Frequency
Loss of power (LOP):	
- TREAT facility related LOP	- Anticipated
- Grid-related LOP	- Anticipated
- Switchyard-centered LOP	- Anticipated
- Weather-related LOP	- Anticipated
- Seismic-related LOP	- Unlikely
- TREAT facility fire or internal MARVEL system fire	- Unlikely
Reactivity and Power Distribution Anomalies [Transient Overpower (TOP)]:	
- Small reactivity changes (e.g., miss-positioning of a single CD through operator error or spurious trip while at power cause enough flux tilt to increase fuel temperature in an assembly)	- Covered under general transients
- Moderate reactivity insertion due to spurious CD or CIA movement due to electronics failures, heat damage, or radiation damage, physical damage, or material, structural, or seismic failures	- Unlikely
- Large reactivity insertion due to core events (misalignment of multiple CD or core configuration change due to bowing, melting or slumping of fuel)	- Extremely Unlikely
- Extreme reactivity insertions (misalignment of all CDs)	- Beyond Extremely Unlikely
- Fuel/cladding, NaK, CD, or reflector material loading error, structural failures, or misplacement/movement leads to greater or less excess reactivity or heat generation than expected	- Unlikely
- NaK voiding (such as gas entrainment)	- Anticipated
- Water intrusion into the core from PGS water line connection or pipe failure and leak during maintenance	- Anticipated
- Overcooling of the primary system by the power conversion unit (increase in heat removal)	- Anticipated

Table 5. Qualitative IE frequency assignments.

IE Group	Qualitative IE Frequency
Core faults:	
- Minor core blockages	- Covered under general transients
- Moderate core blockages	- Covered under decrease in flow rate
Local faults:	
- Enrichment error (Fuel manufacturing error or Uranium mass loading error) or Be material loading error leads to higher or lower enrichment than desired in a fuel rod leading to greater or less than heat generation than expected.	- Extremely Unlikely
Decrease in primary coolant system inventory [Loss of Coolant Accident (LOCA)]:	
- Coolant system small leaks	- Covered under general transients
- PCS Leak Inside Guard Vessel	- Extremely Unlikely
- PCS – Upper Confinement Leak	- Extremely Unlikely
- PCS-IHX Leak	- Extremely Unlikely
- Cover gas space leak	- Unlikely
- PCB penetration leaks/breaks/seal ruptures or support SSC failure	- Extremely Unlikely
- PCMS, IGS NaK system leak or failure	- Unlikely
EXTERNAL EVENTS	
Seismic events:	
- Seismic event (g <OBE)	- Anticipated
- Seismic event (g < SSE)	- Unlikely
- Seismic event (g >SSE)	- Extremely Unlikely
External Floods, Range Fires, High Winds/Tornadoes, and Lightning	- Not IE.

Table 5. Qualitative IE frequency assignments.

IE Group	Qualitative IE Frequency
Radioactive or Hazardous Material Release, or Direct Radiation Exposure, from a System, Subsystem or Component:	
- Radioactive or hazardous material release due to MARVEL PCS or SCS cover gas system breach	- Unlikely
- Radioactive or hazardous material release from contaminated NaK spills during fuel or PCB loading or unloading operations	- Anticipated
- Radioactive or hazardous material release from drops/impacts of fresh fuel or Be reflector materials during MARVEL handing operations during initial core loading.	- Anticipated
- Radioactive or hazardous material release from drops/impacts of used fuel or casks, Stirling Engines, IHX's, or contaminated components such as CD's during PCB repair/replacement/maintenance/unloading operations.	- Anticipated
- Direct radiation exposure during MARVEL or TREAT reactor operations, or used fuel or contaminated components during repair/replacement/maintenance/unloading operations, or failure of pit shielding structure	- Anticipated
- Radioactive or hazardous material release from system impacts such as cranes, other heavy loads, or vehicles	- Unlikely
- Radioactive material or direct radiation release from inadvertent criticality outside of PCB	- Beyond Extremely Unlikely
- IHX failure and leak of NaK outside of reactor confinement.	- Unlikely
- Radioactive material release from helium gas generation due to neutron absorption by B4C.	- Anticipated
- Release of Ar-41 as a result of normal operations.	- Anticipated
- Stirling engine helium tube rupture leads to high energy gas release that would cause secondary coolant release to upper confinement.	- Not IE.
Facility Fires:	
- Fire resulting in a loss of a support system initiating reactor shutdown either in direct response based on a loss of equipment or initiated by operators	- Covered under General Transients

Table 5. Qualitative IE frequency assignments.

IE Group	Qualitative IE Frequency
- Fire resulting in a LOP and initiating a reactor shutdown	- Covered under LOP
- NaK spills and fires	- Anticipated
- IHX failure and leak of NaK outside of reactor confinement.	- Unlikely
Facility Flooding:	
- Loss of a support system initiating reactor shutdown either in direct response based on a loss of equipment or initiated by operators	- Covered under General Transients
- Flooding of MARVEL pit and degradation of passive heat removal from the pit	- Covered under Decrease in Heat Removal

Table 6. SBE sequence qualitative end state success criteria (As adapted from SAR-420 Table 15-1).

Frequency Category	General Guidelines	Reactor Shutdown Fuel/Cladding Guidelines	Non-Fuel Shutdown Heat Removal Guidelines	Shutdown Heat Removal
Anticipated (A)	• The facility should be capable of returning to operation without extensive corrective action or repair.	• No additional barrier damage or failure occurs beyond the IE. • No fuel damage occurs beyond the IE. • No impact on fuel integrity or lifetime.	• No loss of reactor shutdown and decay heat removal functions occurs. • No loss of integrity or function of barriers containing radioactive material	ASME Service Level “B” Limits
Unlikely (U)	• Facility should be capable of returning to operation following corrective action or repair of damage.	• A coolable geometry is maintained for the fuel. • No fuel melting or other condition, such as excessive fuel temperature, occurs that could result in the uncontrolled movement of fission products and/or fuel from their intended location.	• At least one means of reactor shutdown and decay heat removal remains functional. • Confinement functional capability is maintained by at least one barrier to control the release of fission products or other radioactive material to the environment.	ASME Service Level “C” Limits
Extremely Unlikely (EU)	• Facility damage may preclude return to operation.	• Assess design capability with respect to the accident prevention and mitigation strategy to meet EGs.	• Assess design capability with respect to the accident prevention and mitigation strategy to meet EGs.	ASME Service Level “D” Limits
Beyond Extremely Unlikely (BEU)	• No criteria	• No criteria	• No criteria	

Table 7. Qualitative radiological and non-radiological consequence category.

Consequence Category[a]	Facility Workers	Collocated Workers	Offsite Public
High (H)	Greater than 100 rem or PAC-3	Greater than 100 rem or PAC-3	Greater than 25 rem or PAC-2
Moderate (M)	25 rem to 100 rem or PAC-2	25 rem to 100 rem or PAC-2	5 rem to 25 rem or PAC-2
Low (L)	Less than 25 rem or Less than PAC-2	Less than 25[b] rem or Less than PAC-2	Less than 5 rem or Less than PAC-2

a. The numerical consequences category guidelines for the offsite public, onsite (collocated) workers, and facility workers are based on the risk EGs and criteria for the selection of safety SSCs and TSRs established for INL nuclear facilities as supplemental guidance from DOE-ID.

b. When the consequence of a radiological release challenges 25 rem to the collocated worker, the consequence category is moderate, and the application of TSR-level controls is required. Additional safety analysis report commitments (SARCs) should be considered as the collocated worker consequence(s) approach 5 rem (total effective dose [TED]).

NOTES:

1. The offsite public is a hypothetical maximally exposed individual at the INL site boundary.

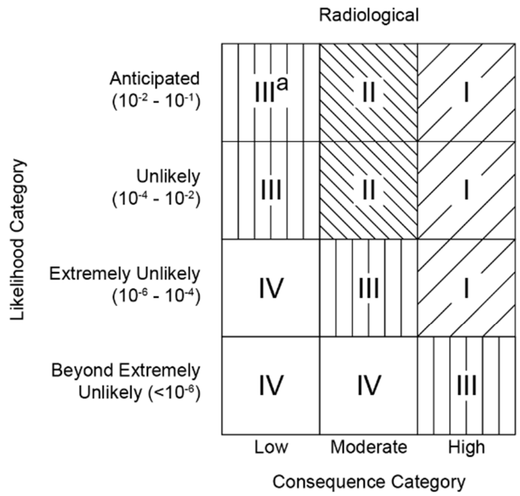
2. The collocated worker is located outside the facility and is assumed to be at least 100 m from the release or, for elevated or buoyant releases, at the point where the release reaches ground level.

3. The facility worker is inside the facility (i.e., in the immediate vicinity of the release).

4. Radiological exposures (rem) are TED.

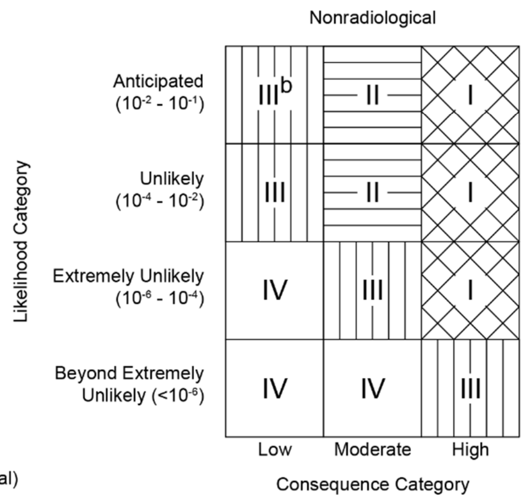
5. Explosion overpressure, expressed as the differential pressure (Δ psi) of the shock wave, is addressed as part of the protective action criteria (PAC) determination.

Ref: MCP-18121.



Severity Level	Risk Class
Major	I
Serious	II
Marginal	III
Negligible	IV

Consequence Category	Off-site Public
High (H)	greater than 25 rem or PAC-2
Moderate (M)	5 rem to 25 rem or PAC-2
Low (L)	less than 5 rem or less than PAC-2



KEY

- Safety-class SSCs and/or TSRs shall be identified to manage off-site public risk; accident analysis may be needed. (Risk class I radiological)
- Safety-class SSCs and/or TSRs should be considered to manage off-site public risk. (Risk class II radiological)
- Safety-significant SSCs and/or TSRs shall be identified to manage off-site public risk; accident analysis may be needed. (Risk class I chemical)
- Safety-significant controls should be considered to manage off-site public risk. (Risk class II chemical)
- Defense-in-depth SSCs or SMPs are adequate. (Risk class III)
- Safety SSCs or TSRs are not required to manage off-site public risk. (Risk class IV)

a. DID provisions are the primary means of mitigating low consequence, anticipated events. However, given the implications of any unplanned exposure, consideration should be given to whether or not adequate DID exists for mitigation of anticipated scenarios. For instances where the DID may be insufficient, establishment of SARCs safety SSCs, or TSRs may be warranted for events that challenge the EG.

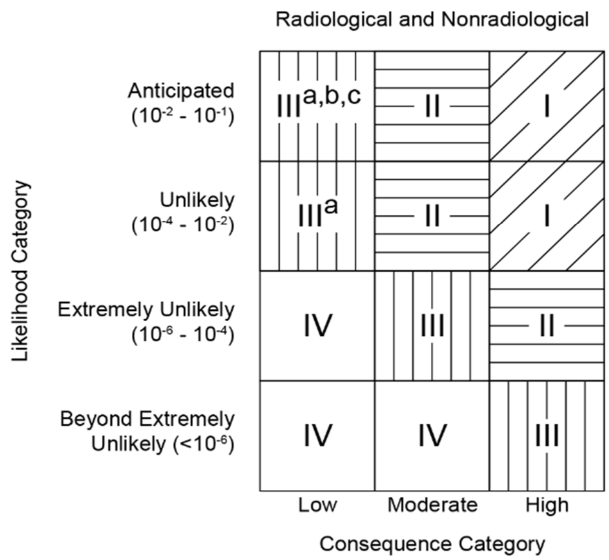
b. For anticipated events that challenge the PAC-2 to the public, safety-significant controls should be considered.

16-GA50208-07-R1

Figure 7. Qualitative risk matrix for the public (Ref: MCP-18121).

Severity Level	Risk Class
Major	I
Serious	II
Marginal	III
Negligible	IV

Consequence Category	On-site (Collocated) Workers
High (H)	greater than 100 rem or PAC-3
Moderate (M)	25 rem to 100 rem or PAC-2
Low (L)	less than 25 rem or less than PAC-2



KEY

Safety-significant SSCs and/or TSRs are required to manage collocated worker risk; accident analysis may be needed. (Risk class I)

Safety-significant controls should be considered to manage collocated worker risk. (Risk class II)

Defense-in-depth SSCs or SMPs are adequate. (Risk class III)

Safety SSCs, TSRs, or safety analysis commitments are generally not required to manage collocated worker risk. (Risk class IV)

a. Application of TSR controls is required when consequences of an anticipated or unlikely radiological release challenges 25 rem (TED) to the collocated worker.
b. Additional DSA commitments should be considered as the consequences of anticipated radiological releases approach 5 rem (TED) to the collocated worker.
c. For anticipated events that challenge the PAC-2 to the collocated worker, safety-significant controls should be considered.

16-GA50208-08-R0

Figure 8. Qualitative risk matrix for the collocated worker (Ref: MCP-18121).

Severity Level	Risk Class
Major	I
Serious	II
Marginal	III
Negligible	IV

Consequence Category	Facility Workers
High (H)	greater than 100 rem or PAC-3
Moderate (M)	25 rem to 100 rem or PAC-2
Low (L)	less than 25 rem or less than PAC-2

Radiological and Nonradiological

Likelihood Category	Low	Moderate	High
Anticipated (10^{-2} - 10^{-1})	III[a]	II	I
Unlikely (10^{-4} - 10^{-2})	III	II	I
Extremely Unlikely (10^{-6} - 10^{-4})	IV	III	II[b]
Beyond Extremely Unlikely (<10^{-6})	IV	IV	III

Consequence Category

KEY

Safety-significant SSCs and/or TSRs are required to manage facility worker risk. (Risk class I)

Safety-significant controls should be considered to manage facility worker risk. (Risk class II)

Defense-in-depth SSCs or SMPs are adequate. (Risk class III)

Safety SSCs, TSRs, or safety analysis commitments are generally not required to manage facility worker risk. (Risk class IV)

a. DID provisions are the primary means of mitigating low consequence, anticipated events. However, given the implications of any unplanned exposure, consideration should be given to whether or not adequate DID exists for mitigation of anticipated scenarios. For instances where the DID may be insufficient, establishment of SARCs, safety SSCs, or TSRs may be warranted for events that challenge the EG.

b. For extremely unlikely, high consequence events, consider the applicability of TSR-400 SMP controls for adequacy. If TSR-400 controls are not adequate, assess the practicality of designating additional controls. Otherwise, worker safety is ensured by TSR-400 controls for SMPs.

16-GA50208-09-R0

Figure 9. Qualitative risk matrix for the facility worker (Ref: MCP-18121).

2.5.1.2 Event Sequence Frequency Analysis Methodology—A fundamental challenge to evaluating the likelihood of a range of event scenarios that could occur for a microreactor is the assessment of the reliability of the different inherent safety features. Assuming perfect reliability of these inherent safety features would lead to the conclusion that no adverse end state for an event scenario could arise for MARVEL. For this simplified analysis for MARVEL, frequency adjustments to the IE in each event sequence analysis for SSCs performing FSFs are conservatively based on the frequency reduction guidelines in Table 2-1 in "R. Boston letter to J. Alvarez, "Department of Energy, Office of Nuclear Energy, Idaho Operations Office 2020 Documented Safety Analysis Review and Oversight Guidance (CLN201105),"[23] as follows:

- Administrative controls (including maintenance) procedures: 1 order of magnitude reduction in frequency.
- Active mechanical/electrical engineered safety features without redundant design: 2 order of magnitude reduction in frequency.
- Active mechanical/electrical engineered safety features with redundant and/or independent design features: 3-4 order of magnitude reduction in frequency.
- Passive SSC: 3-4 order of magnitude reduction in frequency.

Based on the above guidance, and other reactor frequency analyses, the following are conservatively used for evaluating the qualitative risk reductions for crediting SSCs performing FSFs in the event sequences:

- Probability of failure of the active reactivity control FSF mechanism to control reactor power is given a one qualitative frequency category frequency reduction (e.g., Anticipated to Unlikely).
- Probability of failure of the passive reactivity control FSF mechanism to control reactor power is given a two qualitative frequency category frequency reduction (e.g., Anticipated to Extremely Unlikely).
- Probability of failure of manual scram reactivity control FSF mechanism to control reactor power is given a two qualitative frequency category frequency reduction (e.g., Anticipated to Extremely Unlikely).
- Probability of failure of manual scram reactivity control FSF mechanism to control reactor power due to human error is given a one qualitative frequency category frequency reduction (e.g., Anticipated to Unlikely).
- Probability of failure of the active heat removal control FSF mechanism to control core temperature is given a one qualitative frequency category frequency reduction (e.g., Anticipated to Unlikely).
- Probability of failure of the passive heat removal control FSF mechanism to control core temperature is given a two qualitative frequency category frequency reduction (e.g., Anticipated to Extremely Unlikely).

As discussed previously a PRA is not performed. A qualitative judgement is made as to the likelihood and consequences of SSC success and failure of meeting the FSFs. Also as discussed previously, large uncertainty exists in the frequency and consequence assignments due to lack of system/component reliability data for micro-reactors. Due to the large uncertainties, the qualitative frequency and consequence assignments in this evaluation tend to error on the overconservative side of the ranges given.

Different levels of potential challenge to the core are postulated based on the response of the reactor FSFs to an IE, which may or may not result in actual “damage” of the confinement barriers (see Figure 3). A cumulative uncontrolled (unmitigated) core challenge scenario is therefore defined when there is a loss of a reactivity control FSF, active and passive heat removal capabilities are lost, therefore resulting in a loss of a confinement barriers and a radiological release (see Figure 6, ES-13).

2.5.1.3 Event Sequence Consequence Analysis Methodology—If as shown in Figure 6 an event sequence occurs that results in a potential of a radiological or a non-radiological release, based on the bounding dose consequence analysis in ECAR-5127:

- A dose of 2.65 rem at 6,000 m, is, therefore, assumed to be “low” (per Table 7) to the public.
- A dose of 27.5 rem to the collocated worker at the TREAT control room (770 m away) is “moderate” (per Table 7).

If as shown in Figure 6 an event sequence occurs that results in a potential of a gaseous fission product release, based on the dose consequence analysis in ECAR-5127:

- The dose is assumed to be “low” (per Table 7) to the public.
- The dose to the collocated worker at the TREAT control room (770 m away) is “low” (per Table 7).

2.5.1.4 Event Sequence Success Criteria Methodology—For each MARVEL FSF, it is first necessary to determine the applicable success criteria (or the level of performance required to consider successful operation) for the event sequence analysis (see Figure 6) as follows:

<u>Reactivity Control:</u>

Following an IE, the first response considered is the engineered reactivity control system (inward rotation of the CDs by the RPS) to perform the reactivity control FSF. There are two possible outcomes considered for this safety function:

1. Reactivity controlled.
2. Reactivity not controlled.

The MARVEL RPS is composed of the reactor trip system that detects the need for and initiates a reactor shutdown, and the portion of the reactivity control system that implements a shutdown command by rapidly inserting all CDs by means of the return mechanisms associated with each drum. A single CD can bring the reactor subcritical at HFP conditions. With successful RPS trip and passive reactor shutdown by the CDs, the reactivity control FSF is met.

If RPS trip and insertion of the CDs is unsuccessful, as documented in ECAR-6332, the reactor may be brought to a stable state by passive IRF. 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 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.

While such a negative IRF is generally highly reliable based on past experience with nuclear reactors examined, two possible outcomes are considered for this top event:

1. Reactivity controlled.
2. Reactivity not controlled.

Although successful IRF suppresses core power to match the heat removal rate, operators must manually initiate the insertion of the CDs (manual scram) to reach subcriticality. If in the unlikely event IRF fails to control reactor power, manual scram by the operators may be required to shut down the reactor and to ensure that core power matches the heat removal rate. Two possible outcomes are considered for this top event:

1. Reactivity controlled.
2. Reactivity not controlled.

Core Heat Removal:

Decay heat following successful shutdown may be through the normal operation engineered pathway through the active PGS Stirling engine heat exchanger to the ultimate heat sink, or passively conducted to large thermal masses, provided by core structures and shielding, and convection to surrounding air. Through either system, two modes are possible:

1. Core temperature controlled.
2. Core temperature not controlled.

However, passive decay heat removal is capable of removing heat from the core only at decay heat levels. The geometry of the core is assumed to remain coolable through the passive heat removal system. This means fuel temperatures can remain below operating limits relying purely on passive conduction, convection, and radiation as demonstrated in ECAR-6332.

Confinement Barriers:

The MARVEL fuel retains many radionuclides within its matrix. The cladding around the fuel provides a barrier for gaseous fission products (i.e., xenon, krypton). Damage to the fuel cladding releases radionuclides to the primary NaK coolant. Thus, fuel pin fuel and cladding barrier modes of operation include:

1. Structural integrity maintained.
2. Structural integrity lost.

Based on the analyses in INL/RPT-22-68555, “MARVEL Reactor Fuel Performance Report,”[24] both fuel-cladding chemical interactions and fuel-cladding mechanical interactions are negligible throughout the fuel’s operational cycle under both normal and high temperature accident scenario conditions. The MARVEL fuel element maintains its geometric stability and structural integrity during the most extreme accident scenarios predicted for the MARVEL reactor.

A conservative MARVEL fuel meat temperature limit of 950 and 1,000°C for the clad are recommended presently (which is nearly 300°C higher than the peak fuel temperature predicted to occur during the most extreme accident). Based on the known properties and behavior of the MARVEL fuel element, the fuel successfully meets its design and safety requirements under normal and most extreme beyond design basis accident conditions. It is considered beyond design bases that a release of radioactive material could occur during normal operations and accident conditions.

However, for conservatism, as shown in the following event sequence analysis, in certain accident conditions, it is temperatures are postulated to exceed the limits for the fuel/cladding confinement barriers, however, PCB structural integrity maintained. Gaseous fission product release from fuel/cladding gas gap to the primary coolant postulated to occur. It is also postulated that a minor release of gaseous fission products may occur though PCB failures.

As discussed in Section 1.2, the NaK coolant acts as a third radionuclide barrier by retaining fission products by plate-out, chemical solubility, or adsorption mechanisms. The fourth barrier PCB is required to remain intact and may be degraded as a result of increased bulk coolant temperature. Thus, PCB degraded modes of operation include:

1. Structural integrity maintained.
2. Structural integrity lost.

Direct Radiation Exposure:

The evaluated MARVEL micro-reactor design has three strategies for limiting the exposure to direct radiation:

1. Shielding.
2. Control of personnel access during MARVEL and TREAT reactor operations.
3. INL radiation program.

Only shielding is evaluated in the event sequence analysis. Thus, shielding degraded modes of operation include:

1. Shielding integrity maintained
2. Shielding integrity lost.

2.5.2 Event Sequence Analysis Results Summary. Table 8 summarizes the results of the internal SBE analyses, and Table 9 the external analysis The format is similar to a process hazards analysis (PrHA) from Reference 19, and the hazards analysis in Table 8-12 in SAR-406, “Safety Analysis Report for the Neutron Radiography Reactor.”[25] However, to understand all possible accident sequences and response of the reactor FSF SSCs, both mitigated and unmitigated SBE sequences from Figure 6 are analyzed, not just the unmitigated.

An event sequence family (ESF) is defined in NEI-18-04 as “A grouping of event sequences with a common or similar plant operating states, IEs, hazard group, challenges to the plant safety functions, response of the plant in the performance of each safety function, response of each radionuclide transport barrier, and end state. An ESF may involve a single event sequence or several event sequences grouped together. Each release category may include one or more ESFs.

Each ESF involving a release is associated with one and only one release category.” As such, from Figure 6, it can be seen that 5 ESFs can be derived based on end state and potential for radiological or hazardous material release:

- Event sequences with success of either active, passive, or manual reactivity control FSF SSCs, and active heat removal FSF SSCs. Confinement barrier (Fuel/Cladding/PCB) structural integrity maintained, and no radiological release occurs (Figure 6, ES-1, 5, 9).
- Event sequences with success of either active, passive, or manual reactivity control FSF SSCs, but with failure of active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs to within limits for all confinement barriers (Fuel/Cladding/PCB), barrier structural integrity maintained, and no radiological release occurs (ES-2, 6, 10).
- Event sequences with success of either active, passive, or manual reactivity control FSF SSCs, but with failure of both active PGS and passive heat removal FSF SSCs. For conservatism, temperatures are postulated to exceed the limits for the fuel/cladding confinement barriers, however, PCB structural integrity maintained. Gaseous fission product release from fuel/cladding gas gap to the primary coolant is postulated to occur. It is also postulated that a minor release of gaseous fission products may occur though PCB failures. Such a release is qualitatively judged to be a small fraction of the bounding consequence results in ECAR-5127 (ES-3, 7, 11).
- Event sequences with success of either active, passive, or manual reactivity control FSF SSCs, but with failure of both active PGS and passive heat removal FSF SSCs. For conservatism, temperatures postulated to exceed limits for all confinement barriers, all confinement barrier (Fuel/Cladding/PCB) structural integrity lost, and radiological release (fission products released from fuel failure) postulated to occur. Such a release is bounded by the consequence results in ECAR-5127 (ES-4, 8, 12).
- Event sequence with failures of all reactivity control and heat removal FSF SSCs, all confinement barriers (Fuel/Cladding/PCB) structural integrity lost, and radiological release (all fission products released from Fuel/Cladding/PCB failure). Such a release is bounded by the consequence results in ECAR-5127 (ES-13).

Each internal or external SBE event sequence end state in Figure 6 where a radiological release occurs, based on the bounding consequence results in ECAR-5127, results in acceptable risk bins for protection of the public and collocated worker.

For internal events, immediate worker consequences are not evaluated since the TREAT building is unoccupied during reactor operations.

External SBEs where a radiological or non-radiological release occurs where a worker may be present, may result in risk in unacceptable risk bins for protection of the immediate worker. The risk is assumed unacceptable and design and/or operations controls as listed in Table 9 are required to reduce the risk to the immediate worker.

The fundamental conclusion, based on the MARVEL accident IE and scenario evaluation, is that MARVEL's overall design demonstrates compliance with the safety basis requirements in 10 CFR 830, that MARVEL can be operated safely, and that the public and workers are acceptably protected.

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-1: Shutdowns	ES-9	1. Operator manual scram, Test Scram. 2. The CDs passively rotate to shut down the reactor on manual scram. 3. Successful shutdown of the PGS Stirling engines and cooldown of the PCS. 4. Successful heat removal by the passive FSF SSCs. 5. Fuel/Cladding/PCB temperatures controlled to within criteria. 6. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • CIA rod • CIA gray rod • Upper confinement Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
	ES-10	1. Operator manual scram, Test Scram. 2. The CDs passively rotate to shut down the reactor on manual scram. 3. Unsuccessful shutdown of the PGS Stirling engines, but cooldown of the PCS prevents PCB failures. 4. Successful heat removal by the passive FSF SSCs. 5. Fuel/Cladding/PCB temperatures are controlled to within criteria. 6. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-11	1. Operator manual scram, Test Scram. 2. The CDs passively rotate to shut down the reactor on manual scram. 3. Unsuccessful shutdown of the PGS Stirling engines, but cooldown of the PCS prevents PCB failures. 4. Successful heat removal by the passive FSF SSCs. 5. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 6. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Gaseous fission products retained by primary coolant. Minor gaseous fission product release through PCB leak paths postulated. • Facility damage (Cladding and PCB) may preclude return to operation.	EU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-12	1. Operator manual scram, Test Scram. 2. The CDs passively rotate to shut down the reactor on manual scram. 3. Unsuccessful shutdown of the PGS Stirling engines, unsuccessful cooldown of the PCS, PCB failures. 4. Unsuccessful heat removal by the passive FSF SSCs. 5. Cladding/Fuel/PCB temperatures exceed criteria despite scram. 6. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Operator manual scram, Test Scram. 2. Manual scram failure to insert CDs. 3. Unsuccessful shutdown of the PGS Stirling engines, unsuccessful cooldown of the PCS, PCB failures. 4. Unsuccessful heat removal by the passive FSF SSCs. 5. Core temperatures elevate resulting in potential Cladding, Fuel, and PCB structural failure. 6. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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SBE-2: General Transient	ES-1, 5, 9	1. Minor core blockage, CD misposition, spurious trip, small leaks, internal fire or flood, Stirling engine vibration. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Successful heat removal by the active PGS FSF SSCs. 4. Fuel/Cladding/PCB temperatures controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • Upper confinement • Stirling engine automatic stop system Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation power • Backup power Controls: Controlled PCS cooldown following scram.
	ES-2, 6, 10	1. Minor core blockage, CD misposition, spurious trip, small leaks, facility fire or flood, Stirling engine vibration 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs. 4. Fuel/Cladding/PCB temperatures are controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-3, 7, 11	1. Minor core blockage, CD misposition, spurious trip, small leaks, facility fire or flood, Stirling engine vibration 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	EU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Minor core blockage, CD misposition, spurious trip, small leaks, facility fire or flood, Stirling engine vibration 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failures of all heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Cladding, Fuel, and PCB structural failure. 5. Fuel/Cladding/PCB structural integrity lost (Confinement FSF not met).	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Minor core blockage, CD misposition, spurious trip, small leaks, facility fire or flood, Stirling engine vibration. 2. Failures of all reactivity control FSF SSCs. 3. Failures of all heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Cladding, Fuel, and PCB structural failure. 5. Fuel/Cladding/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-3: Loss of Heat Sink (LOHS-1)	ES-1, 5, 9	1. Loss of single PGS Stirling engine or IHX, PGS water line break, or small break in SCS piping results in loss of active PGS heat removal. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Heat removal through the 3 remaining active PGS Stirling Engines are assumed to control core temperature to within limits. The geometry of the core remains coolable. 4. Fuel/Cladding/PCB temperatures controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA Gray Rod • Upper Confinement Non-Safety Related SSCs: • PGS Stirling Engines (3 remaining) • Post-accident monitoring • Instrumentation Power • Stirling Engine Automatic Stop System Controls: • Controlled PCS cooldown following scram.
	ES-2, 6, 10	1. Loss of single PGS Stirling engine or IHX, PGS water line break, or small break in SCS piping results in loss of active PGS heat removal. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of all active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs. 4. Fuel/Cladding/PCB temperatures are controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-3, 7, 11	1. Loss of single PGS Stirling engine or IHX, PGS water line break, or small break in SCS piping results in loss of active PGS heat removal. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	EU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Loss of single PGS Stirling engine or IHX, PGS water line break, or small break in SCS piping results in loss of active PGS heat removal. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Loss of single PGS Stirling engine or IHX, PGS water line break, or small break in SCS piping results in loss of active PGS heat removal. 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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SBE-4: LOHS-2	ES-2, 6, 10	1. Loss of multiple PGS Stirling Engines, SCS SSC failures, IHX failures, freezing of secondary coolant. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Heat removal through the active PGS is assumed to fail (LOHS), therefore, passive heat removal through the core is assumed to control core temperature to within limits. The geometry of the core remains coolable. 4. Fuel/Cladding/PCB temperatures controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • IHX liner/flange • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • Upper confinement Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
	ES-3, 7, 11	1. Loss of multiple PGS Stirling Engines, SCS SSC failures, IHX failures, freezing of secondary coolant. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Loss of multiple PGS Stirling Engines, SCS SSC failures, IHX failures, freezing of secondary coolant. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Loss of multiple PGS Stirling Engines, SCS SSC failures, IHX failures, freezing of secondary coolant. 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-5: LOHS-3	ES-1, 5, 9	1. DHR system clogging due to failure of the pit shielding structures, loose parts or debris, facility fires, internal flooding of TREAT pit, or external NPH. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Heat removal through the active PGS is assumed to control core temperature to within limits. Passive residual heat removal through the core is assumed unavailable (LOHS). The geometry of the core remains coolable. 4. Fuel/Cladding/PCB temperatures controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA Gray Rod • Upper Confinement Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
	ES-3, 7, 11	1. DHR system clogging due to debris, from failure of the pit shielding structures, loose parts or debris, facility fires, internal flooding of TREAT pit, or external NPH. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. DHR system clogging due to debris, from failure of the pit shielding structures, loose parts or debris, facility fires, internal flooding of TREAT pit, or external NPH. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Cladding, Fuel, and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. DHR system clogging due to debris, from failure of the pit shielding structures, loose parts or debris, facility fires, internal flooding of TREAT pit, or external NPH. 2. Failures of all reactivity control FSF SSCs 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Cladding, Fuel, and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-6: Loss of Flow (LOF)	ES-2, 6, 10	1. Core blockage (partial or total) as a result of leakage of secondary coolant into the PCS due to corrosion induced failure of the IHX boundary (IHX wall and liner). 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Heat removal through the active PGS is assumed to fail (LOF), therefore, passive residual heat removal through the core is assumed to control core temperature to within limits. The geometry of the core remains coolable. 4. Fuel/Cladding/PCB temperatures controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	EU	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • IHX liner/flange • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • Upper confinement Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
	ES-3, 7, 11	1. Core blockage (partial or total) as a result of leakage of secondary coolant into the PCS due to corrosion induced failure of the IHX boundary (IHX wall and liner). 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Core blockage (partial or total) as a result of leakage of secondary coolant into the PCS due to corrosion induced failure of the IHX boundary (IHX wall and liner). 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Cladding, Fuel, and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Core blockage (partial or total) as a result of leakage of secondary coolant into the PCS due to corrosion induced failure of the IHX boundary (IHX wall and liner). 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-7: Loss of Power (LOP)	ES-1, 5, 9	1. Grid, facility or weather-related loss of offsite power (LOOP). 2. The non-safety related trip system automatically activates to passively rotate CDs to shut down the reactor automatically on LOP . 3. Successful heat removal by the active PGS FSF SSCs. 4. Fuel/Cladding/PCB temperatures controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (LOP Trip) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (Reactor Vessel, Upper Vessel Head, Distribution Block, Downcomers, and all PCB Penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA Gray Rod • Upper Confinement Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation Power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
	ES-2, 6, 10	1. Grid, facility or weather-related LOOP. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs. 4. Fuel/Cladding/PCB temperatures are controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-3, 7, 11	1. Grid, facility or weather-related LOOP. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	EU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Grid, facility or weather-related LOOP. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failures of all heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Cladding, Fuel, and PCB structural failure. 5. Fuel/Cladding/PCB structural integrity lost (Confinement FSF not met).	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Grid, facility or weather-related LOOP. 2. Failures of all reactivity control FSF SSCs. 3. Failures of all heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Cladding, Fuel, and PCB structural failure. 5. Fuel/Cladding/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-8: Transient Overpower (TOP-1)	ES-1, 5, 9	1. Spurious CD or CIA movement. 2. The safety-related CD relays prevent simultaneous uncontrolled withdrawal of more than one CD as a result of equipment or operator error. The safety related CD stops limit CD movement to ensure that available excess reactivity insertion does not challenge fuel and temperature limits when inserted instantaneously. 3. The non-safety related trip system activates to passively rotate CDs to shut down the reactor. 4. Successful heat removal by the active PGS to control core temperature to within limits. The geometry of the core remains coolable. 5. Fuel/Cladding/PCB temperatures controlled to with criteria. 6. No Fuel/Cladding/PCB structural damage.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip Relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • CIA interlocks relays and hard stops • Upper confinement
	ES-2, 6, 10	1. Spurious CD or CIA movement. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs. 4. Fuel/Cladding/PCB temperatures are controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	EU	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-3, 7, 11	1. Spurious CD or CIA movement. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Spurious CD or CIA movement. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
	ES-13	1. Spurious CD or CIA movement. 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	Non-Safety Related SSCs: • PGS • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
SBE-9: TOP-2	ES-1, 5, 9	1. Overcooling of the primary system by the power conversion unit, or NaK voiding. 2. The non-safety related trip system activates to passively rotate CDs to shut down the reactor. 3. Successful heat removal by the active PGS to control core temperature to within limits. The geometry of the core remains coolable. 4. Fuel/Cladding/PCB temperatures controlled to with criteria. 5. No Fuel/Cladding/PCB structural damage.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip Relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA Gray Rod
	ES-2, 6, 10	1. Overcooling of the primary system by the power conversion unit, or NaK voiding. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs. 4. Fuel/Cladding/PCB temperatures are controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-3, 7, 11	1. Overcooling of the primary system by the power conversion unit, or NaK voiding. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Overcooling of the primary system by the power conversion unit, or NaK voiding. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

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	ES-13	1. Overcooling of the primary system by the power conversion unit, or NaK voiding. 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	• CIA interlocks relays and hard stops • Upper Confinement Non-Safety Related SSCs: • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Controlled PCS cooldown following scram.
SBE-10: Loss of Coolant Accident (LOCA)-1	ES-2, 6, 10	1. Break of low-elevation components (downcomer, lower plenum) inside guard vessel, PCB penetration leaks/breaks/seal ruptures or support SSC failure. 2. PCS NaK leak inside guard vessel. 3. Guard vessel and initial cover gas pressures prevent core from being uncovered. 4. The non-safety related trip system activates to passively rotate CDs to shut down the reactor (Reactivity control FSF met). 5. The active PGS is assumed unavailable due to LOCA. Passive residual heat removal through the core is assumed to control core temperature to within limits. The geometry of the core remains coolable. 6. Fuel/cladding/PCB temperatures controlled to with criteria. 7. No fuel/cladding/PCB structural damage.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	EU	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip Relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding)

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	ES-3, 7, 11	1. Break of low-elevation components (downcomer, lower plenum) inside guard vessel, PCB penetration leaks/breaks/seal ruptures or support SSC failure. 2. PCS leak inside guard vessel. 3. Guard vessel and/or cover gas pressure fails to prevent core from being partially uncovered. 4. Success of either active, passive, or manual reactivity control FSF SSCs. 5. Failure of both active PGS and passive heat removal FSF SSCs. 6. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 7. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	• PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • CIA interlocks relays and hard stops • Upper Confinement Non-Safety Related SSCs: • Post-accident monitoring (pressure monitoring) • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Corrosion control. • Initial primary and GVS cover gas pressures. • Controlled PCS cooldown following scram.
	ES-4, 8, 12	1. Break of low-elevation components (downcomer, lower plenum) inside guard vessel, PCB penetration leaks/breaks/seal ruptures or support SSC failure. 2. PCS leak inside guard vessel. 3. Guard vessel and/or cover gas pressure fails to prevent core from being fully uncovered. 4. Success of either active, passive, or manual reactivity control FSF SSCs. 5. Failure of both active PGS and passive heat removal FSF SSCs. 6. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 7. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Break of low-elevation components (downcomer, lower plenum) inside guard vessel, PCB penetration leaks/breaks/seal ruptures or support SSC failure. 2. PCS leak inside guard vessel. 3. Guard vessel and/or cover gas pressure fails to prevent core from being fully uncovered. 4. Failures of all reactivity control FSF SSCs. 5. Failure of both active PGS and passive heat removal FSF SSCs. 6. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 7. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
SBE-11: LOCA-2	ES-2, 6, 10	1. Failure of the IHX wall between the NaK primary coolant and the secondary coolant due to corrosion. 2. PCS-IHX leak into upper confinement prevented by IHX liner/flange. Core remains covered. 3. Success of either active, passive, or manual reactivity control FSF SSCs. 4. The active PGS is assumed unavailable due to IHX failure. 5. Passive residual heat removal through the core is assumed to control core temperature to within limits. The geometry of the core remains coolable. 6. Fuel/Cladding/PCB temperatures controlled to with criteria. 7. No Fuel/Cladding/PCB structural damage.	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	EU	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Trip Relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops)

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
	ES-3, 7, 11	1. Failure of the IHX wall between the NaK primary coolant and the secondary coolant due to corrosion. 2. IHX liner/flange fails to prevent limited PCS-IHX leak into upper confinement. Core partially covered. 3. Success of either active, passive, or manual reactivity control FSF SSCs. 4. Failure of both active PGS and passive heat removal FSF SSCs. 5. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 6. Fuel/Cladding structural integrity lost. PCB structural integrity maintained. 7. Possible combined activated PCS gaseous fission product and fire/smoke through upper confinement ventilation system.	• Minor gaseous fission product release possible through upper confinement leak paths. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	• IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA Gray Rod • CIA interlocks relays and hard stops • Upper Confinement • IHX liner/flange Non-Safety Related SSCs: • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Corrosion control. • Controlled PCS cooldown following scram.
	ES-4, 8, 12	1. Failure of the IHX wall between the NaK primary coolant and the secondary coolant due to corrosion. 2. IHX liner/flange fails to prevent PCS-IHX leak into upper confinement. Core fully uncovered. 3. Success of either active, passive, or manual reactivity control FSF SSCs. 4. Failure of both active PGS and passive heat removal FSF SSCs. 5. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 6. Cladding/Fuel/PCB structural integrity lost. 7. Possible combined activated PCS fission product and fire/smoke through upper confinement ventilation system.	• Fission product release through failed barriers. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Failure of the IHX wall between the NaK primary coolant and the secondary coolant due to corrosion. 2. IHX liner/flange fails to prevent PCS-IHX leak into upper confinement. Core fully uncovered. 3. Failures of all reactivity control FSF SSCs. 4. Failure of both active PGS and passive heat removal FSF SSCs. 5. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 6. Cladding/Fuel/PCB structural integrity lost. 7. Possible combined activated PCS fission product and fire/smoke through upper confinement ventilation system.	• Fission product release through failed barriers. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-12: LOCA-3	ES-2, 6, 10	1. Weld failure in the top plate of the distribution plenum. 2. PCS NaK leak into upper confinement in the form of vapor and/or droplets. Limited leakage of NaK into upper confinement. Core remains covered. 3. Success of either active, passive, or manual reactivity control FSF SSCs. 4. Active PGS is assumed unavailable due to IHX failure. 5. Passive residual heat removal through the core is assumed to control core temperature to within limits. Geometry of the core remains coolable. 6. Fuel/cladding/PCB temperatures controlled to with criteria. 7. No fuel/cladding/PCB structural damage. 8. Nak (assumed activated) leakage into upper confinement may result in interaction with air causing fire/smoke and release through upper confinement ventilation system.	• Minor gaseous radiological or non-radiological release through upper confinement. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • The facility should be capable of returning to operation without extensive corrective action or repair.	EU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	Safety Related SSCs: • RPS (trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • CIA interlocks relays and hard stops • Upper confinement Non-Safety Related SSCs: • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Controls: • Corrosion control • Controlled PCS cooldown following scram.
	ES-3, 7, 11	1. Weld failure in the top plate of the distribution plenum. 2. Limited PCS NaK leak into upper confinement. Core partially uncovered. 3. Success of either active, passive, or manual reactivity control FSF SSCs. 4. Failure of both active PGS and passive heat removal FSF SSCs. 5. Fuel/cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 6. Fuel/cladding structural integrity lost. PCB structural integrity maintained. 7. Possible combined activated PCS gaseous fission product and fire/smoke through upper confinement ventilation system.	• Minor gaseous fission product release possible through upper confinement leak paths. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • Facility damage (cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Weld failure in the top plate of the distribution plenum. 2. PCS NaK leak into upper confinement. Core uncovered. 3. Success of either active, passive, or manual reactivity control FSF SSCs. 4. Failure of both active PGS and passive heat removal FSF SSCs. 5. Core temperatures elevate despite scram resulting in potential fuel/cladding and PCB structural failure. 6. Cladding/fuel/PCB structural integrity lost. 7. Possible combined activated PCS fission product and fire/smoke through upper confinement ventilation system.	• Fission product release through failed barriers. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (fuel/cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Weld failure in the top plate of the distribution plenum. 2. PCS NaK leak into upper confinement. Core fully uncovered. 3. Failures of all reactivity control FSF SSCs. 4. Failure of both active PGS and passive heat removal FSF SSCs. 5. Core temperatures elevate resulting in potential fuel/cladding and PCB structural failure. 6. Cladding/fuel/PCB structural integrity lost. 7. Possible combined activated PCS fission product and fire/smoke through upper confinement ventilation system.	• Fission product release through failed barriers. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (fuel/cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-13: Seismic Event (g ≤SSE)	ES-1, 5, 9	1. Seismic Event (g ≤ SSE). 2. Safety related seismic trip system activates CDs to passively rotate to shut down the reactor. 3. Successful heat removal by the active PGS is assumed to control core temperature to within limits. The geometry of the core remains coolable. 4. Fuel/Cladding/PCB temperatures controlled to with criteria. 5. No Fuel/Cladding/PCB structural damage .	• No radiological or non-radiological release. • The facility should be capable of returning to operation without extensive corrective action or repair.	U	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: • RPS (Seismic Trip) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, cage, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops) • IRF (fuel, core and internals, reactor support structures) • Primary NaK circulation flowpath and core coolable geometry (fuel, core and internals, barrel, reactor support structures) • Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Fission product barriers (fuel matrix and cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Non-Safety Related with Augmented Requirements SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA Gray Rod • CIA interlocks relays and hard stops • Upper Confinement Non-Safety Related SSCs: • Post-accident monitoring • Instrumentation Power • Backup Power • Stirling Engine Automatic Stop System Controls: • Controlled PCS cooldown following scram.
	ES-2, 6, 10	1. Seismic Event (g ≤ SSE). 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of active PGS heat removal FSF SSCs. However, temperatures are controlled by passive heat removal FSF SSCs. 4. Fuel/Cladding/PCB temperatures are controlled to within criteria. 5. Fuel/Cladding/PCB structural integrity maintained.	• No radiological or non-radiological release. • Facility should be capable of returning to operation following corrective action or repair of damage.	EU	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	
	ES-3, 7, 11	1. Seismic Event (g ≤SSE). 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 5. Fuel/Cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release possible through PCB leak paths. • Facility damage (Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: L Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-4, 8, 12	1. Seismic Event (g ≤ SSE). 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate despite scram resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Seismic Event (g ≤ SSE). 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. Core temperatures elevate resulting in potential Fuel/Cladding and PCB structural failure. 5. Cladding/Fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Figure 6 Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Event Sequence Likelihood[d]	Consequence[e, f]	Risk Bin[g]	Candidate Safety SSCs or Controls (SDS-119)[h]
SBE-14: Seismic Event (g > SSE)	ES-13	1. Seismic Event (g > SSE). 2. SR seismic trip system fails to activate CDs to passively rotate to shut down the reactor. 3. Core damage occurs due to seismic event > design basis of reactor core, internals, and structure, and TREAT reactor building structures and pit. 4. Core rearrangement leads to energetic reactivity insertion. 5. Total disassembly of core. 6. Cladding/PCB/fuel structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (Fuel/Cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	Controls: • Emergency management procedures

Notes:

a. See Table 5.
b. FSFs Figure 5.
c. See Table 6.
d. IE Qualitative Likelihood from Table 5.
e. Qualitative Consequence from Table 7.
f. For internal events, immediate worker consequence N/A since TREAT building unoccupied during reactor operations, and workers relocated to TREAT control room and evaluated as collocated workers at 770m.
g. Risk Bins from Figure 7, Figure 8, and Figure 9.
h. Safety SSCs or controls needed to control risk.

Table 9. MARVEL external SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Likelihood[d]	Consequence[e]	Risk Bin[f]	Candidate Safety SSCs or Controls[g]
SBE-15: Radiological or Hazardous Material Release	ES-1	1. Cover Gas System Breach (Cover Gas Line) during operations. 2. No prior activation. 3. Leak Isolated (Confinement FSF Met).	• No radioactive material release. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Public: N/A Collocated Worker: N/A Facility Worker: N/A	Safety Related SSCs: None Non-Safety Related with Augmented Requirements SSCs: • Cover Gas System, Lines, Leak Isolation • Radiological Monitoring Controls: • Radiological Protection Program • Industrial Hygiene Program • Emergency Management
	ES-2	1. Cover Gas System Breach (Cover Gas Line) during operations. 2. Prior activation. 3. Leak isolated (Confinement FSF met).	• Minor radioactive material release. • The facility should be capable of returning to operation without extensive corrective action or repair.	U	Public: L Collocated Worker: L Facility Worker: L	Public: III Collocated Worker: III Facility Worker: III	
	ES-3	1. Cover Gas System Breach (Cover Gas Line) during operations. 2. Prior activation. 3. Leak NOT isolated (Confinement FSF NOT Met).	• Airborne radioactive material release . • The facility should be capable of returning to operation without extensive corrective action or repair.	EU	Public: L Collocated Worker: L Facility Worker: M	Public: IV Collocated Worker: IV Facility Worker: III	
SBE-16: Radiological or Hazardous Material Release, or Direct Radiation Exposure	ES-1	1. Heavy Load Drop-Crane failure or human error results in drop/Impact of used fuel into core during unloading operations. 2. Fuel/Cladding intact prior to and after drop (Confinement FSF Met). 3. Vessel internals (Be reflectors) intact prior to and after drop (Confinement FSF Met).	• No release of airborne radionuclides or hazardous materials. • Possible direct radiation exposure (Direct Radiation Exposure FSF not met). • The facility should be capable of operation without extensive corrective action or repair.	A	Public: N/A Collocated Worker: N/A Facility Worker: L	Public: N/A Collocated Worker: N/A Facility Worker: III	Safety Related SSCs: None Non-Safety Related with Augmented Requirements SSCs: • TREAT Cranes and Chain Hoists/Falls • Fuel handling hardware • Shielding Structure • Radiological Monitoring Controls: • Radiological Protection Program • Industrial Hygiene Program • Emergency Management • Hoisting and Rigging • Crane restrictions
	ES-2	1. Heavy Load Drop- Crane failure or human error results in drop/Impact of used fuel into core during unloading operations. 2. Cladding damage, fuel intact after drop (Confinement FSF not Met). 3. Vessel internals (Be Reflectors) damaged due to drop (Confinement FSF not Met)	• Release of fission product gasses. • Possible direct radiation exposure (Direct Radiation Exposure FSF not met). • Low release possible of hazardous material (Be). • Facility should be capable of returning to operation following corrective action or repair of damage.	U	Public: L Collocated Worker: L **Facility Worker: M**	Public: III Collocated Worker: III **Facility Worker: II**	
	ES-3	1. Heavy Load Drop-Crane failure or human error results in drop/Impact of used fuel into core during unloading operations. 2. Cladding damage/Fuel damage(Confinement FSF not Met). 3. Vessel internals damaged (Confinement FSF not Met.	• Release of airborne radioactive material. • Possible direct radiation exposure (Direct Radiation Exposure FSF not met). • Low release possible of hazardous material (Be). • Facility should be capable of returning to operation following corrective action or repair of damage.	EU	Public: L Collocated Worker: L **Facility Worker: H**	Public: III Collocated Worker: III **Facility Worker: II**	

Table 9. MARVEL external SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Likelihood[d]	Consequence[e]	Risk Bin[f]	Candidate Safety SSCs or Controls[g]
SBE-17: NaK Spill and Fire	ES-1	1. Failure or leak in NaK unloading system results in NaK Spill and Fire. 2. Prior NaK Contamination from Reactor Operations. 3. Insufficient conditions exist for a NaK fire.	• No release of radionuclides. Worker exposure to aerosolized NaK possible • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • The facility should be capable of returning to operation without extensive corrective action or repair.	U	Public: N/A Collocated Worker: N/A Facility Worker: L	Public: N/A Collocated Worker: N/A Facility Worker: III	Safety Related SSCs: None Non-Safety Related with Augmented Requirements SSCs: • NaK system design • Shielding Structure/Fire barrier • NaK fire and smoke detection and communication • NaK fire extinguisher • Radiological Monitoring Controls: • Radiological Protection Program • Industrial Hygiene Program • Emergency Management
	ES-2	1. Failure or leak in NaK unloading system results in NaK Spill and Fire. 2. Prior NaK Contamination from Reactor Operations. 3. Sufficient conditions exist for fire to occur. 4. Fire detection, communication succeed, suppression succeed.	• Release of radionuclides and aerosolized NaK due to spill and fire prior to successful suppression. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • Facility damage (fire damage, facility contamination) may preclude return to operation.	EU	Public: L Collocated Worker: L Facility Worker: M	Public: IV Collocated Worker: IV Facility Worker: III	
	ES-3	1. Failure or leak in NaK unloading system results in NaK Spill and Fire. 2. Prior NaK Contamination from Reactor Operations. 3. Sufficient conditions exist for fire to occur. Fire detection, communication, and suppression fail.	• Moderate release or radionuclides and aerosolized NaK due to spill and fire. • Fire/smoke and generation of potassium and sodium oxide, potassium and possibly sodium superoxide, potassium and sodium hydroxide, and hydrogen generation. • Facility damage (fire damage, facility contamination) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: H	Public: IV Collocated Worker: IV Facility Worker: III	

Table 9. MARVEL external SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Event Sequence	Accident Progression Summary (IE and key FSF responses)[a, b]	End State[c]	Likelihood[d]	Consequence[e]	Risk Bin[f]	Candidate Safety SSCs or Controls[g]
SBE-18: Radiological or Hazardous Material Release	ES-2, 6, 10	1. Stirling engine helium tube rupture leads to high energy gas release that would cause secondary coolant release to upper confinement. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. The active PGS is assumed unavailable due to Stirling failure. 4. SCS failure and limited leakage of activated secondary coolant into upper confinement. 5. Passive residual heat removal through the core is assumed to control core temperature to within limits. The geometry of the core remains coolable. 6. Fuel/cladding/PCB temperatures controlled to within criteria. 7. No fuel/cladding/PCB structural damage.	• Minor release of radionuclides. Minor release of secondary coolant possible. • The facility should be capable of returning to operation without extensive corrective action or repair.	A	Public: L Collocated Worker: L Facility Worker: N/A	Public: III Collocated Worker: III Facility Worker: N/A	Safety Related SSCs: None Non-Safety Related with Augmented Requirements SSCs: • Bellows • IHX liner/flange Controls: • Corrosion control • Radiological Protection Program • Industrial Hygiene Program • Emergency Management
	ES-3, 7, 11	1. Stirling engine helium tube rupture leads to high energy gas release that would cause secondary coolant release to upper confinement. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. SCS failure and limited leakage of activated secondary coolant into upper confinement. 5. Fuel/Cladding temperatures exceed criteria despite scram. However, PCB temperatures remain within criteria. 6. Fuel/cladding structural integrity lost. PCB structural integrity maintained.	• Fission products retained by primary coolant. Minor gaseous fission product release is possible through SCS leak paths. • Facility damage (cladding and PCB) may preclude return to operation.	U	Public: L Collocated Worker: L Facility Worker: N/A	Public: III Collocated Worker: III Facility Worker: N/A	
	ES-4, 8, 12	1. Stirling engine helium tube rupture leads to high energy gas release that would cause secondary coolant release to upper confinement. 2. Success of either active, passive, or manual reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. SCS failure and limited leakage of activated secondary coolant into upper confinement. 5. Core temperatures elevate despite scram resulting in potential fuel/cladding and PCB structural failure. 6. Cladding/fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (fuel/cladding and PCB) may preclude return to operation.	EU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	
	ES-13	1. Stirling engine helium tube rupture leads to high energy gas release that would cause secondary coolant release to upper confinement. 2. Failures of all reactivity control FSF SSCs. 3. Failure of both active PGS and passive heat removal FSF SSCs. 4. SCS failure and limited leakage of activated secondary coolant into upper confinement. 5. Core temperatures elevate resulting in potential fuel/cladding and PCB structural failure. 6. Cladding/fuel/PCB structural integrity lost.	• Fission product release through failed barriers. • An assessment of the design capability with respect to the accident prevention and mitigation strategy to meet EGs is required. • Facility damage (fuel/cladding and PCB) may preclude return to operation.	BEU	Public: L Collocated Worker: M Facility Worker: N/A	Public: IV Collocated Worker: IV Facility Worker: N/A	

Notes:

a. See Table 5.
b. FSFs Figure 5.
c. See Table 6.
d. Qualitative Likelihood from Table 5.
e. Qualitative Consequence from Table 7.
f. Risk Bins from Figure 7, Figure 8, and Figure 9.
g. Safety SSCs or controls needed to control risk.

2.6 Selection of MARVEL Safety SSCs

Preliminary safety SSCs identified SDS-119 and in the Table 8 and Table 9 SBE sequence analyses are further evaluated for SSC classifications and identification of required SSC safety functions as a result of the SBE analyses. From the analysis in Figure 6 and Table 10, ES-3 and ES-4, ES-7 and ES-8, ES-11, ES-12 and ES-13 demonstrate that reactivity control is insufficient to prevent a radiological release. To eliminate the sequence (end state OK), active or passive heat removal is required. ES-6 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 listed in Table 8 are 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 listed in Table 8 are selected as NSR-SSCs. The DBA analysis that will be performed in the MARVEL SAR-420 Addendum Chapter 15, Accident Analysis, will demonstrate these conclusions, as the DBAs credit only safety-related SSCs, and will identify the specific performance requirements for those passive FSF SSCs for inclusion in the MARVEL Technical Specifications document.

As discussed in SDS-119, MARVEL SSCs are 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 11 below. Table 12 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 Table 7. 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 8 may be designated as SR based on criteria 1, 3, 4 in Table 11.

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 11 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 EGs (SR-2 criterion met), all SR SSCs are equivalent to SC SSCs, and NSR-AR-1 SSCs are equivalent to SS SSCs. 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. NSR-AR-2 SSCs are SS.

If potential accidents could result in onsite consequences greater than EGs (NSR-AR2 criterion met), all NSR-AR2 SSCs are equivalent to SS SSCs. If potential accidents do not result in onsite consequences greater than EGs (NSR-AR2 criterion NOT met), all NSR-AR-2 SSCs are non-safety.

All NSR-AR-3 SSCs are non-safety.

Table 10. Credited SSCs from event sequence analysis.

Event Sequence	Reactivity Control			Heat Removal		Confinement			End State
	CD/Seismic Trip	IRF	Manual Scram	PGS	Passive	Cladding	Fuel	PCB	
ES-1	X			X		X	X	X	OK
ES-2	X				X	X	X	X	OK
ES-3	X							X	Fission Gas Release
ES-4	X								Fission Product Release
ES-5		X		X		X	X	X	OK
ES-6		**X**			**X**	**X**	**X**	**X**	**OK**
ES-7		X						X	Fission Gas Release
ES-8		X							Fission Product Release
ES-9			X	X		X	X	X	OK
ES-10			X		X	X	X	X	OK
ES-11			X					X	Fission Gas Release
ES-12			X						Fission Product Release
ES-13									Fission Product Release

Table 11. 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 offsite consequences greater than the evaluation guidelines?	Exceed offsite 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

Table 12. MARVEL SSC classification summary.

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
Fuel and Core System (FCS)	Fuel System (FS)	Fuel/Cladding/Fuel Assembly	SR (SR1, 4)	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 PCB temperature limits are challenged, or before core damage occurs during anticipated events and postulated accident conditions.
				SS	Heat Removal - Passive heat rejection	- Maintain heat rejection geometry and features and natural circulation ability during all normal operations and accident conditions.
				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.
				SS	Confinement of Radioactive and Hazardous Material Release - Fission product barriers including fuel matrix and Cladding	- 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.
		Radial Be Core Reflector (metal) Inserts	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. - 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.
		Neutron Source	NSR	Non-Safety	N/A	- N/A
	Core Structures System (CSS)	Lower 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.

Table 12. MARVEL SSC classification summary.

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 an 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 12. MARVEL SSC classification summary.

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 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.

Table 12. MARVEL SSC classification summary.

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
			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.
	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 12. MARVEL SSC classification summary.

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
		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.
		CD Torsion Spring	SR (SR1)	SS	Reactivity Control – CD Insertion	- Provide stored energy for drum shutdown (scram) rotation.
		CD Rotary Damper	SR (SR4)	SS	Reactivity Control – CD Insertion	- Reduce impact to hard stops.
		CIA Motor, Motor Controller, Motor Resolver, Motor Gear	NSR-AR (NSR-AR1)	Non-safety	N/A	- N/A
		Ball Screws & Nuts	NSR-AR (NSR-AR3)	Non-safety	N/A	- N/A
		CIA Electromagnet	NSR-AR (NSR-AR3)	Non-safety	N/A	- N/A
	Drum Structures System (DSS)	Control Drum Cage & Rails, Cage Platforms	SR (SR1)	SS	Reactivity Control – CD Insertion	- Structural support of drum motors, switches. - Ensure CD insertion under normal operating and transient conditions to ensure reactor shutdown.
		Control Drum Shaft and Bearings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Connect control drum to drive system. - Ensure CD insertion under normal operating and transient conditions to ensure reactor shutdown.
		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.
		Control Drum (Rotary) Seal & Standoff	SR (SR1)	SS	Reactivity Control – CD Insertion	- Limit leakage of argon (from Guard Vessel) - Minimize seal friction.
		Control Drum Lock	SR (SR1)	SS	Reactivity Control – CD Insertion	- Provide a physical lock of drum in shutdown position.
		Axial Expansion Springs	SR (SR1)	SS	Reactivity Control – CD Insertion	- Accommodate axial expansion of drums. Keep individual BeO plates compressed.
		Couplings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Accommodate misalignment between drive and drum shaft.
		Upper Alignment Bearings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Align control drums, Allow rotary motion, Minimize friction.
		Lower Support Bearings	SR (SR1)	SS	Reactivity Control – CD Insertion	- Support control drums, Align control drums, Allow rotary motion, Minimize friction.
		CIA Cage Standoffs, Rails, Platforms	NSR-AR (NSR-AR3)	Non-safety	N/A	- N/A
	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.
		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.
		CIA Rod (B_4C) and Drive Shaft	NSR-AR (NSR-AR3)	Non-safety	N/A	- N/A
		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. - Provide negative reactivity for excess reactivity management installed at the beginning of life and stays installed stationary in the reactor throughout the reactor life.
	Drum Position Measurement System (DPMS)	Control Drum Position Indicator & Gear	NSR-AR (NSR-AR3)	Non-safety	N/A	- N/A
		Control Drum In Limit Switch, Out Limit Switch	NSR-AR (NSR-AR3)	SS	Reactivity Control – CD Insertion	- N/A
		CIA Position Indicator, In Limit Switch, Out Limit Switch	NSR-AR (NSR-AR3)	Non-safety	N/A	- N/A

Table 12. MARVEL SSC classification summary.

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
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 offsite 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 12. MARVEL SSC classification summary.

System	Subsystem	Major Components	MARVEL Safety Designation (Criterion)	DOE Safety SSC Classification	Safety Functions	Functional Requirements
Power Generation System (PGS)	Electrical Production System (EPS)	QB80 Engine	NSR	Non-Safety	N/A	- N/A
		Water Line Connection and Pipes	NSR	Non-Safety	N/A	- N/A
		Qenergy Engine Control Units (ECUs)	NSR	Non-Safety	N/A	- N/A
		Qenergy Computer/HMI	NSR	Non-Safety	N/A	- N/A
	Engine Cooling System (ECS)	Stirling Engine Automatic Stop System	NSR	Non-Safety	N/A	- N/A
		Compact Heat Exchangers	NSR	Non-Safety	N/A	- N/A
		Water Piping/tubing	NSR	Non-Safety	N/A	- N/A
		Glycol Piping/tubing	NSR	Non-Safety	N/A	- N/A
		Heat Rejection Units (HRUs)	NSR	Non-Safety	N/A	- N/A
		Flow/Temp Sensor	NSR	Non-Safety	N/A	- N/A
		Resistance Temperature Detector, Flow Meter	NSR	Non-Safety	N/A	- N/A
		Pumps	NSR	Non-Safety	N/A	- N/A
		Fill Tanks (water and glycol)	NSR	Non-Safety	N/A	- N/A
		HRU fan	NSR	Non-Safety	N/A	- N/A
		Check Valve	NSR	Non-Safety	N/A	- N/A
		Engine Stall Circuit	NSR	Non-Safety	N/A	- N/A
		Pressure relief valve	NSR	Non-Safety	N/A	- N/A
		Humidity sensor	NSR	Non-Safety	N/A	- N/A
		Accelerometer	NSR	Non-Safety	N/A	- N/A
		Drain	NSR	Non-Safety	N/A	- N/A

2.7 Selection of Design Basis Accidents

MARVEL DBAs are postulated event sequences that are used to set design criteria and performance criteria for the design of SR-SSCs identified in Table 12. The Table 13 DBAs were derived from the SBEs identified in the event sequence analysis in Table 8 for internal events, and Table 9 for 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 Table 12, are assumed to be available in each DBA scenario.

Based on the safety SSCs identified in Table 12 the NSR-AR RPS is therefore assumed to be unavailable to initiate the insertion of the CDs to perform the reactivity control function (unprotected). Reactivity control is provided exclusively by passive IRF. The NSR PGS is also assumed to be unavailable and core heat removal is provided exclusively by passive conduction/convection to the ambient air.

Table 13 provides a summary description of the limiting SBE scenario from Table 8 or Table 9 assuming all applicable SSCs are available to mitigate the accident sequence (Column 2), available SR-SSCs only (Column 3), and DBA sequence of events assuming only SR-SSCs are available to mitigate the accident sequence (Column 4).

Note that SBE-1 and SBE-2 are considered as anticipated events in Table 8 and as such are analyzed as “Shutdown” events. SBE-3 is considered a “load-following” event.

The performance criteria for SR-SSCs will be identified in the MARVEL SAR-420 Addendum Chapter 15 based on the results of the DBA analyses.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-1: Transient Overpower (TOP) [SBE-8, -9]	Initiating Event (IE): Spurious CD movement results in a positive reactivity insertion and resulting increase in core power. Reactivity Control: The SR CD relays prevent simultaneous uncontrolled withdrawal of more than one CD as a result of equipment or operator error. The SR CD stops limit CD movement to ensure that available excess reactivity insertion does not challenge fuel and temperature limits when inserted instantaneously. The NSR-AR trip system activates the remaining 3 CDs to shut down the reactor (Reactivity control FSF met). Heat Removal: Successful heat removal by the NSR PGS to control core temperature to within limits. The geometry of the core remains coolable. Fuel/Cladding/PCB temperatures controlled to within criteria (Heat removal FSF met). Confinement: No Fuel/Cladding/PCB structural damage (Confinement FSF met). End State: No radiological release.	1. CD Relays/Stops 2. IRF 3. Primary NaK circulation flowpath and core coolable geometry 4. Passive heat rejection 5. Fission product barriers including fuel matrix and cladding 6. PCB including reactor barrel 7. GVS 8. Manual scram 9. CD insertion capability	1. Uncontrolled rotation of one CD to the mechanical stop position resulting in insertion of the total excess reactivity at the highest possible rate. 2. Power increases in response to the positive reactivity insertion, which leads to higher temperatures in the core. 3. The NSR-AR RPS and insertion of CDs is assumed unavailable. 4. Reactor stabilizes via SR IRF (Reactivity control FSF met). 5. The NSR active PGS is assumed unavailable for the heat removal FSF. 6. Successful SR passive decay heat removal (Heat removal FSF met). 7. Fuel/Cladding/PCB temperatures controlled to within criteria. 8. SR confinement barriers remain intact (Confinement FSF met). 9. No radiological release. 10. Ultimate reactor shutdown is by manual scram.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-2: Loss of Heat Sink (LOHS) [SBE-3, -4, -5]	IE: Loss of Stirling engines (multiple) results in loss of the active PGS and decrease in heat removal from the core. Reactivity Control: The NSR-AR trip system activates the CDs to shut down the reactor (Reactivity control FSF met). Heat removal: Core temperature is controlled to within limits by SR passive heat removal measures (Decay heat removal FSF met). Confinement: No Fuel/Cladding barrier or PCB structural damage (Confinement FSF met). End State: No radiological release.	1. IRF 2. Primary NaK circulation flowpath and core coolable geometry 3. Passive heat rejection 4. Fission product barriers including fuel matrix and cladding 5. PCB including reactor barrel 6. Manual scram 7. GVS 8. CD insertion capability	1. Loss of NSR active heat removal from the reactor core to the ultimate heat sink (loss of all four Stirling engines). 2. The NSR-AR RPS and insertion of CDs is not available. 3. Reactor stabilizes via SR IRF (Reactivity control FSF met). 4. Successful SR passive decay heat removal (Decay heat removal FSF met). 5. Fuel/Cladding/PCB temperatures controlled to within criteria. 6. SR confinement barriers remain intact (Confinement FSF met). 7. No radiological release. 8. Ultimate reactor shutdown is by manual scram.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-3: Loss of Flow (LOF) [SBE-6]	IE: Core blockage (partial or total) due to a failure in a SCS IHX and leakage of secondary coolant into PCS resulting in a flow reduction and loss of natural circulation. Reactivity control: The NSR-AR trip system activates the CDs to rotate to shut down the reactor (Reactivity control FSF met). Heat removal: Unsuccessful heat removal by the NSR PGS due to LOF. Core temperature is controlled to with limits by SR passive heat removal measures (Decay heat removal FSF met). Confinement: No Fuel/Cladding barrier or PCB structural damage (Confinement FSF met). End State: No radiological release.	1. IRF 2. Primary NaK circulation flowpath and core coolable geometry 3. IHX design 4. Passive heat rejection 5. Fission product barriers including fuel matrix and cladding 6. PCB including reactor barrel 7. GVS 8. Manual scram 9. CD insertion capability	1. Loss of Flow due to an IHX failure and leakage of secondary coolant into the PCS. 2. The NSR-AR RPS and insertion of CDs is not available. 3. Reactor stabilizes via SR IRF (Reactivity control FSF met). 4. Heat removal by the NSR active PGS is unavailable. 5. Successful SR passive decay heat removal (Decay heat removal FSF met). 6. Fuel/Cladding/PCB temperatures controlled to within criteria. 7. SR confinement barriers remain intact (Confinement FSF met). 8. No radiological release. 9. Reactor shutdown is by manual scram.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-4: Loss of Power (LOP) [SBE-7]	IE: Grid related Loss of Offsite Power (LOOP). Reactivity control: On loss of power, loss of energized coupling initiates a reactor shutdown by insertion of the SR CDs due to the potential energy of springs (Reactivity control FSF met). Heat removal: Successful heat removal by the NSR-AR PGS. Core temperature is controlled to with limits (Decay heat removal FSF met). Confinement: No Fuel/Cladding barrier or PCB structural damage (Confinement FSF met). End State: No radiological release.	1. CD insertion capability 2. IRF 3. Primary NaK circulation flowpath and core coolable geometry 4. Passive heat rejection 5. Fission product barriers including fuel matrix and cladding 6. PCB including reactor barrel 7. GVS 8. Manual scram 9. CD insertion capability	1. Loss of Offsite Power. 2. Insertion of CDs on LOOP (Reactivity control FSF met). 3. Heat removal by the NSR active PGS is unavailable. 4. Successful SR passive decay heat removal (Decay heat removal FSF met). 5. Fuel/Cladding/PCB temperatures controlled to within criteria. 6. SR confinement barriers remain intact (Confinement FSF met). 7. No radiological release. 8. Reactor shutdown is by manual scram.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-5: Seismic Event (g≤SSE) [SBE-13]	IE: A Seismic Event (g ≤ SSE) occurs. Reactivity control: The SR seismic trip system activates CDs to shut down the reactor (Reactivity control FSF met). Heat removal: Successful heat removal by the NSR-AR PGS. Core temperature is controlled to with limits (Decay heat removal FSF met). Confinement: No Fuel/Cladding barrier or PCB structural damage (Confinement FSF met). End State: No radiological release.	1. Seismic Trip 2. CD insertion capability 3. Primary NaK circulation flowpath and core coolable geometry 4. Passive heat rejection 5. Fission product barriers including fuel matrix and cladding 6. PCB including reactor barrel 7. GVS	1. Seismic Event (g ≤ SSE). 2. SR seismic trip system activates CDs to rotate to shut down the reactor (Reactivity control FSF met). 3. Heat removal by the NSR active PGS is unavailable. 4. Successful SR passive decay heat removal due to system SDC-2 design (Decay heat removal FSF met). 5. Fuel/Cladding/PCB temperatures controlled to within criteria. 6. SR confinement barriers remain intact (Confinement FSF met). 7. No radiological release.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-6: Loss of Coolant Accident (LOCA) [SBE-10, -11, -12]	IE: Break of low-elevation components (downcomer, lower plenum) inside guard vessel, PCB penetration leaks/breaks/seal ruptures or support SSC failure. SR Guard vessel and cover gas pressure prevent core from being uncovered. Reactivity control: The NSR-AR trip system activates CDs to shut down the reactor (Reactivity control FSF met). Heat removal: SR Guard vessel prevents core from being uncovered and allows for successful passive heat removal (Decay heat removal FSF met). Confinement: No Fuel/Cladding barrier or PCB structural damage (Confinement FSF met). End State: No radiological release.	1. IRF 2. Primary NaK circulation flowpath and core coolable geometry 3. GVS 4. IHX design 5. Passive heat rejection 6. Fission product barriers including fuel matrix and cladding 7. PCB including reactor barrel 8. Manual scram 9. CD insertion capability	1. Break of the low-elevation components (downcomer, lower plenum). 2. The NSR-AR RPS and insertion of CDs is not available. 3. Reactor stabilizes via SR IRF (Reactivity control FSF met). 4. SR Guard Vessel design and cover gas pressure prevent core from being uncovered and allows for successful passive heat removal (Decay heat removal FSF met). 5. Fuel/Cladding/PCB temperatures controlled to within criteria. 6. SR confinement barriers remain intact (Confinement FSF met). 7. No radiological release. 8. Reactor shutdown is by manual scram.
DBA-7: NaK Spill and Fire [SBE-17]	IE: NaK release and fire during unloading. Reactivity control: N/A. The reactor is shutdown. Heat removal: N/A. The reactor is shutdown. Confinement: Contaminated NaK release and fire outside of the reactor (Confinement FSF not met). End State: Radiological and non-radiological release.	1. IHX design	1. NaK release and fire. 2. The NaK is assumed to be contaminated from prior reactor operations. 3. The entire volume of NaK is assumed to leak and be engulfed in fire. 4. Radiological and non-radiological hazardous material release.

Table 13. MARVEL DBA summary.

DBA Identifier [Table 8 or Table 9 SBE Identifier]	Limiting SBE Description (All FSFs met)	Available SR-SSCs for the DBA (Table 12)	DBA Sequence of Events (only SR-SSCs assumed available) [Figure 6, ES-6]
DBA-8: Radioactive or Hazardous Material Release, or Direct Radiation Exposure, from a System, Subsystem or Component [SBE--15, -16, -18]	IE: System impact results in in-core fuel and Be reflector damage. Reactivity control: N/A. The reactor is shutdown. Heat removal: N/A. The reactor is shutdown. Confinement: Reactor is open during unloading (Confinement FSF not met). End State: Radiological and non-radiological release.	1. Fission product barriers including fuel matrix and cladding. 2. Reflector Be insert metal design.	1. Heavy load drop due to crane failure or human error results in impact and breach to up to 12 used fuel elements (2 assemblies). 2. Be reflector damage and airborne release. 3. Radiological and non-radiological hazardous material release.
BDBA-1: Seismic Event (g>SSE) [SBE-14]	N/A	Due to seismic event > design basis, all SR-SSCs assumed fail.	1. Seismic Event (g $\geq$ SSE). 2. Core damage occurs (due to seismic event > design basis) to reactor core, internals, and structure, and TREAT reactor building structures and pit. 3. Core rearrangement or compaction leads to energetic reactivity insertion. 4. Total disassembly of core. 5. Bounding radiological and nonradiological release [Maximum Hypothetical Accident (MHA)].

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Appendix A

MARVEL Failure Modes and Effects Analyses

Methodology

A team consisting of MARVEL design engineers, qualified nuclear safety analysts, other MARVEL project personnel, and TREAT personnel undertook a significant effort to identify the failure modes and failure effects of the MARVEL components. The team identified the components of MARVEL, then hypothesized the possible modes of failure for each component, and finally postulated the effect of the failure. There was no consideration of failure or consequence mitigations during this effort. Design features or other mitigations may change the probability of failure and the magnitude or type of consequences. The failure modes and effects analysis (FMEA) was performed to inform the hazard evaluation and the classification of MARVEL SSCs.

Table A-1. Fuel and core system (FCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Fuel System (FS)	Fuel	Achieve criticality and generate heat.	Fuel pin Uranium mass loading error.	Design or manufacturing error.	Greater or less excess reactivity or heat generation than expected.	Quality assurance and configuration management programs to ensure Uranium loading is per design requirements.	Reactor unable to achieve criticality or produce expected heat output.
		Fission product retention meat/cladding.	Cladding leak or failure.	Design or manufacturing error. Internal pressure build up (fission gas, hydrogen release), pellet cracking, fuel swelling, weld failure, excessive temperatures, fission product cladding interactions, fretting (unlikely due to velocities).	Release of loose material to coolant. Core flow blockages (localized). Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure cladding failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Net negative temperature coefficient.	Fuel/Cladding material loading error, or structural failure.	Design or manufacturing error.	Unexpected decrease in net negative temperature coefficient. Uncontrollable reactor. Loss of margin to failure.	Quality assurance program. Design features to ensure that 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.	Moderate to large reactivity insertion.
		Heat transfer.	Fuel/Cladding material loading error, or structural failure. Bowing of fuel pins.	Design or manufacturing error.	Insufficient heat transfer from fuel to primary coolant. Mismatch between flow and power. Decrease in PCS natural convection flow rate. Temperature gradients. Fuel temperature limits challenged. Loss of confinement; fission product release. Excessive hydrogen loss.	Quality assurance program. Design features and operating provisions to ensure Fuel/Cladding failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Neutron economy.	Fuel pin Uranium mass loading error.	Design or manufacturing error.	Greater or less excess reactivity or heat generation than expected.	Quality assurance and configuration management programs to ensure Uranium loading is per design requirements.	Reactor unable to achieve criticality or produce expected heat output.
		Coolable geometry.	Temperature gradients. Fuel assembly structural failure.	Design or manufacturing error.	Inadequate heat transfer within core or blockage. Pin bowing. Mechanical defects (straightness, tolerance). Decrease in PCS natural convection flow rate. Fuel Cladding temperature limits challenged. Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure fuel assembly failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.

Table A-1. Fuel and core system (FCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
		Gas retention (separation from NaK).	Cladding leak or failure.	Corrosion of steel, formation of zirconium oxide, formation of uranium oxide, inferior end of life strength or premature cladding failure.	Mechanical defects (pin holes). CIA assembly insertion. Release of loose material to coolant. Core flow blockages. Loss of confinement; fission product release to coolant. Fuel handling damage.	Corrosion prevention design features and programs.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Erosion control (fretting).	Cladding leak or failure.	Chemical interactions.	Release of loose material to coolant. Decrease in PCS natural convection flow rate. Hydrogen leakage into PCS. Core flow blockages. Loss of confinement; fission product release to coolant.	Corrosion prevention design features and programs.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Structural integrity post seismic.	Fuel assembly structure failure.	Seismic event.	Loss of coolable geometry. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. fuel and cladding temperature limits challenged. Loss of confinement and fission product release to coolant and environment.	SSC design to SDC-2 seismic requirements.	Reduction in natural circulation through the core. Radioactive material release to environment.
	Radial Be Core Reflector (metal) Inserts	Neutron economy.	Be material or mass loading error.	Design or manufacturing error.	Higher or lower Be reflection than desired Greater or less excess reactivity or heat generation than expected.	Quality assurance and configuration management programs to ensure Be loading is per design requirements.	Reactor unable to achieve criticality or produce expected heat output.
		Coolable geometry.	Reflector structural failure.	Design or manufacturing error.	Inadequate heat transfer within core or blockage. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure Be reflector failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Shielding.	Reflector structural failure.	Design or manufacturing error.	Increased neutron radiation exposure to outer parts of reactor.	Quality assurance program. Design features and operating provisions to ensure Be reflector failure is not credible.	Direct radiation exposure to core SSCs or personnel.
		Neutron thermalization/reflection.	Reflector structural failure.	Design or manufacturing error.	Greater or less excess reactivity or heat generation than expected.	Quality assurance program. Design features and operating provisions to ensure Be reflector failure is not credible.	Reactor unable to achieve criticality or produce expected heat output.
		Heat transfer.	Reflector structural failure.	Design or manufacturing error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel temperature limits challenged. Loss of confinement; fission product release. Excessive hydrogen loss.	Quality assurance program. Design features and operating provisions to ensure Be reflector failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
	Neutron Source	Startup neutron population, approach to critical.	Neutron source failure.	Neutron source is not securely housed near the reactor core.	Inability to adequately monitor initial criticality and to keep startup instrumentation within range during reactor startup.	Design features and operating provisions to ensure neutron source housing failure is not credible.	Reactor unable to achieve criticality or produce expected heat output.

Table A-1. Fuel and core system (FCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Core Structures System (CSS)	Lower Grid Plate Structures	Coolable geometry.	Lower grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure lower grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Structural support of fuel.	Lower grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure lower grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Structural support of neutron source.	Neutron source failure.	Neutron source is not securely housed near the reactor core.	Inability to adequately monitor initial criticality and to keep startup instrumentation within range during reactor startup.	Design features and operating provisions to ensure lower neutron source housing failure is not credible.	Reactor unable to achieve criticality or produce expected heat output.
		Net negative temperature coefficient.	Lower grid plate structural failure.	Design or manufacturing error.	Unexpected decrease in net negative temperature coefficient. Uncontrollable reactor. Loss of margin to failure.	Quality assurance program. Design features to ensure that 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.	Moderate to large reactivity insertion.
		Fuel rod separation.	Lower grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure lower grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
	Upper Grid Plate Structures	Coolable geometry.	Upper grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure upper grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Structural support fuel.	Upper grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core.. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure upper grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.

Table A-1. Fuel and core system (FCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
		Provide for axial expansion.	Upper grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure upper grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Fuel handling.	Failure of lift handling hardware.	Design or manufacturing error. Hoisting or rigging error.	Fuel drop. Loss of confinement and fission product release.	Quality assurance program. Design features and operating provisions to ensure fuel lifting hardware failure is not credible.	Radioactive material release to environment.
		Fuel rod separation.	Upper grid plate structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant.	Quality assurance program. Design features and operating provisions to ensure upper grid plate failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Net negative temperature coefficient.	Upper grid plate structural failure.	Design or manufacturing error.	Unexpected decrease in net negative temperature coefficient. Uncontrollable reactor. Loss of margin to failure.	Quality assurance program. Design features to ensure that 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.	Moderate to large reactivity insertion.
Stationary Core Reflector System (SCR)	Stationary BeO Core Reflector Plates (Outside Reactor Barrel)	Neutron economy (neutron reflection back into core).	Structural failure or misplacement/movement of stationary reflector plates.	Design or manufacturing error.	Core reactivity is unstable and insufficient to enable core operation.	Quality assurance program. Design features and operating provisions to ensure stationary reflector failure is not credible.	Reactor unable to achieve criticality or produce expected heat output.
		Shielding.	Structural failure or misplacement/movement of stationary reflector plates.	Design or manufacturing error.	Increased neutron radiation exposure to outer parts of reactor.	Quality assurance program. Design features and operating provisions to ensure stationary reflector failure is not credible.	Direct radiation exposure to core SSCs or personnel.
		Neutron thermalization/reflection.	Structural failure or misplacement/movement of stationary reflector plates..	Design or manufacturing error.	Greater or less excess reactivity or heat generation than expected.	Quality assurance program. Design features and operating provisions to ensure stationary reflector failure is not credible.	Reactor unable to achieve criticality or produce expected heat output.
		Heat transfer.	Structural failure or misplacement/movement of stationary reflector plates.	Design or manufacturing error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel temperature limits challenged. Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure stationary reflector failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.

Table A-1. Fuel and core system (FCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
		Void space reduction.	Structural failure or misplacement/movement of stationary reflector plates.	Design or manufacturing error.	Cannot keep core covered (cooled) on LOCA. Coolable geometry. Decrease in PCS natural convection flow rate. Insufficient heat transfer from fuel to primary coolant. Fuel cladding temperature limits challenged. Loss of confinement; fission product release. Fuel temperature limits challenged. Excessive hydrogen loss.	Quality assurance program. Design features and operating provisions to ensure stationary reflector failure is not credible.	PCB penetration leaks/breaks/seal ruptures Reduction in natural circulation through the core. Radioactive material release to environment. NaK leaks and fires
		Maintain adequate gap with control drums.	Structural failure or misplacement/movement of stationary reflector plates.	Design or manufacturing error.	SCR interferes with the control drum rotation. Binding of drums. Thermal expansion. Debris from crumbling BeO.	Quality assurance program. Design features and operating provisions to ensure stationary reflector failure is not credible.	Moderate to large reactivity insertion.

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Primary Coolant System (PCS)	NaK	Fission product retention (plate-out, chemical solubility, and/or adsorption mechanisms).	NaK material loading error.	Design or manufacturing error.	Pressure beyond normal. Loss of confinement; fission product release.	Quality assurance and configuration management programs to ensure NaK loading is per design requirements.	Radioactive material release to environment
		Heat transfer	NaK material loading.	Design or manufacturing error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement; fission product release. Excessive heat transfer.	Quality assurance and configuration management programs to ensure NaK loading is per design requirements.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Compatibility with cladding	Cladding corrosion and failure.	Chemical interactions.	Release of loose material to coolant. Decrease in PCS natural convection flow rate. Loss of confinement; fission product release to coolant. Hydrogen leakage into PCS. Core flow blockages.	Corrosion prevention design features and programs.	Reduction in natural circulation through the core. Radioactive material release to environment.
		Net negative temperature coefficient	NaK material loading error, or structural failure.	Design or manufacturing error.	Unexpected decrease in net negative temperature coefficient. Uncontrollable reactor. Loss of margin to failure.	Quality assurance program. Design features to ensure that 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.	Moderate to large reactivity insertion.
	Reactor Vessel, Upper Vessel Head, Distribution Block, Downcomers	Fission product retention	Barrel/Vessel failure.	Design or manufacturing error.	Loss of PCB confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure PCB failure is not credible.	Radioactive material release to environment.
		NaK retention	Barrel/Vessel failure.	Design or manufacturing error.	Loss of PCB confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure PCB failure is not credible.	PCB penetration leaks/breaks/seal ruptures NaK leak/fire.
		Natural circulation. Coolable geometry.	PCS SSC structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant. Insufficient conduction path of decay heat removal.	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	Reduction in natural circulation through the core. Radioactive material release to environment.

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
		Maintain pressure	Low PCS pressure, NaK boiling under accident conditions.	Design or manufacturing error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement; fission product release. NaK boiling. Vessel cracking. Water leak/spray on hot vessel. Heatup rates causing excessive stress	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	Reduction in natural circulation through the core. PCB penetration leaks/breaks/seal ruptures Radioactive material release to environment.
		Support Core Structure System	Structural material loading error or failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core. Decrease in PCS natural convection flow rate. Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement and fission product release to coolant. Potential LOCA. Fuel failure.	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	PCB penetration leaks/breaks/seal ruptures Reduction in natural circulation through the core. Radioactive material release to environment. NaK leaks and fires
		Negative temperature coefficient	Structural material loading error.	Design or manufacturing error.	Unexpected decrease in net negative temperature coefficient. Uncontrollable reactor. Loss of margin to failure.	Quality assurance program. Design features to ensure that 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.	Moderate to large reactivity insertion.
		Decay Heat Removal	Structural material loading error or failure.	Design error or manufacturing error. PCS material error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	Reduction in natural circulation through the core.
		Heat Transfer to Secondary Coolant	Structural material loading error or failure.	Design error or manufacturing error. PCS material error. Impurities.	Noncondensible dissolved gases. Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	Reduction in natural circulation through the core.
		Compatibility with primary coolant (NaK)	Structural material corrosion and failure,	Corrosion of primary coolant boundary	Loss of confinement; fission product release.	Corrosion prevention design features and programs.	Radioactive material release to environment. NaK leak. NaK spill/fire.

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
		PV boundary/ containment (Valves (Pressure transmitter, Argon and NaK connection)	Structural material loading error or failure.	Design error or manufacturing. Material error.	Loss of pressure (Potential to allow core to be uncovered). Decrease in PCS natural convection flow rate. Freeze closed/open. NaKLeak	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	PCB penetration leaks/breaks/seal ruptures Reduction in natural circulation through the core. Radioactive material release to environment. NaK leaks and fires
		Radionuclide sampling port	Structural material loading error or failure.	Design error or manufacturing. Material error.	Loss of confinement; NaK release.	Quality assurance program. Design features and operating provisions to ensure PCS SSC structural failure is not credible.	PCB penetration leaks/breaks/seal ruptures. Radioactive material release to environment. NaK leaks and fires
	Insulation	Component temperature control	Control system failure.	Design or manufacturing error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement; fission product release. Decreased ability to remove decay heat.	Quality assurance program. Design features and operating provisions to ensure PCS insulation failure is not credible.	Reduction in natural circulation through the core.
		Heat transfer	Insulation failure.	Design or manufacturing error. Material error.	Insufficient heat transfer within core. Decrease in PCS natural convection flow rate. Fuel cladding temperature limits challenged. Loss of confinement; fission product release. Increased stresses. Damage to instrumentation, Stirling engines, Control drum components.	Quality assurance program. Design features and operating provisions to ensure PCS insulation failure is not credible.	Reduction in natural circulation through the core.
Primary Coolant Management System (PCMS)	NaK Storage Tank, Piping (Removable), Valves	Fill/drain NaK, purify NaK, NaK makeup	NaK system leak or failure.	Design or manufacturing error.	Loss of confinement; LOCA, NaK spill/fire, Radioactive material release.	Quality assurance program. Design features and operating provisions to ensure NaK loading SSC failure is not credible.	Radioactive material release to environment. NaK leaks and fires
		Overpressure protection for PCMS	High PCS pressure, PCS component or vessel/barrel failure.	Design or manufacturing error.	Loss of confinement; fission product release (on maintenance/drain), LOCA, NaK spill/fire.	Quality assurance program. Design features and operating provisions to ensure PCMS failure is not credible.	PCB penetration leaks/breaks/seal ruptures Radioactive material release to environment. NaK leaks and fires

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Guard Vessel System (GVS)	Guard Vessel	Maintain pressure Differential	Low PCS pressure, GV Leak or failure	Design or manufacturing error.	Potential to allow core to be uncovered. Decrease in primary coolant system flow rate due to inadequate heat transfer. Radioactive material release to environment. Leak of PCS on side walls or CIA..	Quality assurance program. Design features and operating provisions to ensure GV failure is not credible.	PCB penetration leaks/breaks/seal ruptures Reduction in natural circulation through the core. Radioactive material release to environment. NaK leaks and fires
		Reduce void space	Wrong filler material .	Design or manufacturing error.	Decrease in primary coolant system flow rate due to inadequate heat transfer. Radioactive material release to environment.	Quality assurance and configuration management programs to ensure GV material loading is per design requirements.	Reduction in natural circulation through the core.
		Heat transfer (insulation)	Wrong material.	GV material or design error	Insufficient heat transfer to T-REXC space. Fuel cladding temperature limits challenged. Loss of confinement; fission product release. Failure of neutron detectors.	Quality assurance and configuration management programs to ensure GV insulation loading is per design requirements.	Reduction in natural circulation through the core.
		Shielding	Wrong shielding material, structural failure.	Design or manufacturing error.	Increased neutron radiation exposure to outer parts of reactor.	Quality assurance and configuration management programs to ensure GV shielding is per design requirements.	Direct radiation exposure to core SSCs or personnel.
		Secondary NaK confinement	GV leak or failure.	Design or manufacturing error.	Loss of confinement; NaK release. Potential core uncovering.	Quality assurance program. Design features and operating provisions to ensure GV failure is not credible.	NaK leak. NaK spill/fire.
	Upper Confinement Structure	Radionuclide control	Structure leak or failure	Design or manufacturing error.	Loss of confinement	Quality assurance program. Design features and operating provisions to ensure upper confinement structural failure is not credible.	Radioactive material release.
		Secondary NaK confinement	Structure leak or failure, material error	Design or manufacturing error.	Loss of confinement; NaK release Instrumentation failure.	Quality assurance program. Design features and operating provisions to ensure upper confinement structural failure is not credible.	NaK leak. NaK spill/fire.
		Material temperature control (ventilation flow and insulation)	System or component failure. Ventilation failure.	Design or manufacturing error.	Loss of confinement; NaK release, fission product release. Instrumentation failure. Failure of primary vessel (from insulation failure) Failure of control drum actuators.	Quality assurance program. Design features and operating provisions to ensure upper confinement structural failure is not credible.	NaK leak. NaK spill/fire.
		Interface with electronics	System or component failure.	Design or manufacturing error.	Loss of confinement; NaK release, fission product release. Loss of instrumentation signals,	Quality assurance program. Design features and operating provisions to ensure upper confinement structural failure is not credible.	NaK leak. NaK spill/fire.
		Interface with fluids	Structure leak or failure	Design or manufacturing error.	Loss of confinement; NaK release, fission product release. Water leak, gas leak.	Quality assurance program. Design features and operating provisions to ensure upper confinement structural failure is not credible.	NaK leak. NaK spill/fire.

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Inert Gas System (IGS)	Argon gas bottle/tank, Distribution pipes/hoses/valves, Inert Gas System Guard Vessel, Gas sampling port	Overpressure protection	System or component failure.	Design or manufacturing error.	Loss of confinement; fission product release. Over pressurize primary or GV.	Quality assurance program. Design features and operating provisions to ensure IGS SSC structural failure is not credible.	NaK leak. NaK spill/fire.
		Contain activation products (valves and ports)	System or component failure.	Design or manufacturing error.	Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure IGS SSC structural failure is not credible.	NaK leak. NaK spill/fire.
		Maintenance of pressure	Improper vessel (primary and GV)	Design or manufacturing error.	Insufficient heat transfer within core. Fuel cladding temperature limits challenged leading to Loss of confinement; fission product release.	Quality assurance and configuration management programs to ensure GV material is per design requirements.	Reduction in natural circulation through the core.
		Measure Fission Product Gas	NaK vapor condensation	Design or manufacturing error.	Failure to detect cladding breach	Quality assurance program. Design features and operating provisions to ensure IGS SSC structural failure is not credible.	Radioactive material release.
		Prevent NaK oxidation	System or component failure. (wrong gas)	Design or manufacturing error.	Loss of confinement; fission product release. Heat transfer reduction. Potential flow blockage.	Quality assurance program. Design features and operating provisions to ensure IGS SSC structural failure is not credible.	NaK leak. NaK spill/fire.
Reactor Support Frame (RSF)	Support Frame	Structural support of: primary vessel, guard vessel	Material or structural failure.	Design or manufacturing error.	Failure to maintain the primary coolant pressure drop across the core (from a non vertical vessel). Decrease in PCS natural convection flow rate (from a non vertical vessel). Insufficient heat transfer within core. Fuel cladding temperature limits challenged leading to Loss of confinement and fission product release to coolant. Breach of GV and primary. Restrict air flow for decay heat removal. Prevent reactor shutdown.	Quality assurance program. Design features and operating provisions to ensure RSF SSC failure is not credible.	Reduction in natural circulation through the core. Nak leak. NaK fire.
		Survive seismic event (most items)	Material or structural failure during seismic event.	Seismic Event	Failure to maintain the primary coolant pressure drop across the core (from a non vertical vessel). Decrease in PCS natural convection flow rate (from a non vertical vessel). Insufficient heat transfer within core. Fuel cladding temperature limits challenged leading to Loss of confinement and fission product release to coolant. Breach of GV and primary. Restrict air flow for decay heat removal.	SSC design to SDC-2 seismic requirements.	Reduction in natural circulation through the core. Nak leak. NaK fire.
		Pathway for thermal expansion	Material or structural failure.	Design or manufacturing error.	Increased stresses in GV causing GV failure.	Quality assurance program. Design features and operating provisions to ensure RSF SSC failure is not credible.	Reduction in natural circulation through the core. Nak leak. NaK fire.

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Reflector Support System (RSS)	Reflector Straps, Upper Reflector Support Plate, Lower Reflector Support Plate	Support of stationary BeO	Failure/breakage of support structure	Design or manufacturing error.	Insufficient heat transfer. Fuel cladding temperature limits challenged leading to Loss of confinement; fission product release. Impact to primary coolant boundary causing primary containment failure and NaK release. Impact/restriction to control drums preventing rotation	Quality assurance program. Design features and operating provisions to ensure RSS SSC structural failure is not credible.	Reduction in natural circulation through the core. Nak leak. NaK fire.
		Support of control drums	Failure/breakage of support structure	Design or manufacturing error.	Failure/breakage of support structure. Uncontrollable reactor.	Quality assurance program. Design features and operating provisions to ensure RSS SSC structural failure is not credible.	Moderate to large reactivity insertion.
		Spacing of upper and lower plates	Failure/breakage of support structure	Design or manufacturing error.	Failure/breakage of support structure. Uncontrollable reactor.	Quality assurance program. Design features and operating provisions to ensure RSS SSC structural failure is not credible.	Moderate to large reactivity insertion.
		Leveling control (alignment)	Failure/breakage of support structure	Design or manufacturing error.	Failure/breakage of support structure. Uncontrollable reactor.	Quality assurance program. Design features and operating provisions to ensure RSS SSC structural failure is not credible.	Moderate to large reactivity insertion.
		Shielding (from structural materials)	Shielding material or design error, structural failure.	Design or manufacturing error.	Increased neutron radiation and activation exposure to outer parts of reactor and electronics above reactor.	Quality assurance program. Design features and operating provisions to ensure RSS SSC structural failure is not credible.	Direct radiation exposure to core SSCs or personnel.
		Reduce void space	Wrong filler material or design error	Design or manufacturing error.	Potential to allow core to be uncovered.	Quality assurance and configuration management programs to ensure filler material loading is per design requirements.	Moderate to large reactivity insertion.
	Zirc Debris Shield	Prevent binding of control drums	Breakage of shield	Design or manufacturing error.	Binding of control drums preventing rotation.	Quality assurance program. Design features and operating provisions to ensure RSS SSC structural failure is not credible.	Moderate to large reactivity insertion.
Secondary Coolant System (SCS)	IHXs	Separate primary and secondary coolants	Loss of IHX boundary due to Stirling engine force translation to the IHX (ie., the Stirling engine can translate force to the IHX if the secondary coolant is solid or if the Stirling engine has a helium leak).	Stirling engine or IHX material or design error.	Flow blockage, Loss of confinement; NaK release. Change in heat removal capability. Loss of secondary coolant into primary.	Quality assurance program. Design features and operating provisions to ensure IHX structural failure is not credible.	Reduction in natural circulation through the core. NaK leak. NaK fire.
		Heat transfer	IHX material or design error, structural failure.	IHX material or design error.	Decreased heat loss from core. Limit reactor power. Loss of confinement; fission product release. Inability to generate power. Loss of natural circulation.	Quality assurance program. Design features and operating provisions to ensure IHX structural failure is not credible.	Reduction in natural circulation through the core. NaK leak. NaK fire.

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Secondary Output Structure (SOS)	Mounting Brackets, Vibration Isolators (Frame)	Maintain Stirling engines vertical.	Material or structural failure.	Material or design error.	Stirling engine tip, contacting IHX	Quality assurance program. Design features and operating provisions to ensure SOS SSC structural failure is not credible.	Undercooling or decrease in heat removal
	Vibration Isolators	Dampen the vibration caused by the operating engines to minimize impacts on the rest of the MRS.	Material or structural failure.	Material or design error.	Transmit vibration to distribution plenum.	Quality assurance program. Design features and operating provisions to ensure SOS SSC structural failure is not credible.	Undercooling or decrease in heat removal
Secondary Support Structure (SSS)	Steel Frame Above PCS Distribution Block	Supports Stirling engines	Material or structural failure.	Material or design error.	Engine hits other components.	Quality assurance program. Design features and operating provisions to ensure SSS SSC failure is not credible.	Undercooling or decrease in heat removal
		Position vibration isolators	Material or structural failure.	Material or design error.	Transmit vibration to distribution plenum.	Quality assurance program. Design features and operating provisions to ensure SSS SSC failure is not credible.	Undercooling or decrease in heat removal
	Guide Pins	Contain vibration motion to the vertical plane	Material or structural failure.	Material or design error.	Transmit horizontal vibration to distribution plenum.	Quality assurance program. Design features and operating provisions to ensure SSS SSC failure is not credible.	Undercooling or decrease in heat removal
Secondary Cover Gas System (SCGS)	Exhaust Ductwork, Back Pressure Regulator, HEPA Filter, Bellows	Overpressure protection	Stirling engine helium tube failure. Breach of bellows, bending or failure of seal	Material or design error.	Loss of IHX boundary due to Stirling engine force translation to the IHX (i.e., Stirling engine has a high-pressure helium leak). Release of radioactive products into Upper Confinement	Quality assurance program. Design features and operating provisions to ensure SCGS SSC failure is not credible. Pressure relief or opening to upper confinement.	Undercooling or decrease in heat removal
		Remove activation products	Flow blockage in inlet or exhaust tubes	Material or design error.	Buildup of activation products and pressure within SCCGS.	Quality assurance program. Design features and operating provisions to ensure SCGS SSC failure is not credible.	Undercooling or decrease in heat removal
		Maintain secondary coolant O2 content	Breach of bellows, bending or failure of seal, O2 sensor failure	Material or design error.	Formation of oxide within SCS. Flow blockage in IHX resulting in LOHS	Quality assurance program. Design features and operating provisions to ensure SCGS SSC failure is not credible.	Undercooling or decrease in heat removal
		Seal contents from environment	Breach of bellows, bending or failure of seal	Material or design error.	Release of radioactive products into Upper Confinement	Quality assurance program. Design features and operating provisions to ensure SCGS SSC failure is not credible.	Undercooling or decrease in heat removal
	Actuated Valves	Build and maintain pressure within IHX	Failed valve. Failed actuator.	Material or design error.	Not enough pressure. O2 ingress. Actuation of pressure relief (bellows) from high pressure	Quality assurance program. Design features and operating provisions to ensure SCGS SSC failure is not credible.	Undercooling or decrease in heat removal

Table A-2. MARVEL reactor structure system (MRS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Secondary Coolant Management System (SCMS)	Purification Skid, Piping (removable), Valves [If Used]	Contain and move secondary coolant	Line or tank breach, rupture, or failure	Material or design error.	Release of radioactive products	Quality assurance program. Design features and operating provisions to ensure SCMS SSC failure is not credible.	Undercooling or decrease in heat removal
		Purify and control O2 content of secondary coolant	O2 sensor failure	Material or design error.	Incorrect O2 purification. Formation of oxide. Corrosion of IHX materials	Quality assurance program. Design features and operating provisions to ensure SCMS SSC failure is not credible.	Undercooling or decrease in heat removal
		Fill, and drain the secondary coolant.	Heater burnout or failure. Over or under fill.	Material or design error.	Solidification of coolant within SCMS, inability to fill SCS. Limit heat transfer and power output. Gas bubbling through IHX.	Quality assurance program. Design features and operating provisions to ensure SCMS SSC failure is not credible.	Undercooling or decrease in heat removal
	Vacuum pump [If Used]	Siphon coolant	Vacuum seal failure. Pipe clog. Piping failure. Pump failure. Controls failure.	Material or design error.	Can't siphon coolant Failure to drain coolant. Coolant freezing/thawing around engine coils and IHX.	Quality assurance program. Design features and operating provisions to ensure SCMS SSC failure is not credible.	Undercooling or decrease in heat removal
	Regulator [If Used]	Push coolant	Wrong setting. Failed regulator.	Material or design error.	Wrong pressure (over/under). Can't siphon coolant. Can't operate at full power. Actuation of pressure relief (bellows) from high pressure.	Quality assurance program. Design features and operating provisions to ensure SCMS SSC failure is not credible.	Undercooling or decrease in heat removal
Reactor Shielding System (SHLD)	Radial Gamma and Neutron Shields Outside Guard Vessel, Axial Gamma and Neutron Shields Above Core Reflectors	Radiological shielding	Material or structural failure.	Shielding material or design error, structural failure.	Increased neutron and gamma radiation exposure to outer parts of reactor and electronic components. Failure of electronics from radiation exposure. Bias in temperature readings.	Quality assurance program. Design features and operating provisions to ensure SHLD SSC failure is not credible.	Increased radiation exposure to core SSCs. Direct Radiation Exposure to Personnel
		Decay heat convection flow path	Material or structural failure.	GV material or design error, structural failure.	Insufficient heat transfer within core. Fuel cladding temperature limits challenged. Loss of confinement; fission product release.	Quality assurance program. Design features and operating provisions to ensure SHLD SSC failure is not credible.	Reduction in natural circulation through the core.

Table A-3. MARVEL reactivity control system (RCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Drum Forcing System (DFS)	Control Drum Motors, Motor Controllers, Motor Resolvers	Actuate a control element	Electronics failure	Design /material error	Oscillations, overshoots	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
		Position indication (resolver)	Electronics failure, incorrect reference point	Design /material error	Motion too fast (or slow) or no motion	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
		Current monitoring	Incorrect commands	Design /material error	Loss of position conformance check. No indication of drum rotation issues (current monitoring)	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Control Drum EM Clutch	Separate shaft from drive (scram)	Electronics failure, Heat damage, Radiation damage	Design /material error	Unplanned scram, unable to scram, No indication of drum rotation issues (current monitoring)	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
		Current monitoring		Design /material error			Moderate to large reactivity insertion.
Drum Structures System (DSS)	Control Drum Cage & Rails, Cage Platforms	Structural support of drum motors, switches	Seismic failure, heat damage, radiation damage	Design /material error	Drum misposition (change in reactivity) Drum binding, unable to scram, unable to startup	Quality assurance program. Design to SDC-2 requirements. SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	Control Drum Shaft	connect control drum to drive system	Heat damage, radiation damage	Design /material error	Drum shaft binding, unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	Control Drum Hard Stops	Excess reactivity control	Cam slips on shaft, Breakage (from impact), Improper placement (positioning)	Design /material error	Reactivity insertion limit exceeded leading to high temperature and fuel cladding failure, fission product release	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
		Max and min drum rotation physical limit		Design /material error	Violation of safety basis		Moderate to large reactivity insertion.
	Control Drum (Rotary) Seal & Standoff	Limit leakage of argon (from GV)	Heat damage, radiation damage, mechanical wear/damage, misalignment	Design /material error	leak of GV (or PV) gas, leak of Nak	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
		Minimize seal friction		Design /material error	Drum shaft binding, unable to scram, unable to startup		Moderate to large reactivity insertion.
	Control Drum Lock	Physical lock of drum in shutdown position	Physical damage	Design /material error	unexpected motion of control drum, unplanned criticality	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Axial Expansion Springs	Accommodate axial expansion of drums. Keep individual BeO plates compressed.	Heat damage, radiation damage	Design /material error	Drum binding, unable to scram, unable to startup, changes in reactivity. Change of drum configuration.	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	Springs	Stored energy for drum shutdown (scram) rotation	Heat damage, radiation damage, mechanical damage, debris	Design /material error	Slow scram, unable to scram, slow drum movement	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	Couplings	Accommodate misalignment between drive and drum shaft	design /material error, heat damage, radiation damage, mechanical damage	Design /material error	Drum shaft binding, unable to scram, unable to startup, slow drum movement	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.

Table A-3. MARVEL reactivity control system (RCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
	Upper Alignment Bearings	Align control drums, Allow rotary motion, Minimize friction	Heat damage, radiation damage, mechanical damage	Design /material error	Drum shaft binding, unable to scram, unable to startup, slow drum movement	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	Lower Support Bearings	Support control drums, Align control drums, Allow rotary motion, Minimize friction	Heat damage, radiation damage, mechanical damage	Design /material error	Drum shaft binding, unable to scram, unable to startup, slow drum movement	Quality assurance program. Design features and operating provisions to ensure DSS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
Drum Neutronics System (DNS)	Poison Plates	Shutdown reactor, Control neutron population, Seismic survival	Heat damage, radiation damage, mechanical damage	Design /material error	poison misposition (change in reactivity) Drum binding, unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure DNS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	BeO Plates	Support and position poison	Heat damage, radiation damage, mechanical damage	Design /material error	poison misposition (change in reactivity) Drum binding, unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure DNS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
Drum Position Measurement System (DPMS)	Control Drum Position Indicator & Gear	Indicate control element absolute position	Electronics failure, Heat damage, Radiation damage.	Design /material error	Loss of drum position indication to operator	Quality assurance program. Design features and operating provisions to ensure DPMS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	In Limit Switch, Out Limit Switch	Indication of endpoints position	Electronics failure, (heat/radiation damage).	Design /material error	Loss of drum max/min position indication.	Quality assurance program. Design features and operating provisions to ensure DPMS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
Drum Forcing System (DFS)	CIA Motor, Motor Controller, Motor Resolver, Motor Gear	Actuate a control element	Design flaw.	Design /material error	Position overshoot	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
		Position indication (resolver), Current monitoring	Electronics failure.	Design /material error	Loss of position conformance check, Motion too fast (or slow) or no motion	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible..	Moderate to large reactivity insertion.
	CIA Linear Bearings, Ball Screws Nuts	minimize friction and wear	Heat damage, radiation damage.	Design /material error	CIA binding, unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	CIA Electromagnet	release poison from drive mechanism	Heat damage, radiation damage.	Design /material error	Unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure DFS electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
Drum Structures System (DSS)	CIA Cage Standoffs, Rails, Platforms	Structural support of rod motors, switches	Seismic failure, heat damage, radiation damage.	Design /material error	Rod misposition (change in reactivity) Rod binding, unable to scram, unable to startup	Quality assurance program. Design to SDC-2 seismic requirements.	Moderate to large reactivity insertion.

Table A-3. MARVEL reactivity control system (RCS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Drum Neutronics System (DNS)	CIA Rod (B_4C) and Drive Shaft	Connect poison to drive system	Heat damage, radiation damage.	Design /material error	Unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure CIA electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
	CIA Gray Rod (Hafnium)	Reactivity control, Additional hold down, Excess reactivity control	Heat damage, radiation damage, seismic failure.	Design /material error	Poison misposition (change in reactivity) Drum binding, unable to scram, unable to startup	Quality assurance program. Design features and operating provisions to ensure CIA electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
Drum Position Measurement System (DPMS)	CIA Position Indicator, In Limit Switch, Out Limit Switch	Indicate control element absolute position	Electronics failure, (heat/radiation damage).	Design /material error	Loss of drum max/min position indication. Loss of absolute position indication. Loss of motor cutoff switch.	Quality assurance program. Design features and operating provisions to ensure CIA electronics SSC failure is not credible. Shielding. Insulation.	Moderate to large reactivity insertion.
		Indication of endpoints position		Design /material error			Moderate to large reactivity insertion.
		Drive cutoff.		Design /material error			Moderate to large reactivity insertion.

Table A-4. MARVEL instrumentation and control system (ICS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Interlocks	Control Drum and CIA Motor Relays	Restrict movement to one drum or the CIA rod at a time	Coil burns out, Contacts welded shut, (power loss circuit failure)	Design /material error	Can't choose between rods as desired.	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
Human Machine Interface (HMI)	HMI Screen	Interface to software	Electronics failure	Design /material error	No display (improper input)	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Analog Pressure Indication	Pressure indication	Electronics failure, mechanical failure	Design /material error	Wrong pressure (no pressure, low pressure, high pressure)	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	LED Lights	Full in and full out position indication	LED failure, Electrical failure	Design /material error	Not full inserted indication No full withdrawn indication	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
Control System	I/O Modules	Provide inputs and outputs to control system	Electronics failure	Design /material error	Cease to provide input or output	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Chassis	Supports the control system	Electronics failure, DC power supply failure	Design /material error	Failure of communication failure of communication and I/O of all components in rack.	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Computer	Runs control system	Electronics failure, Power supply failure	Design /material error	No indications No operator interaction capability	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	UPSs	Supply uninterrupted power to control system	Battery dies, won't charge electronics fail	Design /material error	Power loss to control system (either on source power loss or failure of UPS electronics) (Loss of shutdown indication)	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	DC Power Supply Unit	Power chassis Power instruments	Electronics failure with no DC power supplied. Over voltage Over current	Design /material error	Reactor scram on loss of DC power Loss of/improper indication Loss of chassis power Loss of communication (to peripheral equipment)	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
Reactor Protection System (RPS)	Scram Button	Scram reactor	Electrical failure, Mechanical failure	Design /material error	Reactor won't scram when commanded Reactor scrams unexpectedly	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	DC Power Supply Unit	Provide DC power to clutch mechanisms.	Electronics failure with no DC power supplied. Over voltage	Design /material error	Reactor scram on loss of DC power Burnup of coil with scram on over voltage.	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Key Switch	Power termination	Electrical failure, Mechanical failure	Design /material error	Reactor won't scram when commanded Reactor scrams unexpectedly	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.

Table A-4. MARVEL instrumentation and control system (ICS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
		Scram reactor	Electrical failure, Mechanical failure.	Design /material error	Reactor won't scram when commanded Reactor scrams unexpectedly	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Seismic Sensor	Sense seismic wave	Software programming error, Electronics failure	Design /material error	Fail to shutdown before S-wave arrives	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
		Scram reactor	Electrical failure, Mechanical failure.	Design /material error	Spurious scrams Fail to scram when needed	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Scram Circuit (breakers, relays, latch coils)	Scrams reactor when needed	Loss of power, Coil burns out, Contacts welded shut	Design /material error	Reactor scrams	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Neutron detectors and Supporting Equipment	Dynamic power indication	Electronics failure, Temperature induced damage to cables and detectors, detector failure	Design /material error	Loss of channel indication	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
			Electronics failure, Temperature induced damage to cables and detectors, detector failure	Design /material error	Improper indication of period	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
			Electronics failure, Temperature induced damage to cables and detectors, detector failure	Design /material error	Spurious scrams no scram when needed	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
			Electronics failure, Temperature induced damage to cables and detectors, detector failure	Design /material error	Improper limit indication (don't detect limit reached, indicate limit reached falsely)	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Thermocouples	Measure temperature	Mechanical failure, Wrong positioning, electronics failure, Radiation effects (heating)	Design /material error	Wrong temperature, No temperature (open circuit), Calibration shift (bias), Wrong indication of reversed flow	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
				Design /material error	Conflicting NaK Level Indication	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
				Design /material error	Conflicting power indication	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
				Design /material error	fails to scram when should or scrams when shouldn't	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.

Table A-4. MARVEL instrumentation and control system (ICS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
				Design /material error	improper indication of relation to limits	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Leak Detectors	Detect NaK leak	NaK doesn't flow to sensor Bridging, Corrosion, Open circuit (wire break)	Design /material error	Fail to sense a leak, Falsely sense a leak	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Pressure Sensors	Sense pressure differential between primary and guard vessel	Electronics failure, Mechanical failure	Design /material error	Wrong pressure (no pressure, low pressure, high pressure). Radiation/Temperature (Environment) Damage to Heater	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
			Electronics failure, Mechanical failure	Design /material error	Wrong Power level indicated	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
		Pressure boundary	Mechanical failure	Design /material error	Leak causing loss of pressure or NaK vapors, Reduce margin to limits, Loss of heat removal capability, Loss of fission products	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Instrumentation Seals	Pressure boundary	Mechanical failure, Manufacturing defect, Improper material	Design /material error	Leak causing loss of pressure or NaK vapors, Reduce margin to limits, Loss of heat removal capability, Loss of fission products	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Smoke Detector	Indication of a NaK leak in the upper confinement	Power Failure	Design /material error	Failure to detect fire, False indication of fire, Radiation/Temperature (Environment) Damage to smoke detector	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.
	Accelerometer	Sense vibration signature from engines	Power failure, Radiation effects, Improper setup, Temperature effects	Design /material error	Loss of indication of engine motions, Radiation/Temperature (Environment) Damage to accelerometer	Quality assurance program. Design features and operating provisions to ensure ICS electronics SSC failure is not credible.	Moderate to large reactivity insertion.

Table A-5. MARVEL power generation system (PGS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
Electrical Production System (EPS)	QB80 Engine	Heat removal.	Coolant system cracks, stresses, leaks, Corrosion.	Design /material error	Release of water	Quality assurance program. Design features and operating provisions to ensure EPS SSC failure is not credible. Corrosion prevention Quality assurance program. Design features and programs.	Undercooling or decrease in heat removal
			Coolant flow to heat mismatch.	Design /material error	Water phase change	Quality assurance program. Design features and operating provisions to ensure EPS SSC failure is not credible.	Undercooling or decrease in heat removal
		Generate electricity.	Radiation damage.	Design /material error	No electricity generation, Electrical short (+/-250V)	Shielding to protect Stirling Engines.	Undercooling or decrease in heat removal
		Helium pressure retention.	Tubing stresses from secondary coolant freezing/thawing, impacts to solid secondary coolant, Corrosion.	Design /material error	Release 50 bar helium, Pipe whip (damage to primary containment)	Quality assurance program. Design features and operating provisions to ensure EPS tubing SSC failure is not credible from temperature changes. Corrosion prevention design features and programs.	Undercooling or decrease in heat removal
	Water Line Connection and Pipes	Entrains engine cooling water.	Leaks, Over pressure, pressure relief, Line failure.	Design /material error	Release of water, Over heating engine (damage), release of water into core.	Quality assurance program. Design features and operating provisions to ensure EPS line failure is not credible.	Undercooling or decrease in heat removal Reactivity and Power Distribution Anomalies
		Prevents contamination spread.	Leaks, Over pressure, pressure relief, Line failure.	Design /material error	Release of activate water, create a contamination area, Pipe whip (damage to control drum drives)	Quality assurance program. Design features and operating provisions to ensure EPS line failure is not credible.	Undercooling or decrease in heat removal
	Qenergy Engine Control Units (ECUs)	Control engine.	Electronics failure, design error, radiation damage, corrosion, condensation (cause electronics failure).	Design /material error	Abnormal control of the engine: more vibration and out of control. Stop the engine	Quality assurance program. Design features and operating provisions to ensure EPS electronics failure is not credible. Shielding to protect Stirling Engines. Corrosion prevention design features and programs.	Undercooling or decrease in heat removal
		Dissipate excess electricity.	Electrical hazards (Arcs, flashes, shorts).	Design /material error	Destroy engine, Stop engine	Quality assurance program. Design features and operating provisions to ensure EPS electronics failure is not credible.	Undercooling or decrease in heat removal
		Condition electricity output.	Electronics failure, design error, radiation damage, operator error.	Design /material error	Unconditioned electricity supply	Quality assurance program. Design features and operating provisions to ensure EPS electronics failure is not credible.	Undercooling or decrease in heat removal
	Qenergy Computer/HMI	Interface with control unit.	Operator error, programing error, computer (screen) failure.	Design /material error	Loss of interface to ECU	Human factors design.	Undercooling or decrease in heat removal
Engine Cooling System	Compact Heat Exchangers	Transfer heat from water to glycol system.	Pressure relief, clogging, Scale buildup.	Design /material error	Reduced heat transfer	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal Facility Fires
	Water Piping/tubing	Entrains engine cooling water.	Leaks, Over pressure, pressure relief.	Design /material error	Release of water, Over heating engine (damage)	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
		Prevents contamination spread.	Leaks, Over pressure, pressure relief, Line failure.	Design /material error	Release of activate water create a contamination area Pipe whip	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal

Table A-5. MARVEL power generation system (PGS) failure modes and effect analysis.

Subsystem	Major Components	Functions	Failure Modes	Cause	Failure Effects	Preventative Measures	Postulated Resulting Initiating Events
	Glycol Piping/tubing	Entrains engine glycol.	Leaks, Over pressure, pressure relief.	Design /material error	Release of glycol, loss of heat transfer to environment, Overheat engine	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal Facility Fires
	Heat Rejection Units (HRUs)	Transfer heat to environment.	Physical damage, clogging/debris.	Design /material error	Loss of heat transfer to environment, Overheat engine	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Flow/Temp Sensor	Inform engine controller of flow and temp of engine coolant.	Design error, electronics failure.	Design /material error	Abnormal control of the engine: more vibration and out of control. Inaccurate indication. Stop the engine	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Resistance Temperature Detector, Flow Meter	Power calibration (redundancy for system flow and temperature indication).	Design error, electronics failure.	Design /material error	Inaccurate power indication, Unknown power level	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
		Engine coolant over temp detection.		Design /material error	Damaged engine	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
		Engine coolant leak detection.		Design /material error	Failure to detect leak	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Pumps	Generate coolant flow.	Corrosion, blockage, physical damage.	Design /material error	Loss of heat transfer	Quality assurance program. Corrosion prevention design features and programs.	Undercooling or decrease in heat removal
	Fill Tanks (water and glycol)	Maintain reserve fluid.	Design error, electronics failure, leak, physical damage.	Design /material error	Loss of heat transfer, loss of coolant flow	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
		Over pressure protection.		Design /material error	Failure to detect leak	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	HRU fan	Heat transfer (force air circulation over radiator).	Electronics failure, physical damage, signal error.	Design /material error	Loss of heat transfer	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Check Valve	Force flow in correct direction.	Clogging, stuck shut/open.	Design /material error	Loss of heat transfer from engine (overheat engine)	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Engine Stall Circuit	Stop the engine.	Sensor failure, electronics failure.	Design /material error	Doesn't stop engine when needed, electrical hazard, (electrical fire)	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Pressure relief valve	Relieve pressure.	Stuck open/closed, blockage.	Design /material error	Over pressurization, inefficient decreased heat transfer, boiling of cooling water	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal
	Drain	Drain whole system.	Clogging, debris, leak.	Design /material error	Inability to drain system, loss of cooling fluid leading to loss of heat transfer and overheated/damaged engine	Quality assurance program. Design features and operating provisions to ensure ECS SSC failure is not credible.	Undercooling or decrease in heat removal