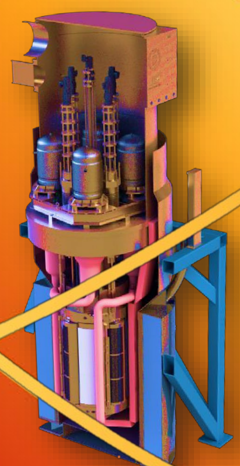
(Microreactor Applications Research, Validation & EvaLuation),

ANS Risk-informed, Performance-based Principles and Policy Committee (RP3C) Community of Practice (CoP):



Application of a Qualitative Risk-Informed, Performance-Based Approach for the MARVEL Microreactor at the Idaho National Laboratory

August 29, 2025

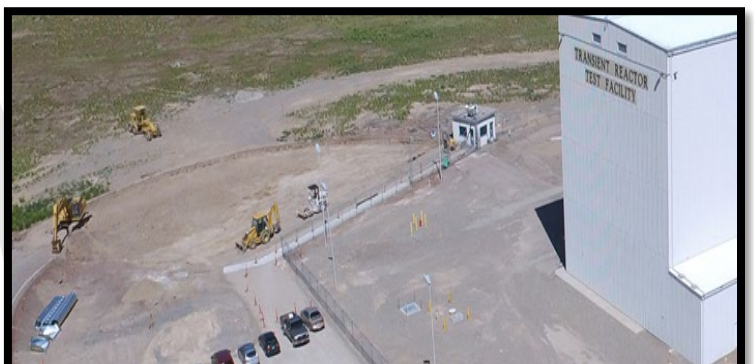
Doug Gerstner
Nuclear Safety Engineer
Idaho National Laboratory, USA

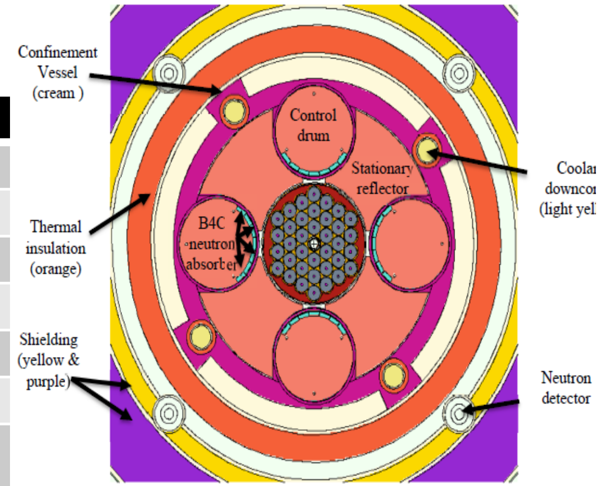
PRESENTATION OVERVIEW

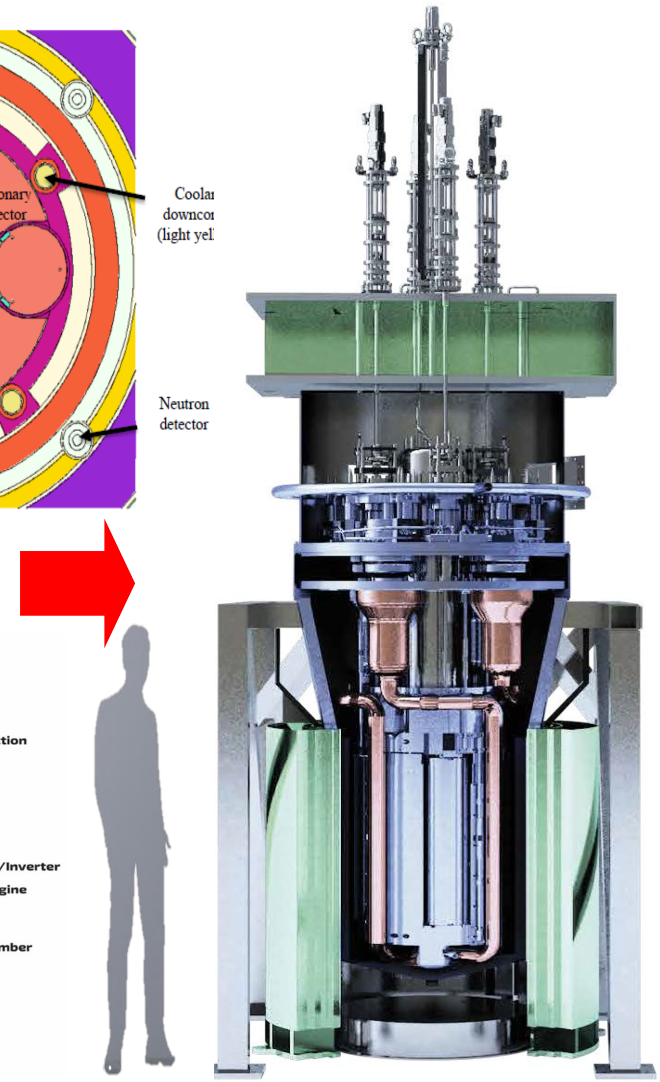
- MARVEL Design Overview
- Regulatory Framework
- Hazards & Accident Analyses Approach
- Qualitative Risk Analysis Methodology
- Qualitative RIPB Results/Conclusions

MARVEL Design Overview

Key Design Features	
Thermal Power	85 kW nominal
Electrical Power	20 kWe
Weight	< 12 US ton
Primary Coolant	Sodium-Potassium eutectic (NaK)
Secondary Coolant	NaK
Coolant Driver	Natural Convection, single phase
Fuel	TRIGA HALEU (UZrH), 304SS clad, end caps
Moderator	Hydrogen
Neutron Reflector	Graphite, Beryllium (S200), Beryllium oxide
Reactivity Control	Radial Control Drums, Central Absorber
Primary Coolant Boundary	SS316H





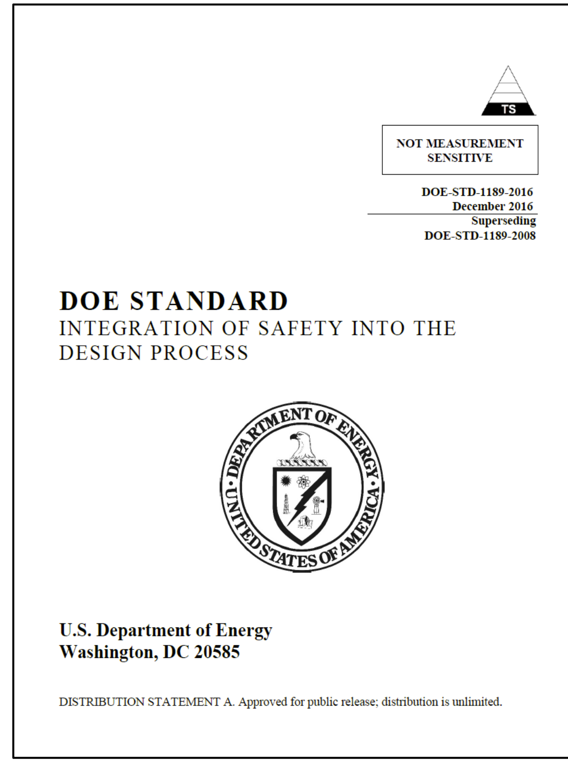


Evolution of MARVEL from P-53 to current design

MRP Microreactor Program

Safety-in-Design Regulatory Framework

- **10 CFR 830, Subpart B, Nuclear Safety Management**
  - Establishes the safety basis requirements for Department of Energy (DOE) nuclear facilities.
  - Requires the development of a documented safety analysis (DSA) to ensure that hazard controls are established to provide for the adequate protection of workers, the public, and the environment.
  - RG 1.70 "Safe Harbor" for reactors.
- **DOE O 420.1C, Facility Safety**
  - Establishes facility and programmatic requirements for:
    - Nuclear safety design criteria
    - Fire protection
    - Criticality safety
    - Natural phenomena hazards (NPH) mitigation
- **DOE-STD-1189-2016, Integration of Safety into the Design Process**


TS

NOT MEASUREMENT SENSITIVE

DOE-STD-1189-2016
December 2016
Superseding
DOE-STD-1189-2008

DOE STANDARD
INTEGRATION OF SAFETY INTO THE DESIGN PROCESS

U.S. Department of Energy
Washington, DC 20585

DISTRIBUTION STATEMENT A. Approved for public release; distribution is unlimited.

> ***Process outlined in DOE-STD-1189-2016, "Integration of Safety Into the Design Process," provides an opportunity to force a common understanding and agreement at key phases.***

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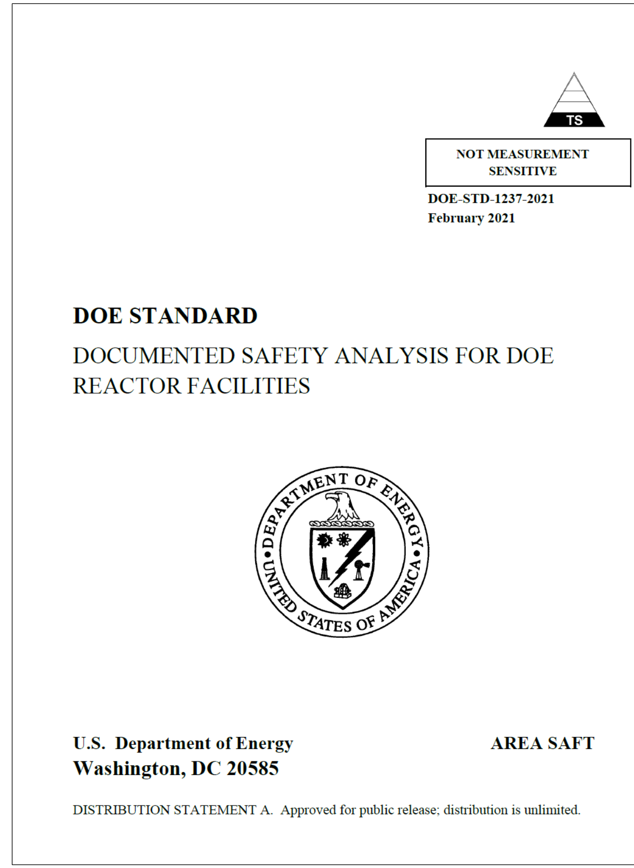
DSA Format and Content

- **DOE-STD-1237-2021, Documented Safety Analysis for DOE Reactor Facilities**
  - DSA Format and Content: DOE-STD-3009-2014 (Section 4) shall be applied for the general DSA Content. ANSI/ANS-15.21-2012 "Format and Content for Safety Analysis Reports for Research Reactors," ANSI/ANS-14.1-2004, and NUREG-1537 may be used to supplement the DSA format and content as appropriate.
  - Hazard Identification and Evaluation: DOE-STD-3009-2014 (Section 3.1) shall be applied. The following standards may be used, as applicable, to supplement hazard evaluation methodologies: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.
  - Design Basis Accident Identification: DOE-STD-3009-2014 (Section 3.2.1) shall be applied. The following standards may be used, as applicable, for guidance in identifying design basis accidents: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.
  - Accident Consequence Analysis: DOE-STD-3009-2014 (Section 3.2.2 – 3.2.4) shall be applied. The following standards may be used, as applicable, for accident analysis guidance: NUREG-1537, NRC Reg Guide 1.70, and NRC Reg Guide 1.203.
  - Hazard Control Selection: DOE-STD-3009-2014 (Section 3.3) shall be applied.
  - SSC Classification Hierarchy: DOE-STD-3009-2014 (Section 3.3) shall be applied. SSCs that protect the reactor core shall be designated at least safety-significant.
  - Probabilistic Risk Assessment: Not expected.
  - Defense in Depth: DOE-STD-3009-2014 (Section 3.3.2) shall be applied. See also DOE-STD-3009-2014 Appendix A.9 and DOE G 420.1-1 for additional information.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NOT MEASUREMENT SENSITIVE

DOE-STD-1237-2021
February 2021

DOE STANDARD

DOCUMENTED SAFETY ANALYSIS FOR DOE REACTOR FACILITIES

DEPARTMENT OF ENERGY • UNITED STATES OF AMERICA •

U.S. Department of Energy **AREA SAFT**
Washington, DC 20585

DISTRIBUTION STATEMENT A. Approved for public release; distribution is unlimited.

- ***TREAT SAR and MARVEL PDSA in the form of a 17-chapter RG 1.70 (RG 1.206) SAR***

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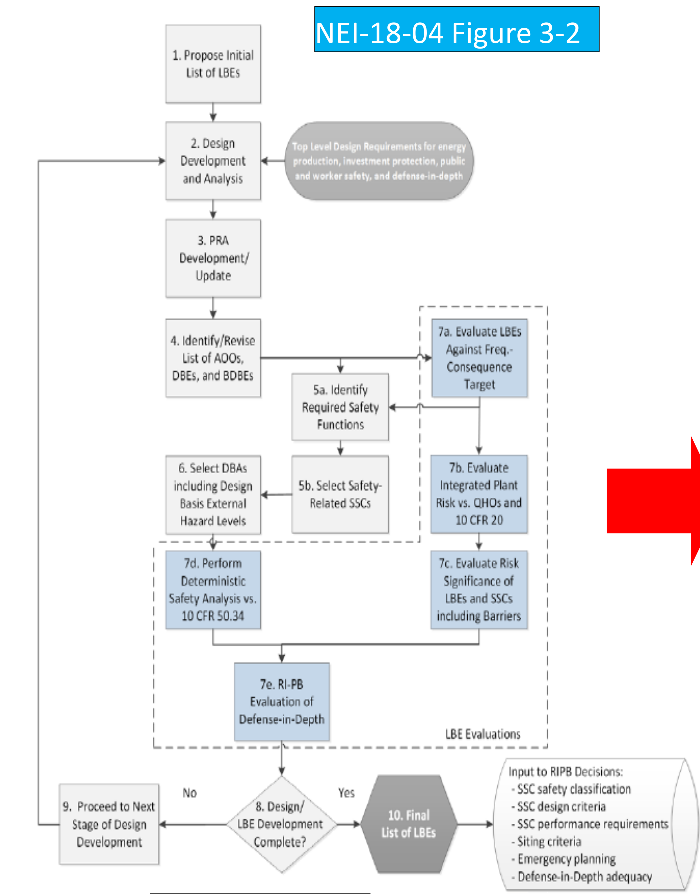
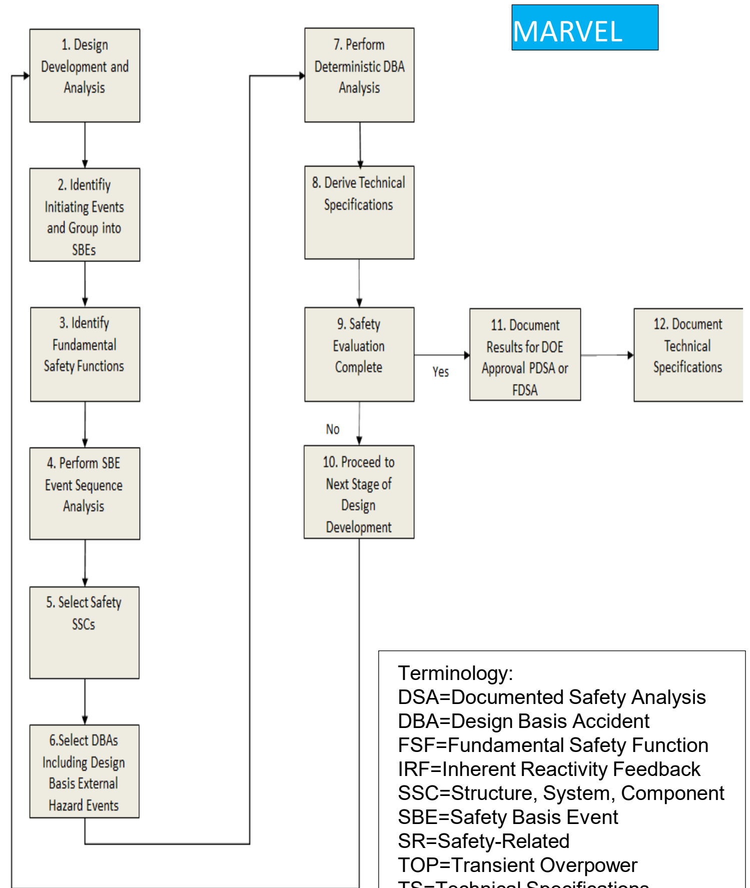
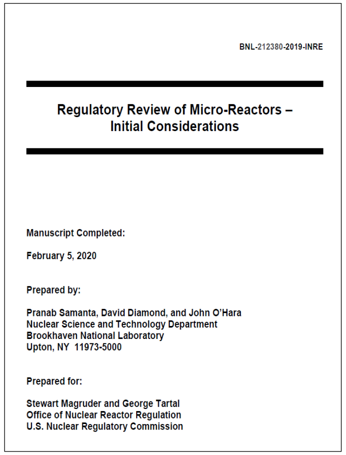
MARVEL Hazard Evaluation Methodology



Source: NEI 18-04



Terminology:
DSA=Documented Safety Analysis
DBA=Design Basis Accident
FSF=Fundamental Safety Function
IRF=Inherent Reactivity Feedback
SSC=Structure, System, Component
SBE=Safety Basis Event
SR=Safety-Related
TOP=Transient Overpower
TS=Technical Specifications


BNL-212380-2019-INRE

Regulatory Review of Micro-Reactors – Initial Considerations

Manuscript Completed:

February 5, 2020

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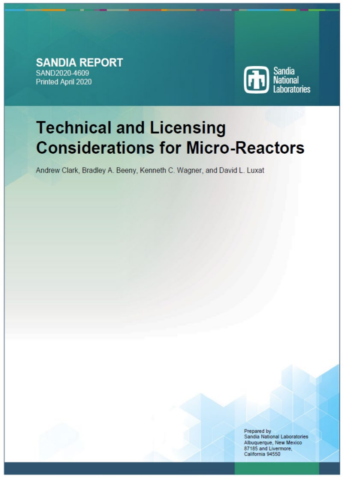
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission


SANDIA REPORT
SAND2020-4609
Printed April 2020

Sandia National Laboratories

Technical and Licensing Considerations for Micro-Reactors

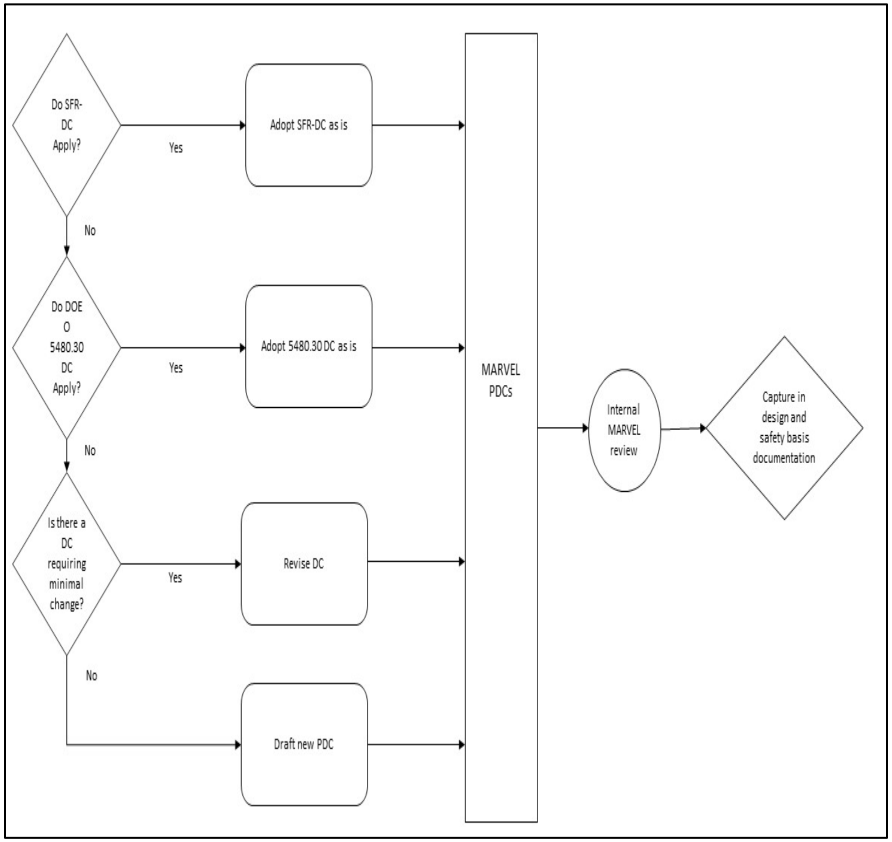
Andrew Clark, Bradley A. Beeny, Kenneth C. Wagner, and David L. Luxat

- *PRA not performed – MARVEL Hazard evaluation is largely qualitative.*
- *Meets the intent of 10CFR830 for a hazard evaluation and for protection of the public, worker, and environment. Process similar to LMP methodology.*

MRP Microreactor Program

Design Development and Analysis (Step 1)

- Principal Design Criteria (PDC):
  - RG 1.232 Appendix B sodium fast reactor (SFR) design criteria (SFR-DC) used as the guidance to develop PDC for the MARVEL .
  - 3 SFR-DC rewritten as new PDCs corresponding to the passive safety strategy and Fundamental Safety Functions.
    - PDC-16: Functional Confinement
      - Confinement strategy is performance-based in that the confinement performance requirements are derived from the accident analysis and not prescriptively identified.
      - Based on the results of the Chapter 15 Accident Analyses, a leak-tight reactor containment barrier against the uncontrolled release of radioactivity to the environment is not required for protection of the public or the worker.
      - Multiple passive barriers result in complicated success paths for fission product release.
    - PDC-26: Reactivity Control
      - Provide passive insertion of negative reactivity to shut down the reactor and maintain in shutdown condition.
    - PDC-36: Core Flow/Heat removal
      - Robust passive design ensures adequate residual heat removal during off-nominal events to ambient air.



- ***PDCs were incorporated into system Technical and Functional Requirements (T&FR) documents for implementation in the design.***

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Identification of Initiating Events (IEs) (Step 2)

Source: DOE-STD-3009-2014

Internal Events:

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

Transient Overpower (TOP) Example

External Events:

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

Table 4. Qualitative frequency categories.

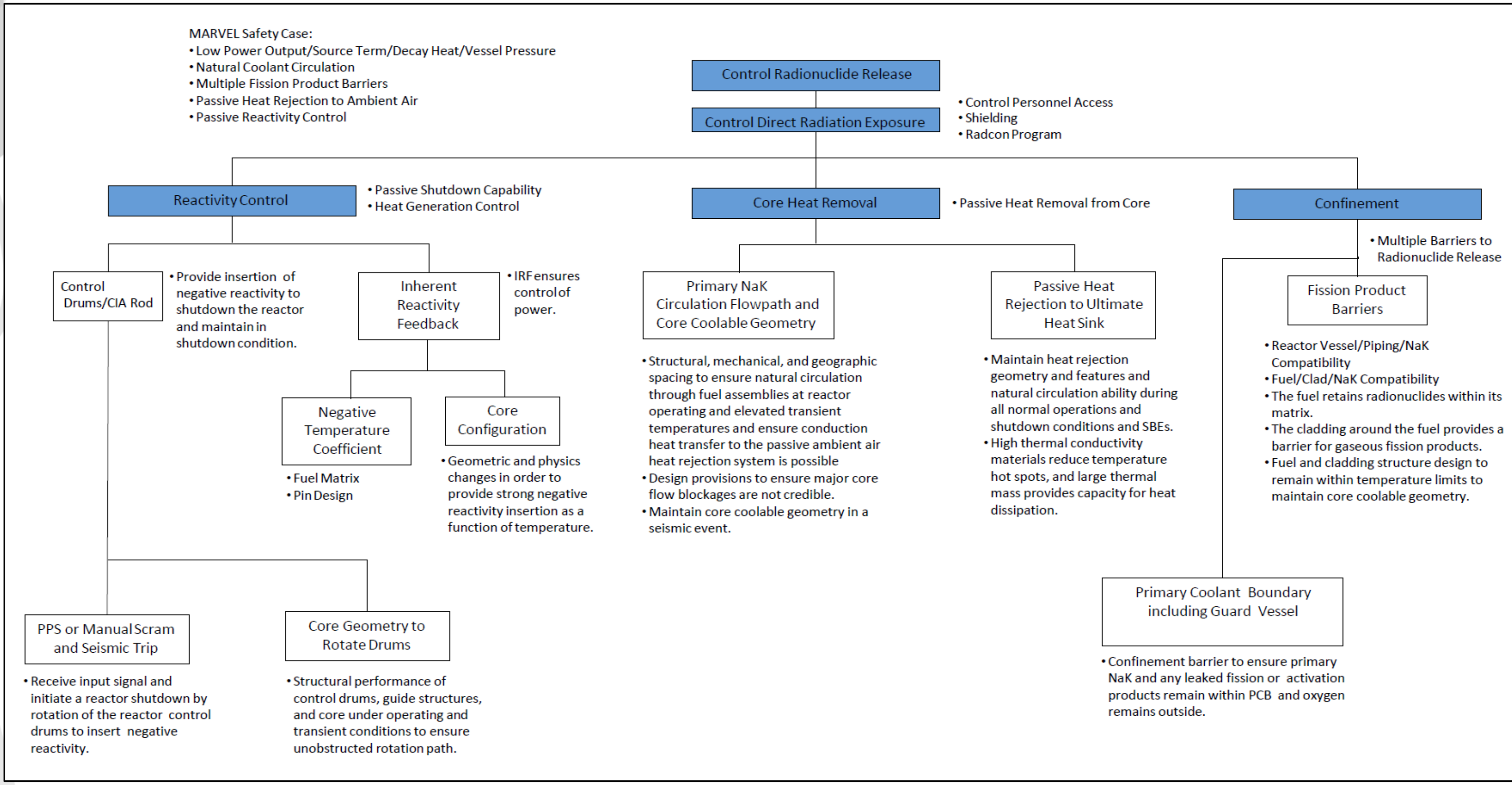
Frequency	Description	Likelihood Range (per year)
Anticipated (A)	Events that may occur during the lifetime of the facility (incidents that commonly occur).	Likelihood $>10^{-2}$
Unlikely (U)	Events that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this likelihood 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 $<10^{-6}$

Reactivity and Power Distribution Anomalies [Transient Overpower (TOP)]:	
– Small reactivity changes (e.g., miss-positioning of a single CD through operator error, spurious trip/operation, or improper drum selection while at power cause enough flux tilt to increase fuel temperature in an assembly)	– Covered under general transients
– Small to moderate reactivity insertion due to a single CD movement or CIA movement due to improper drum position command (software programming error), electronics failures, heat damage, or radiation damage, physical damage, or material, structural, or seismic failures	– Unlikely
– Water line leaks, overpressure, line failure leads to release of water, over heating engine (damage), release of water onto reactor vessel. Thermal stresses, warping of vessel, and constriction of control drum shafts	– Unlikely
– MARVEL control system computer failures results in commands for more than one CD or CIA to go to wrong positions or selection of wrong drives resulting in moderate to large reactivity insertion	– Beyond Extremely Unlikely
– Large reactivity insertion due to core events (misalignment of multiple CD or core configuration change due to bowing, melting or slumping of fuel)	– 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

➢ ***Conservative qualitative assignment of IE frequencies to "bins".***

MRP Microreactor Program

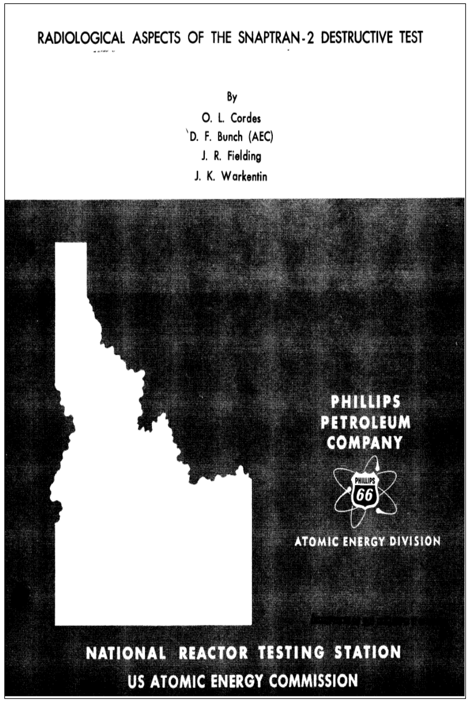
MARVEL Fundamental Safety Functions (FSFs) (Step 3)



- ***Functions that if fulfilled, will keep Initiating Events (IEs) from progressing to end states that could result in core damage and release of radioactive or hazardous material.***

Consequence Analysis (Step 4)

- A hypothetical rearrangement of the core due to a Beyond DBA seismic event (g>SSE), resulting in an extreme reactivity insertion and disassembly of the core similar to the SNAPTRAN-2 and KIWI-TNT tests, results in the maximum possible hypothetical accident consequences.
- The material-at-risk (MAR) results from a reactor run calculated for 7,600,000 megajoules (MJ), which is two years of 24/7 operation at ~120 kW.
- The radiological MAR considers the fuel at end-of-life, and activated primary coolant (NaK), primary cover gas (argon), secondary cover gas (helium), Boron Carbide (B_4C), Secondary Coolant (eGa-In-Sn), and the air in the TREAT pit that contains MARVEL.
- Although not required for such a noncredible, hypothesized event, the consequences to the collocated worker and the public were compared to the extremely unlikely consequence guidelines.
- Based on SNAPTRAN test results, it would be expected the Be from the MARVEL reactor core and reflectors would not represent a source term for non-radiological release consequences in such a similar event analyzed above for bounding radiological consequences.


RADIOLOGICAL ASPECTS OF THE SNAPTRAN-2 DESTRUCTIVE TEST

By
O. L. Cordes
D. F. Bunch (AEC)
J. R. Fielding
J. K. Warkentin

PHILLIPS PETROLEUM COMPANY
ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

- ***Bounding dose of about 4 rem at 6,000 m, well within the offsite guideline to the public of 25 rem (for an extremely unlikely event),***
- ***Bounding dose of about 48 rem to the collocated worker at the TREAT control room (770 m away), well within the onsite guideline to the onsite worker of 100 rem (for an extremely unlikely event).***
- ***Bounding dose to the collocated worker at 100 m away challenges the onsite guideline to the onsite worker of 100 rem (for an extremely unlikely event). As a DID measure, it is recommended that all collocated workers be relocated to the TREAT control room during reactor or used fuel handling operations.***

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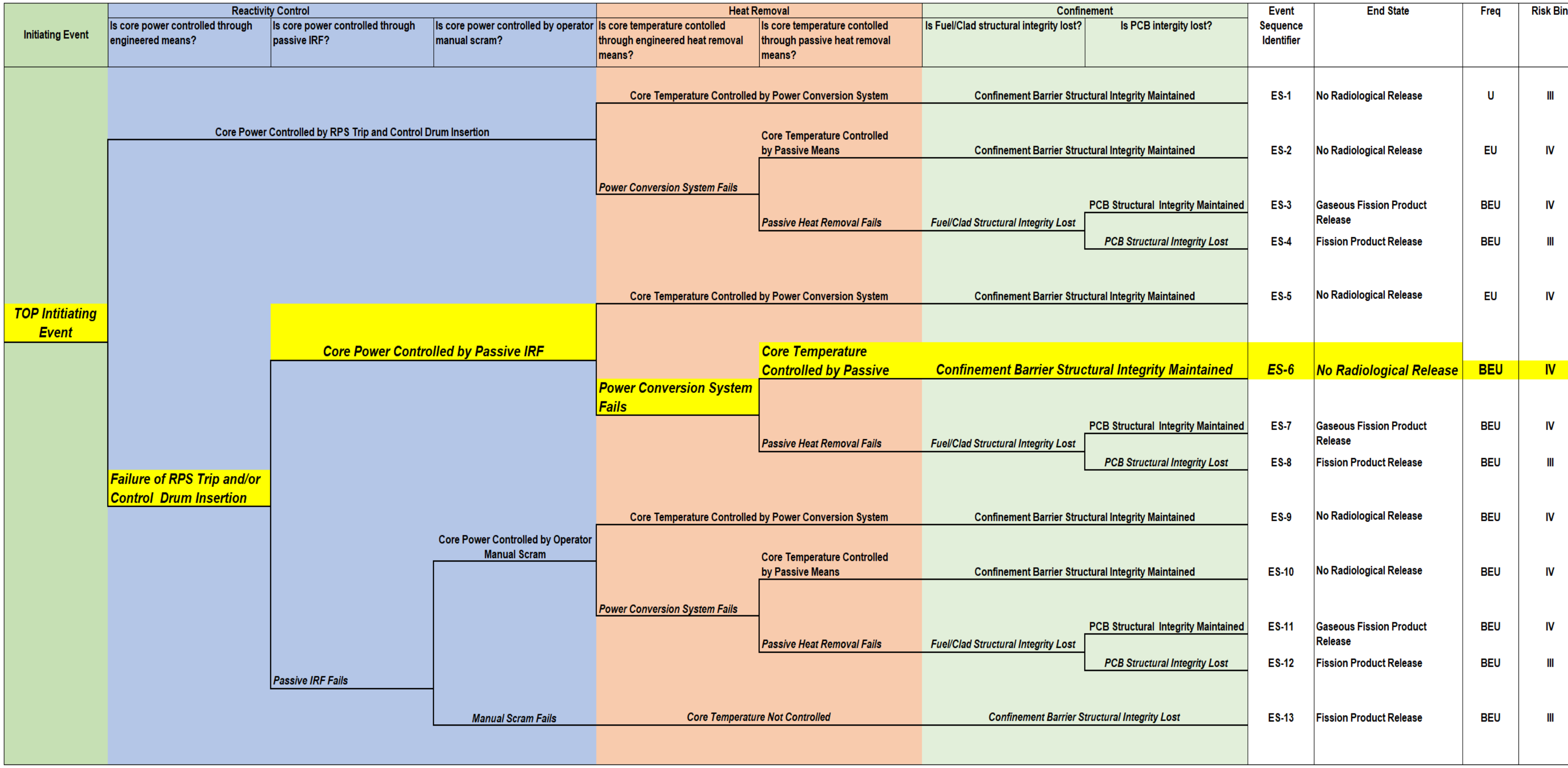
Event Sequence Analysis (Step 4) - TOP Example Only



- *ES-6 relies on passive SSCs alone to perform the FSFs – Result is an end state with no core damage of rad release.*
- *Passive SSCs performing the Reactivity Control (IRF) and Heat Removal (Core SSCs providing conduction and convection to ultimate heat sink) are therefore designated as Safety-Related (SR).*
- *"Unprotected" ES-6 Frequency = (IE) x (Failure of Active Reactivity Control) x (Failure of Active Heat Removal)*

MRP Microreactor Program

Qualitative Event Sequence Analysis (Step 4) – TOP Example Only

Table 8. MARVEL internal SBE accident progression analysis.

SBE Identifier/IE (Table 5)	Event Sequence (Figure 7)	Accident Progression Summary (Table 5 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. Table 5 TOP-1 IE. 2. The SR 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 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	Candidate SR-SSCs: • RPS (Trip Relays) • Reactivity insertion magnitude control (CD interlocks relays and hard stops) • 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 cladding) • PCB (reactor vessel, upper vessel head, distribution block, downcomers, and all PCB penetrations) • GVS • SCB (IHX, IHX bellows) Candidate NSR-AR SSCs: • RPS (SSCs other than listed SR SSCs) • CIA rod • CIA gray rod • CIA interlocks relays and hard stops • Upper confinement Candidate NSR-SSCs: • Drum position indicators • CD drive shaft gearing • CD drive controller software • CD Limit Switches • PGS • Post-accident monitoring • Instrumentation power • Backup power • Stirling engine automatic stop system Candidate ACs: • Controlled PCS cooldown following scram. • Limit DBA total excess reactivity insertion to less than analyzed in ECAR-6332 (< 0.40$). • Software quality assurance program and testing.
	ES-2, 6, 10	1. Table 5 TOP-1 IE. 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. Table 5 TOP-1 IE. 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. Table 5 TOP-1 IE. 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 confinement 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. Table 5 TOP-1 IE. 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	

MRP Microreactor Program

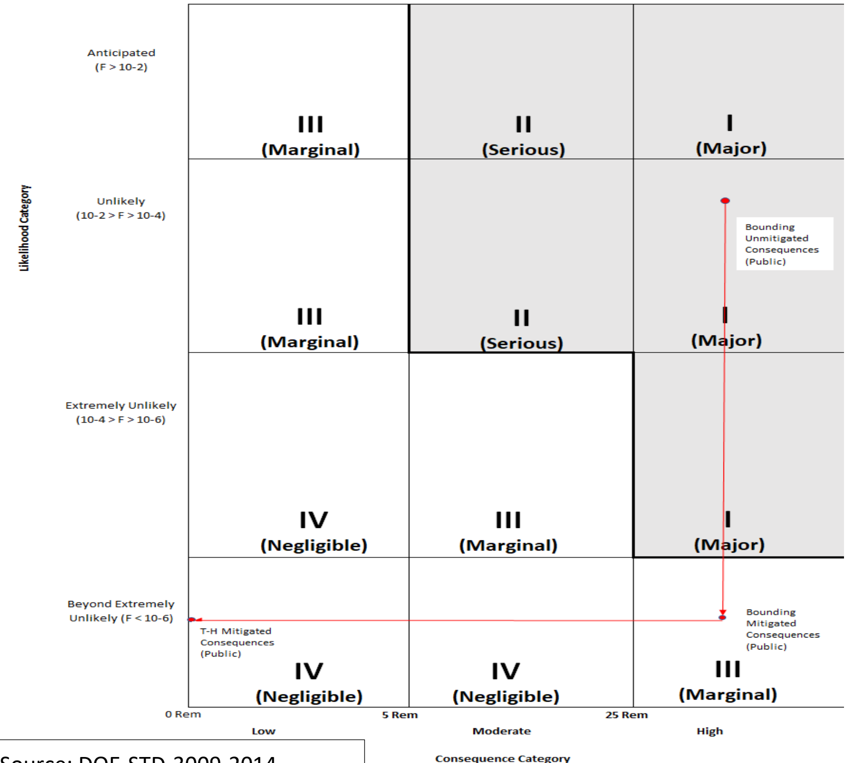
Risk Analysis (Step 4)

DOE-ID Qualitative Frequency Reduction Guidelines:

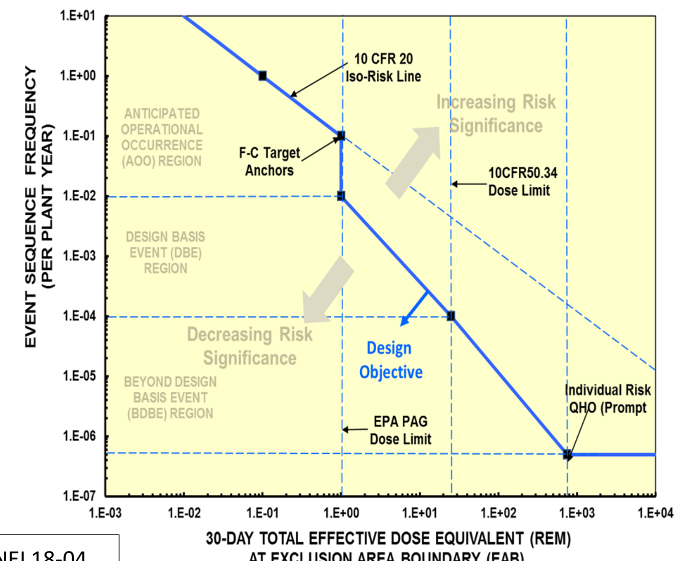
- Administrative controls (including maintenance) procedures: 1 order of magnitude reduction in frequency.
- Active mechanical/electrical engineered safety features without redundant design: 2 order of magnitude reduction in frequency.
- Active mechanical/electrical engineered safety features with redundant and/or independent design features: 3-4 order of magnitude reduction in frequency.
  - Failure of an active FSF SSC (RPS Scram or Stirling Engine heat removal)
- Passive SSC: 3-4 order of magnitude reduction in frequency.
  - Failure of a passive FSF SSC (IRF or passive conduction/convection to ultimate heat sink)

> ***Based on DOE guidelines:***
>
> - **Probability of failure of an active FSF SSC is given a one qualitative frequency category frequency reduction (e.g., Anticipated to Unlikely).**
> - **Probability of failure of a passive FSF SSC is given a two qualitative frequency category frequency reduction (e.g., Anticipated to Extremely Unlikely).**

> ***"Unprotected" TOP Frequency = (IE) x (Failure of Active Reactivity Control) x (Failure of Active Heat Removal) = Beyond Extremely Unlikely***



Source: DOE-STD-3009-2014



Source: NEI 18-04

Safety SSC Classification Results (Step 5)

MARVEL Safety SSC Classification Criteria and Results.

SSC Classification	Criteria		Fundamental Safety Functions		
	Type	Description	Reactivity Control SSCs	Heat Removal SSCs	Confinement SSCs
Safety Related (SR)	Deterministic	1. Is the SSC required to shut down the reactor and maintain it in a safe shutdown condition?	• Passive IRF (fuel, core and internals, reactor support structures) • RPS (Trip relays) • Seismic early warning trip (sensors and trip relays) • Manual scram (trip relays and switches) • Negative reactivity insertion capability (CD cylinders, Be plates, forcing mechanisms, clutch, shafts) • Reactivity insertion magnitude control (CD interlocks relays and hard stops)	• Passive heat rejection (fuel, core and internals, barrel, reactor support structures) • Primary NaK natural circulation and core coolable geometry (fuel, core and internals, barrel, reactor support structures)	• Fission product barriers (fuel cladding) • Primary coolant boundary (Reactor barrel, Downcomers, Upper head, Lower plenum, IHX, Pressure transducers) • Secondary coolant boundary (Guard vessel, CD seal, Pressure transducers, IHX bellows)
	Risk-Informed	2. Is the SSC required to ensure capability to prevent or mitigate the consequences of accidents that could result in potential consequences greater than the consequence guidelines?	• None	• None	• None
	Deterministic	3. Does the SSC contain an item required to establish an SR/NSR interface such that an SR system is isolated from a NSR system?	• None	• None	• Guard vessel • CD seal • Pressure transducers • IHX bellows
		4. Could failure of the SSC prevent reactor shutdown or inhibit a SR SSC function?	• CD cylinders, Be plates, forcing mechanisms, clutch, shafts • CIA gray rod (when used)	• Fuel pins • Reactor core and internals • Reactor barrel • Reactor support structures • Decay heat exhaust system • Outer shell plenum	• Fuel and cladding • Reactor barrel • Downcomers • Upper head, Lower plenum • IHX and bellows • Pressure transducers

MRP Microreactor Program

Selection of Design Basis Accidents (DBAs) (Step 6)

- 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)

Non-DBA Radioactive or Hazardous Material Releases:

- NaK Spill and Fire
- Radioactive or Hazardous Material Release, or Direct Radiation Exposure, from a System, Subsystem or Component

- ***DBAs performed assumed only the SR-SSCs are available***

- ***Reactivity Control:***
  - ***Non-SR (NSR) Reactor Protection System (RPS) and Scram – NOT Available***
  - ***SR Passive Inherent Reactivity Feedback (IRF) – Available***

- ***Heat Removal:***
  - ***NSR Stirlings – NOT Available***
  - ***SR Passive Conduction and Convection SSCs – Available***

MRP Microreactor Program

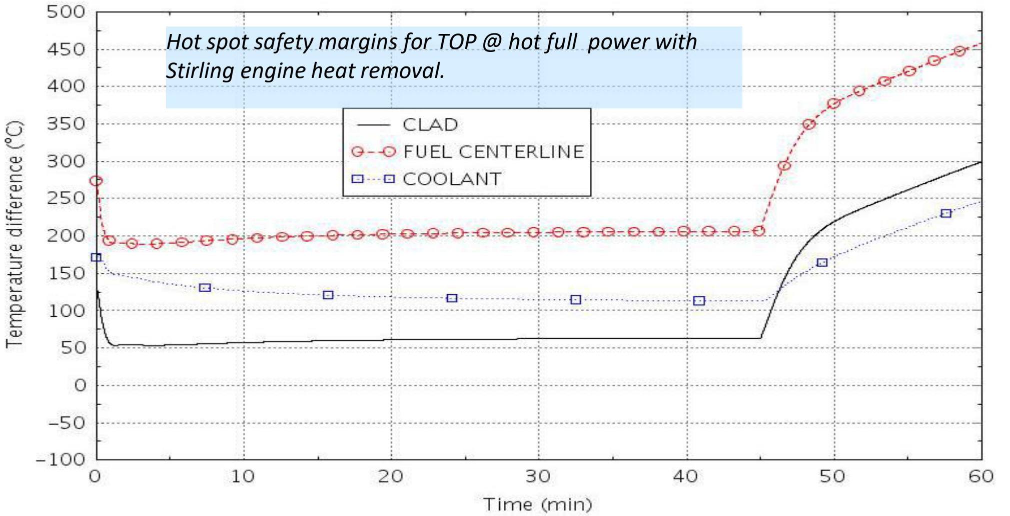
Transient Overpower (TOP) DBA Quantitative Thermal-Hydraulic (T-H) Evaluation (Step 7)

Initiating Event:

- Reactivity insertion ($0.40) due to a single CD movement
- Freq = Unlikely

Available SSCs performing FSFs:

- Active RPS trip of CDs assumed unavailable
- Active heat removal by Stirling Engines assumed unavailable
- Reactivity control is provided exclusively by IRF.
- Core heat removal is provided exclusively by passive conduction/convection to ultimate heat sink.

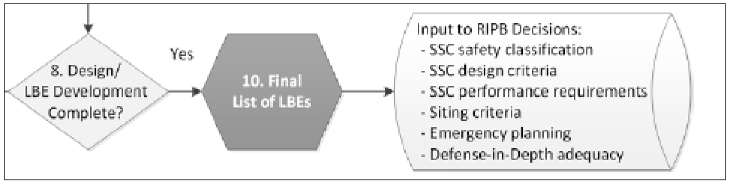


- ➢ ***Despite active FSF failures, passive IRF and heat removal ensure T-H acceptance criteria are met, no core damage occurs, and no radiological release occurs.***
- ➢ ***Confirms the qualitative hazard evaluation assumptions!***

MRP Microreactor Program

Passive FSF SSC Performance Requirements (Step 8)

From NEI-18-04 Figure 3-2:



MARVEL FSF SSC Performance Requirements –

- Specific requirements to provide assurance that the FSF safety SSCs will be capable of performing their intended reactivity control and heat removal safety functions.
- For MARVEL, these are for the SR Passive FSF SSCs.

➢ ***For passive reactivity control and heat removal FSF SR-SSCs, the following performance requirements are defined:***

- **Provide the required feedback as determined in the neutronics analysis and as utilized in the thermal-hydraulic analysis.**
- **Preserve the material properties of core SSCs as assumed in the safety analyses.**
- **Maintain structural integrity at temperatures in all operating conditions.**
- **Withstand material stresses (e.g., creep, swelling) imposed by the operating environment and thermal cycles of the reactor.**
- **Meet the seismic criteria of IBC-2015, using the response coefficients in DOE-STD-1020.**
- **Facilitate the passive removal of fission and decay heat from the core region in the amount assumed in the safety analysis.**

➢ ***Captured in the MARVEL Technical Specifications (TS) Document "Design Features" Section.***

➢ ***Design features "protected" by Administrative Controls (i.e., Quality Assurance, Configuration Management, Maintenance Programs)***

➢ ***No TS Safety Limits (SLs), Limiting Controls Settings (LCSs) identified.***

➢ ***No active SSC Limiting Conditions for Operation (LCO) identified.***

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Conclusions

The proposed MARVEL design:

- Accommodates various beyond-design-basis accident initiators without producing conditions that might lead to a severe accident.
- The passive inherent negative reactivity feedback, hydraulic, and thermal performance characteristics of the MARVEL design provide margin in many beyond-design-basis sequences to limit accident consequences without activation of active engineered systems or operator actions.
- These same passive characteristics will also provide margin for operational and design basis events.

The MARVEL hazard evaluation meets the intent of NEI-18-04 as follows:

- ***<u>Risk-Informed</u>:*** **Although a PRA was not performed, the MARVEL hazard evaluation provides a qualitative approach for risk assessment that provides reasonable assurance of protection of the public and worker.**

- ***<u>Performance-Based</u>:*** **The MARVEL hazard evaluation provides a simplified analysis of the performance requirements for the <u>passive</u> SSCs performing the FSFs necessary for preventing IEs from progressing into sequences that could result in core damage and radiological or non-radiological releases.**

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Thank-you

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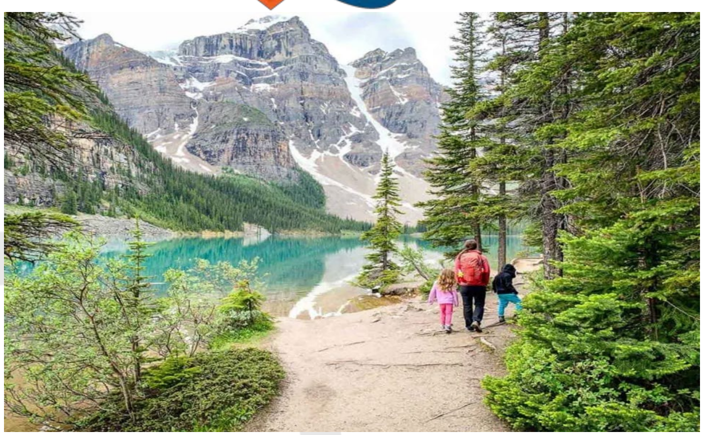

Questions?