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Technical and Licensing Considerations for Micro-Reactors

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

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico
87185 and Livermore,
California 94550

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) has interacted with vendors pursuing the commercialization of micro-reactors (i.e., reactors capable of producing about 1 MW(th) to 20 MW(th) of energy from nuclear fission). It is envisioned that micro-reactors could be assembled and fueled in a factory and shipped to a site. Many of the sites are expected to be remote locations requiring off-grid power or in some cases military bases. The objective of this effort is to explore the technical issues and the approach required to reach a finding of "reasonable assurance of public health and safety" for this new and different class of reactors. The analysis performed here leverages available micro-reactor design and testing data available from national laboratory experience as well as commercial design information to explore technical issues. Some factors considered include source term, accidents that would need to be analyzed, and the extent of the probabilistic risk assessment (PRA). The technical evaluation was performed within the framework of the Licensing Modernization Project (LMP) to identify licensing basis events, classification of structures, systems and components, and defense-in-depth needed to provide regulatory certainty. With this framework and technical evaluation in mind, the scope and content of a micro-reactor licensing application is discussed.

ACKNOWLEDGEMENTS

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ACRONYMS AND DEFINITIONS

Abbreviation	Definition
ACRS	Advisory Committee on Reactor Safeguards
AOOs	anticipated operational occurrence
APET	Accident Progression Event Tree
ASME	American Society of Mechanical Engineers
BDBE	Beyond Design Basis Event
BOP	Balance of Plant
BWR	Boiling water reactor
CANDU	Canada Deuterium Uranium
CASL	Consortium for Advanced Simulation of Light Water Reactors
CCS	canister containment subsystem
CFR	Code of Federal Regulations
COL	Combined license
CP	Construction permit
DBA	design basis Accident
DBE	design basis Event
DC	design certification
DCD	design control document
DHRS	decay heat removal system
DHX	decay heat exchanger
DID	Defense-in-Depth
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DUFF	Demonstration Using Flattop Fission
EAB	Exclusion Area Boundary
EBR-II	Experimental Breeder Reactor II
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute

Abbreviation	Definition
ESP	early site permit
F-C	Frequency-Consequence
FMEA	Failure modes and effects analyses
FOAK	first-of-a-kind
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
HALEU	High-Assay Low-Enriched Uranium
HEU	High-Enriched Uranium
HP	Heat Pipe
HPLs	Heat pipe limits
HPR	Heat pipe reactor
HTGC-PBR	High-temperature Gas-Cooled Pebble Bed Reactor
HTGRs	High-temperature gas-cooled reactors
IAEA	International Atomic Energy Agency
IDP	Integrated Decision-Making Panel
INL	Idaho National Laboratory
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
KRUSTY	Kilowatt Reactor Using Stirling Technology Demonstration
LANL	Los Alamos National Laboratory
LBE	Licensing Basis Event
LEU	Low-Enriched Uranium
LMP	Licensing Modernization Project
LOCA	Loss of Coolant Accident
LWR	Light water reactors
MSR	Molten salt reactors
MST	Mechanistic Source Term
NEPA	National Environmental Protection Act
NRC	U.S. Nuclear Regulatory Commission

Abbreviation	Definition
NSRST	Non-safety related with special treatment
OL	Operating license
PDC	Principal Design Criteria
PHX	Primary Heat Exchanger
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PRISM	Power Reactor Innovative Small Module
PSF	PRA Safety Functions
PWRs	Pressurized water reactors
RCCS	Reactor Cavity Cooling System
RFDCs	Required functional design criteria
RSF	Required Safety Function
SCRAM	Sudden shutting down of a nuclear reactor
SDA	Standard design approval
SFR	Sodium-cooled Fast Reactor
SRP	Standard review plan
SSC	Structures, Systems, and Components
TOP	Transient Overpower
VERA	Virtual Environmental for Reactor Application
VVERs	Water-water Energetic Reactor
VVUQ	Verification, Validation and Uncertainty Quantification

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1. INTRODUCTION

The idea of a small, transportable nuclear power plant for remote applications is not a new one. This is illustrated in Figure 1-1 which reproduces the cover of a 1968 primer (Reference [1]) on the topic. While micro-reactors of the 1960s required dozens of shipping containers full of components and months to assemble them at the operating site, modern designs have been proposed which reduce the shipping and on-site assembly requirements [2]. The United States Department of Energy (DOE), Office of Nuclear Energy, defines micro-reactors as "plug-and-play reactors able to produce 1-20 megawatts of thermal energy used directly as heat or converted to electric power". Micro-reactor designs are proposed to be factory-built with no fuel handling capabilities at the operating site. They are designed to run for an extended period with no refueling before being returned to a factory for refurbishment or directly entered into interim or permanent storage. They are generally proposed to be transportable with installation and removal taking a fraction of the time currently required for traditional power reactors. Finally, they propose less operator interaction than traditional light water reactors (LWRs).

All current micro-reactor designs (2012 forward) allow for black start with no connection to a larger electric grid. They are proposed as solutions for sites with limited grid access that currently rely on diesel generators for power, such as mining operations, military bases, or isolated towns (e.g., in remote parts of Alaska). This analysis will be limited to applications where the NRC would have licensing authority and thus may not encompass all micro-reactor applications and designs (i.e., forward/remote operating bases) [3]. Many current micro-reactor designs use heat pipes for transport of energy from the fuel to the power conversion system. While the general idea of a heat pipe reactor (HPR) is not new [4], historically they have typically been proposed for space use and only recently has one undergone kilowatt-scale testing [5]. Thus, there are still significant questions in proving the designs for commercial, terrestrial use.

1.1. Objective

Micro-reactor designs are in relatively early stages of development for commercialization in the United States. Beyond limited physical design similarities, some parallels to the development of sodium-cooled fast reactors (SFRs) may exist, as documented in References [6] and [7]. While traditional SFRs are not micro-reactors, a number of proposals consider utilizing liquid metal for the heat transport fluid. In addition, the fuel designs proposed for different micro-reactor concepts are significantly different from the traditional fuel used in water-moderated reactors. In this sense, micro-reactors and SFRs both exhibit a similar degree of extension of the technology beyond the experience base of the civilian nuclear program in the United States. In light of this, some sources of uncertainty relevant to design, licensing, and deployment of micro-reactor concepts may benefit from insights identified during the development of the Power Reactor Innovative Small Module (PRISM) reactor design in 1994. Examples of specific insights/concerns from the PRISM development are as follows [8]:

- limited performance and reliability data for passive safety features,
- lack of final design information,
- unverified analytical tools used to predict plant response,
- limited supporting technology and research,

- limited construction and operating experience, and
- incomplete information on the proposed fuel.

However, from the perspective of a reactor system, micro-reactors are fundamentally different from proposed SFRs in many of the same ways that they differ from traditional LWR systems. A fundamental difference between micro-reactors and SFRs or LWRs is that a micro-reactor safety design often implements safety functions through inherent safety measures (similar to the passive safety systems introduced for Generation III+ LWRs). Safety systems designed around the principle of inherent safety utilize naturally occurring physical processes to achieve critical safety functions, such as reactivity control or decay heat removal.

This focus on inherent safety, providing in many situations a single passive safety system to implement a safety function, challenges traditional methods of evaluating the level of safety in a reactor design. For example, how can the reliability of a passive safety system be established to quantify the risk from operation of a micro-reactor? This fundamental technical challenge must be resolved to enable the NRC to provide reasonable assurance that a micro-reactor design does not pose undue risk to public health and safety.

This report addresses the details of these challenges and other potential gaps as they relate to the licensing review of micro-reactors under the Licensing Modernization Project (LMP). While a motivation for this report was to develop insights on how implementation of the LMP for typical micro-reactor concepts may need additional special treatments or methods, the discussion of many of the topics in this report are more broadly applicable and of relevance to micro-reactor applicants not following the LMP. In particular, this report identifies a number of key areas where development of the safety basis for a micro-reactor concept requires an evolution of traditional safety analysis and risk assessment methods. In this sense, this report focuses on assessing areas in which the conduct and documentation of a traditional safety case for a micro-reactor requires some further specialization.

1.2. Report Overview

Section 2 of this report presents some prominent micro-reactor designs and their unique features. The designs are distilled into two general concepts which are expected to have different technical questions in the licensing review process.

Section 3 provides an overview of the LMP, which is an industry proposal for a risk-informed, performance-based, technology-inclusive process for [9]

- Selecting Licensing Basis Events (LBEs),
- Performing safety classification of structures, systems and components (SSCs) and associated risk-informed special treatments, and
- Determining the adequacy of Defense-in-Depth (DID).

NOTE: The overall goal of the LMP is to provide one means by which applicants can demonstrate that a specific design provides "reasonable assurance of radiological protection" [9].

Section 4 examines issues associated with the application of the LMP to micro-reactors. More broadly, however, it focuses on areas of safety analysis and risk assessment methodology not typically encountered in the more extensive application to the current fleet of LWRs in the United States. As the LMP is a very significant risk-informed application, its implementation in the preparation of a risk-informed, performance-based demonstration of adequate protection is dependent on the appropriateness of risk assessment methods for the unique challenges posed by the technology under evaluation. Since the LMP process has not yet been implemented in detail for non-LWRs under consideration, its applicability is generally assumed based on overall similarities with the current fleet of LWRs. By contrast, micro-reactors operate at much lower thermal power, and, as an integrated, engineered system, are significantly less complex than large-scale power reactors. The extent to which traditional risk assessment methods are appropriate for assessing micro-reactors is thus assessed in Section 4.

The roadmap and associated challenges of micro-reactor licensing using the existing reactor licensing standard review plan (SRP) is discussed in Section 5. Since the SRP has been developed for LWRs, it is not the only structure by which the safety basis for a non-LWR design could be documented. However, given the familiarity with the SRP, this section presents an approach to documenting micro-reactor safety basis that is as similar to the existing structure for LWRs as possible. Section 6 summarizes the analysis and presents conclusions relating to the application of the LMP to micro-reactors.

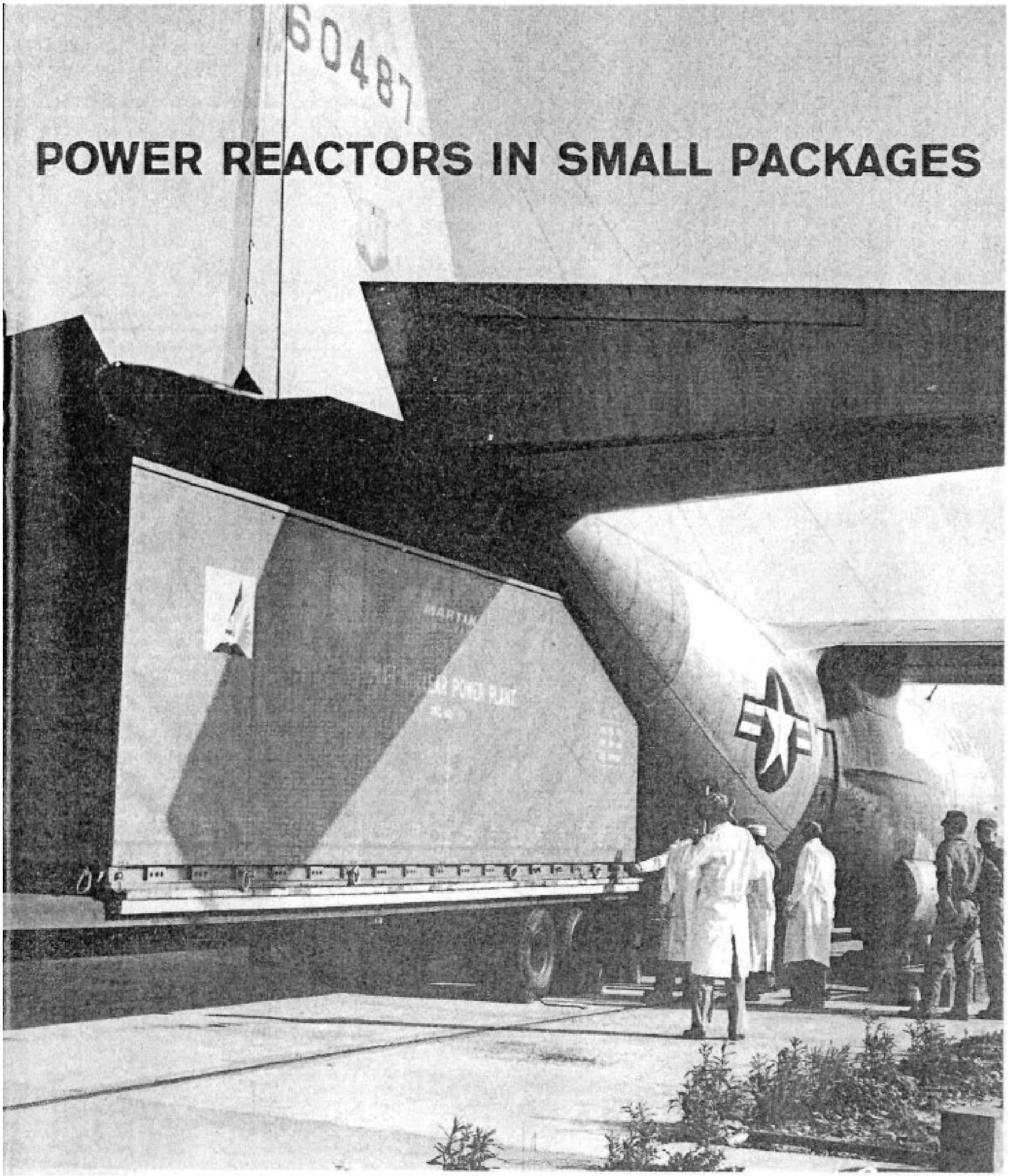

Figure 1-1. Micro-Reactor, circa 1968 [1]

2. MICRO-REACTOR DESIGNS

This section provides an overview of the publicly available design information on multiple national laboratory and commercial micro-reactors. There is considerably more information available on the national laboratory designs than the commercial designs. The reactors included are:

- Kilopower, Los Alamos National Laboratory (LANL)
- Megapower, LANL
- eVinci, Westinghouse Electric Company
- Oklo, Oklo Inc.

2.1. Kilopower

The Kilopower (or KiloPower) reactor was designed at LANL as a small (originally 4 kWth, 1 kWe) space reactor system to power deep space missions [10]. Kilopower uses a small bare cylinder of high-enriched uranium (HEU) alloyed with 7% Mo as shown in Figure 2-1. A central B_4C rod provides startup control while BeO reflectors provide necessary reactivity worth for the fast neutron reactor. Studies were performed to evaluate core materials [10] as well as the enrichment level of the uranium fuel [11] [12]. Once the startup rod is removed to its running position, the reactor is regulated by inherent reactivity feedbacks.

The heat generated within the fissile material of the Kilopower design is transported to an ultimate heat sink through a conductive coupling to heat pipes. A heat pipe is an effective heat transport device, capable of moving thermal energy generated at high rates over large distances with small cross-sectional heat transport area. As such, the effective thermal conductivity of a heat pipe significantly exceeds that of metallic materials. A heat pipe is a sealed container (e.g., a cylindrical tube) consisting of pipe walls and end caps. In the tube, a small amount of working fluid is placed. The working fluid is in equilibrium with its own vapor. In operation, three distinct regions occur in the heat pipe tube: (1) the evaporator section in which the working fluid resides and where heat is applied to the working fluid to vaporize it; (2) the adiabatic section through which vaporized working fluid is driven by its vapor pressure; and (3) the condenser section where the heat pipe is thermally coupled to an ultimate heat sink to which the thermal energy is transported. In the condenser section, the vapor is condensed as a result of thermal energy transport to the ultimate heat sink. The condensed vapor flows down to the evaporator section as a thin film along the heat pipe wall. The forces between the thin liquid film and the wall of the pipe creates a capillary pressure, that represents the pressure between the vapor and the condensed liquid. The interaction forces between the liquid and solid pipe wall promote adherence of the liquid to the wall (i.e., the wall is "wettable" by liquid working fluid). The resulting capillary pressure acts to drive motion of the liquid working fluid down the wall of the pipe, from the condenser section to the evaporator section. In this manner, condensed fluid is recirculated back to the evaporator section, achieving a closed-loop heat transport process. Due to the action of the capillary pressure, condensed fluid passively recirculates to the evaporator without the need for any active pumping of the fluid.
The amount of thermal energy that can be transported away from the fissile material depends on the amount of heat being removed at the condenser section of the heat pipe. Should more energy be transported away from the condenser section, the fissile material would be cooled more effectively and reach lower temperatures. At lower fuel temperatures, fuel temperature reactivity feedback in Kilopower is such that an increase in reactivity would occur. This results in higher fission power

production capable of accommodating the energy removal at the condenser section of the heat pipe. Thus, the Kilopower design is able to effectively accommodate load-following applications, which is often found across different HPR concepts. One space application mates the Kilopower reactor with a Stirling engine as the ultimate heat sink. The Stirling engine converts thermal energy removed from the condenser section to mechanical energy.

NOTE: A Stirling engine converts thermal energy to mechanical energy utilizing a cyclic compression and expansion of a gaseous working fluid. The working fluid exists at different temperatures in different sections of the Stirling engine. Different configurations exist for the Stirling engine; however, it can be generically defined in terms of a governing thermodynamic process the working fluid experiences. In the first stage of the process, the working fluid undergoes an isothermal expansion as the gas absorbs heat from the heat source. In the second stage, the expanded gas passes through a portion of the engine to which it rejects energy. This region of the Stirling engine is known as the regenerator. This portion of the Stirling engine serves as a temporary heat sink. During this stage of the process, the gas stays at constant volume. This second stage is an isochoric heat removal. In the third stage, the gas enters a compression space with a heat exchanger maintained at constant temperature, which removes energy from the gas and cools it. The gas undergoes an isothermal compression as it rejects energy to the heat sink. In the fourth stage, the compressed gas passes back through the regenerator where it recovers energy it lost in the second stage. This fourth stage heat removal occurs at constant volume, representing an isochoric heat addition. These stages of expansion and compression of the gas are used in a Stirling engine to displace a mechanical piston, converting thermal energy to mechanical energy.

Kilopower uses sodium-filled heat pipes to remove heat from the reactor and drive Stirling engines which in turn produce electricity as shown in Figure 2-2. A radiator, not shown in Figure 2-2, is thermally bonded to the opposite ends of the Stirling engines from the heat pipes and is used to reject heat into space. Figure 2-2 also shows how the heat pipes are thermally bonded to the bare fuel cylinder. Note that Figure 2-2 does not include the radial neutron reflectors so that greater detail of the core may be shown.

The Demonstration Using Flattop Fission (DUFF) experiment helped to prove the concept of using heat pipes and Stirling engines to generate electricity from fission heat [13] [14]. The experimental setup is shown in Figure 2-3. Fission heat was provided using the LANL Flattop critical assembly.

An additional series of tests called the Kilowatt Reactor Using Stirling Technology Demonstration (KRUSTY) was conducted using the LANL Comet critical test assembly [15] [5]. The experimental setup is shown in Figure 2-4. The upper section of the Kilopower assembly, including the Stirling engines, was placed within a vacuum chamber to simulate operation in space. In some experiments, electrical heating was used. In later experiments, the moving platen of the Comet assembly was used to move an HEU core into a critical configuration to produce heat.

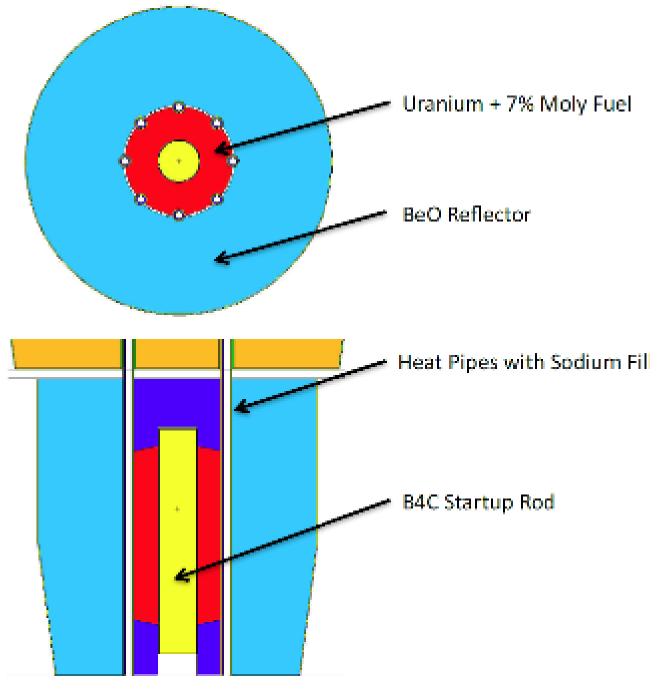

Figure 2-1. Kilopower Reactor Schematic [10]

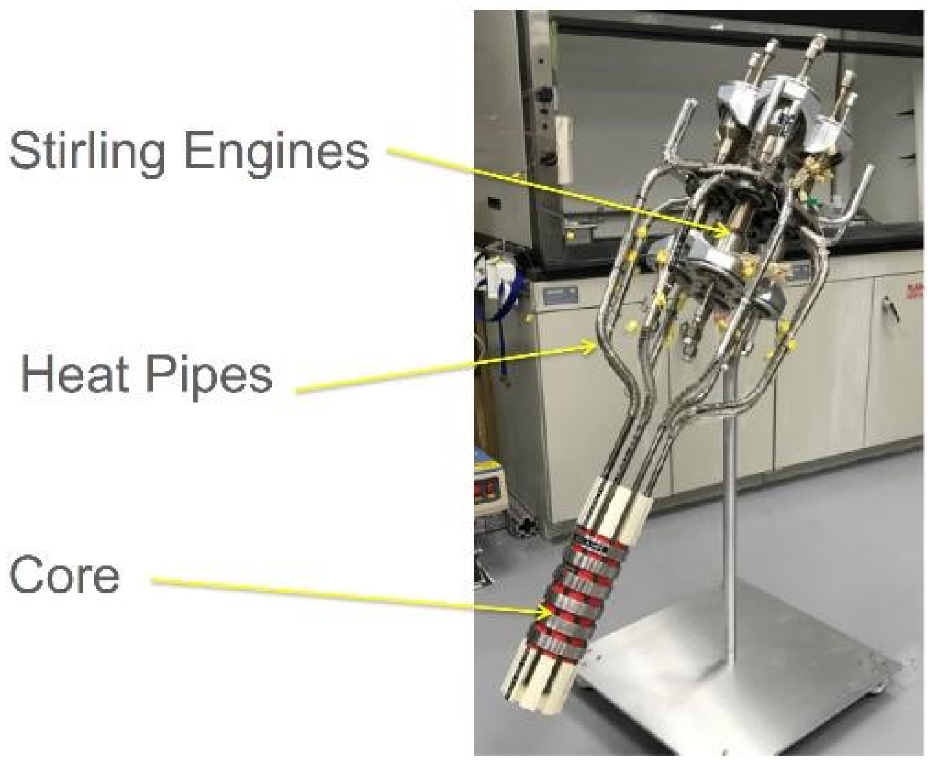

Figure 2-2. Kilopower Assembly [16]

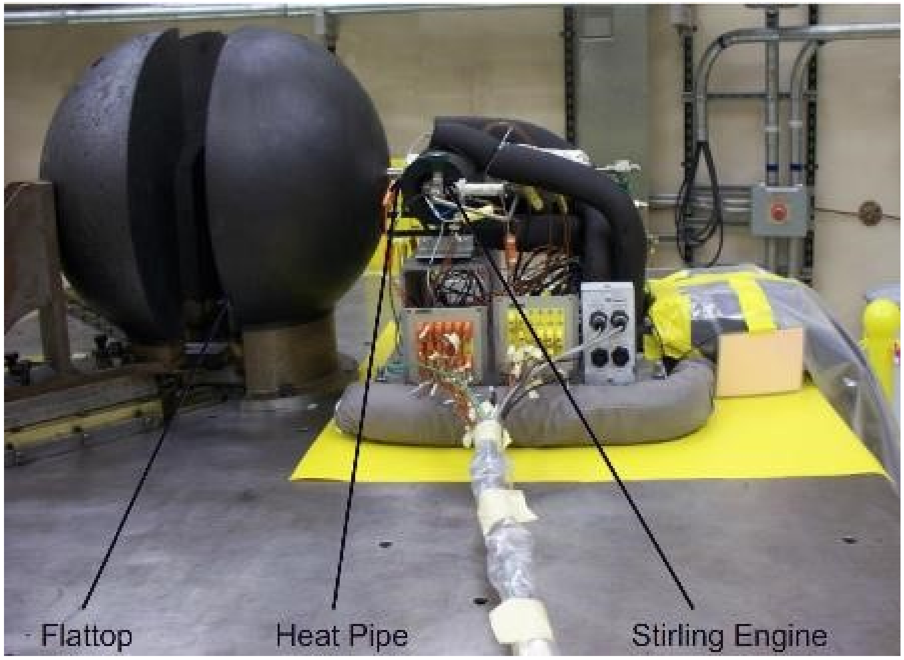

Figure 2-3. DUFF Experimental Setup [16]

The Kilopower design was analyzed to evaluate its safety during launch [17] [18]. The only radioactive content of the assembly before use is the un-irradiated HEU fuel cylinder, and so the launch safety analysis focused on the potential for criticality.

The Kilopower design is not considered to be directly applicable to the present study of the technical issues for licensing commercial micro-reactors for the following reasons:

- It uses HEU which is very tightly regulated and uncommon in non-government facilities. Other micro-reactor designs use High-Assay Low-Enriched Uranium (HALEU) or Low-Enriched Uranium (LEU) which are more easily available for commercial facilities.

- The core is bare U-Mo alloy with no provisions for containment of fission products.

- The power level is so low that other alternatives are likely to be more attractive for terrestrial energy systems.

Despite these critical differences from expected systems, the research provides a technical basis and demonstration of the basic concept of a fissile material acting as the heat source for a heat pipe, serving to transport energy to an ultimate heat sink. The Stirling engine utilized in the tests provides one example of an ultimate heat sink that converts thermal energy to mechanical energy. Alternate energy conversion systems could be utilized, as this is not germane to the overall concept of a HPR.

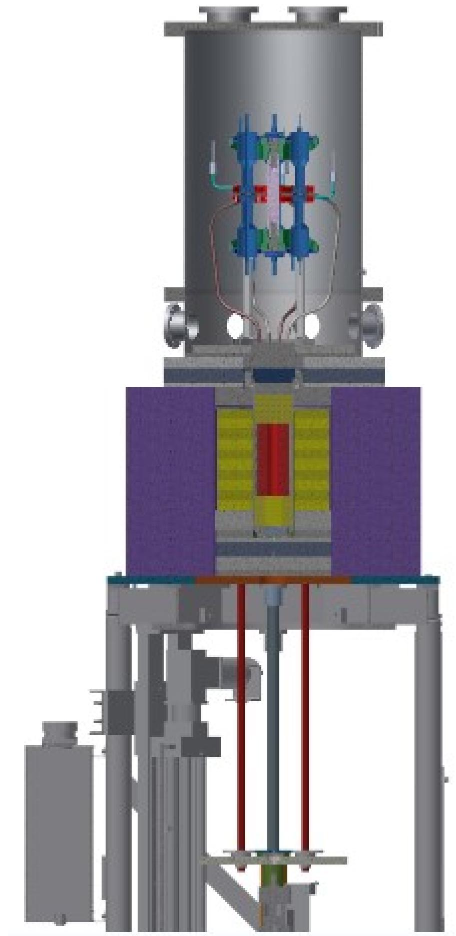

Figure 2-4. KRUSTY Experimental Setup [15]

2.2. Megapower

The Megapower design is also being developed at LANL but differs significantly from the Kilopower design [19]. As with Kilopower, it was originally designed for space reactor applications (see Reference [20]). It is a 5 MW_{th} compact fast reactor using HALEU UO_2 fuel pellets with 19.75% enrichment and potassium-filled heat pipes operating at 675°C. Pellets and helium (acting as the thermal bonding medium) are added to each fuel hole which is in turn sealed by welding on an end cap. A recent report suggests that drop-in fuel elements and heat pipes have been considered in lieu of the original integral approach [21]. The system is intended to run for five years before being refueled or stored.

A diagram of the system is shown in Figure 2-5. Six monolithic stainless-steel core sections are bonded together to form a hexagonal monolith. Fuel and heat pipes are arranged in a hexagonal pattern within holes in the monolith. The fuel is fully contained within the monolith. The primary heat exchanger (PHX) consists of the condenser sections of the heat pipes covered by an annular pipe, similar to the eVinci[1] design (see Section 2.3). Control drums are embedded within the alumina radial reflector. A B_4C rod is used in the center of the monolith to start and shutdown the reaction.

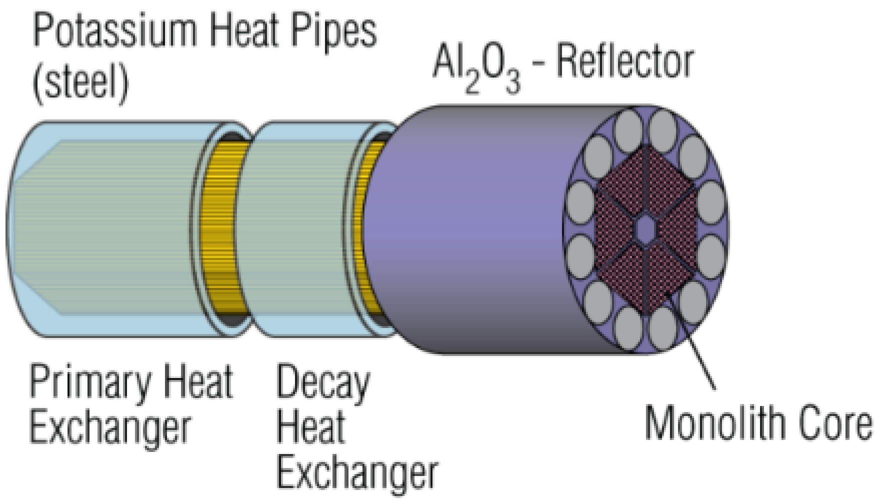

Figure 2-5. Megapower Reactor Concept [22]

An initial analysis was performed by LANL to evaluate the temperature effects of a failed heat pipe [22]. Figure 2-6 shows the distributions of temperatures in the case of a single failed heat pipe while Figure 2-7 shows the strain in the stainless steel monolith for the same case. The core is designed to stay within the American Society of Mechanical Engineers (ASME) thermal stress limits with the failure of two heat pipes adjacent to the same fuel pin. The designers acknowledge that manufacturing the monolith would be a significant engineering challenge for this reactor design [22]. Manufacturability is the main factor that led to the proposal of six core sections bonded together rather than a single machined block.

[1] Los Alamos and Westinghouse have partnered to further develop the Megapower concept under the eVinci name. [68]

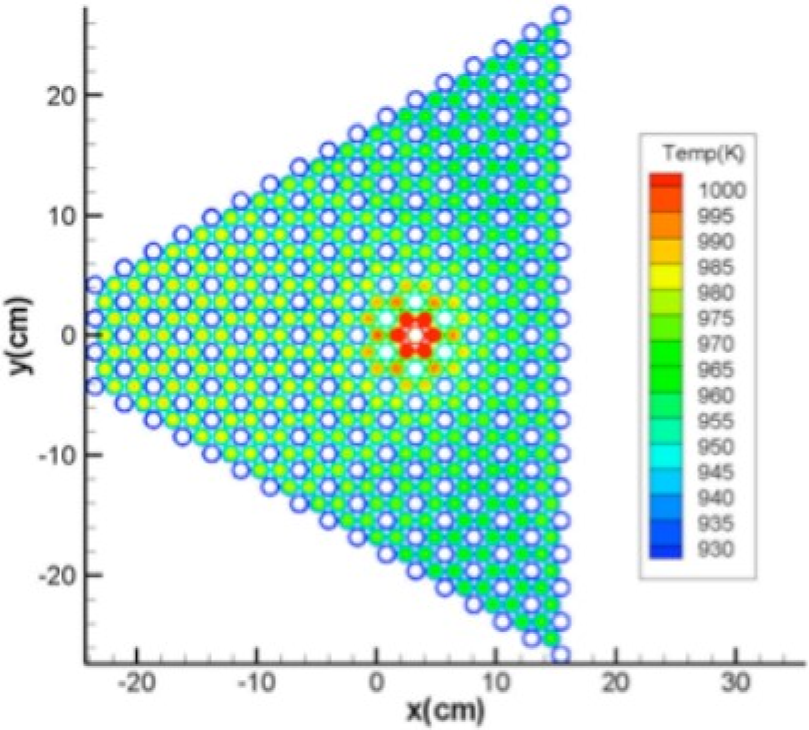

Figure 2-6. Temperature Distribution for Megapower Failed Heat Pipe [22]

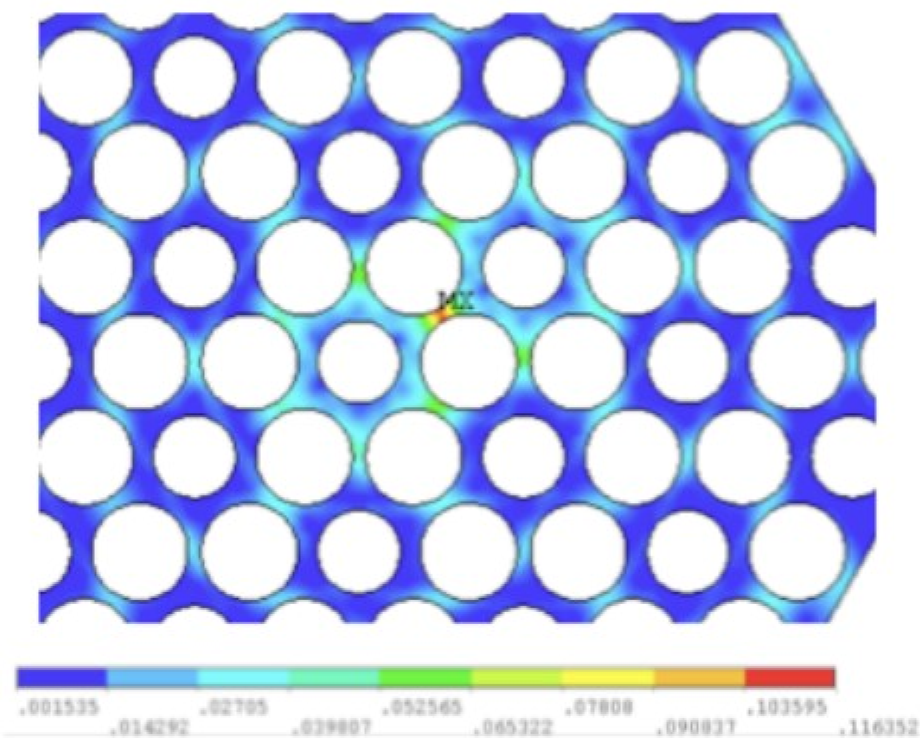

Figure 2-7. Strain Distribution for Megapower Failed Heat Pipe [22]

Idaho National Laboratory (INL) performed a phenomena identification and ranking table (PIRT) analysis of the Megapower design, making judgments for features that the design did not fully address [23]. A central focus of the PIRT was the geometry of the monolith. It is approximately 1.5 m long with over 3,000 holes for fuel and heat pipes. Spacing is tight (1.75 mm between fuel holes and 1 mm between heat pipe and fuel holes) to achieve sufficient reactivity worth and reduce thermal gradients. The geometry presents challenges for manufacturing, inspecting, and its ability to tolerate transients. The horizontal orientation of the core was also of concern to the INL team due to the potential for unbalanced stresses across the monolith.

The wall of the evaporator section of each heat pipe is the stainless-steel wall of its hole in the monolith. This means that the rest of the heat pipe is bonded to the monolith and loaded with the wick and working fluid after attachment to the monolith. With the tight spacing between heat pipes and fuel hole end caps, this may be difficult to manufacture and inspect. Note that a recent report suggests that drop-in fuel elements and heat pipes have been considered in lieu of the original integral approach and may be preferred for the eVinci design [21].

Analyses performed for the PIRT indicate that a single failed heat pipe could result in monolith thermal stresses in excess of the ASME pressure vessel limits. This has the potential to result in a failure between a fuel hole and a heat pipe. The PIRT also cites a lack of defense in depth which would allow such a failure to introduce fission products to the heat pipe. A subsequent heat pipe failure in the condenser section would release radionuclides from the system.

The PIRT focused on the following areas:

- Reactor Accident and Normal Operations
- Heat Pipe
- Materials
- Power Conversion Unit (assumed)

Table 2-1 shows the issues identified in the PIRT that have both high importance and a low knowledge level. In Table 2-1, heat pipe is abbreviated HP.

Table 2-1. PIRT Issues/Phenomena with High Importance and Low Knowledge Level [23]

Issue/Phenomena	Comments
Defense In Depth (DID)	System satisfies single failure criterion, but does not sufficiently address a DID approach.
Stainless steel monolith web failure between HP-fuel channel	Fission product transport from fuel channel thru monolith into HP.
Loss of weld integrity between HP and monolith	HPs are welded to the top of the monolith block to form the complete HP. Welding techniques and Quality Assurance (QA) measures are essential to achieve high levels of confidence in heat pipe integrity.
Inability to weld and inspect coupling	Welding of HP to monolith with very limited physical access is not well understood, nor is inspection and testing.
Machining	A very large number of holes with very tight tolerances are required in the monolithic blocks if conventional wrought products are selected. Stainless steels are difficult to machine and drill.
Welding	Careful design could be required to eliminate, or minimize, the number of welds in high temperature and high stress regions.
Inspection	With the complex geometry and high number of interfaces, it is not clear that methods are available for the entire assembly.
Structural design	ASME Code design rules applicable for anticipated design conditions for this reactor have not been vetted by the NRC.
Structural weight of the core	The ability of the lower monolith webs and sector steel to support the weight of the fuel and the balance of the sector/core mass above is uncertain, especially at operating temperature, where the SS monolith may have reduced strength.
Seismic event	The additional forces due to a seismic event may over-stress the core structure.
Catastrophic turbine failure	Turbine blades are damaged to the point of generating shrapnel which damages the heat pipes, releasing radioactive materials to the atmosphere.
Heat exchanger catastrophic failure	Heat exchanger no longer functions and heat from the heat pipes cannot exchange with the air. The heat pipes are embedded within the heat exchange, so alternate heat removal may not be possible.

The eVinci design (see Section 2.3) is an evolution of Megapower and so many of the concerns INL identified for the Megapower design could also be relevant to eVinci. The INL team later proposed two alternative Megapower core designs, both in a vertical orientation and both intended to eliminate significant issues identified with the LANL Megapower design while maintaining similar dimensions and other parameters [24].

In the first INL core alternative, designated Design A, the heat pipes are manufactured as separate units and later integrated into a fuel element as shown in Figure 2-8. Compared to the LANL design, this allows inspection and testing of individual heat pipes before they are integrated into the core.

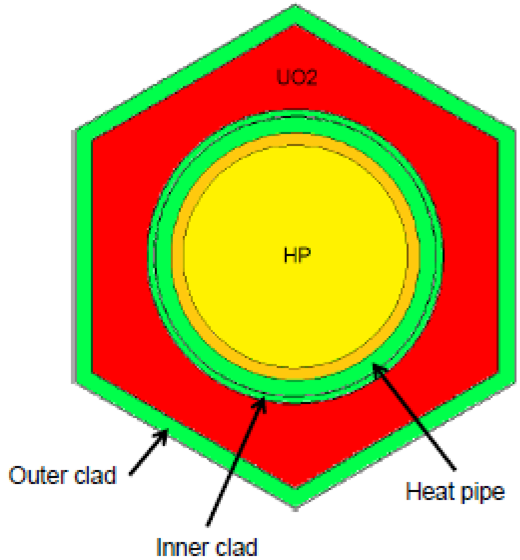

Figure 2-8. Cross-section of Design A Fuel Element [24]

Figure 2-9 shows the proposed core layout for Design A which no longer includes a stainless-steel monolith but uses a core barrel to hold the fuel elements in place. This design presents its own challenges but resolves many of the concerns relating to manufacturability and stresses in the core. The heat exchanger and reactivity control systems are relatively unchanged from the LANL Megapower design.

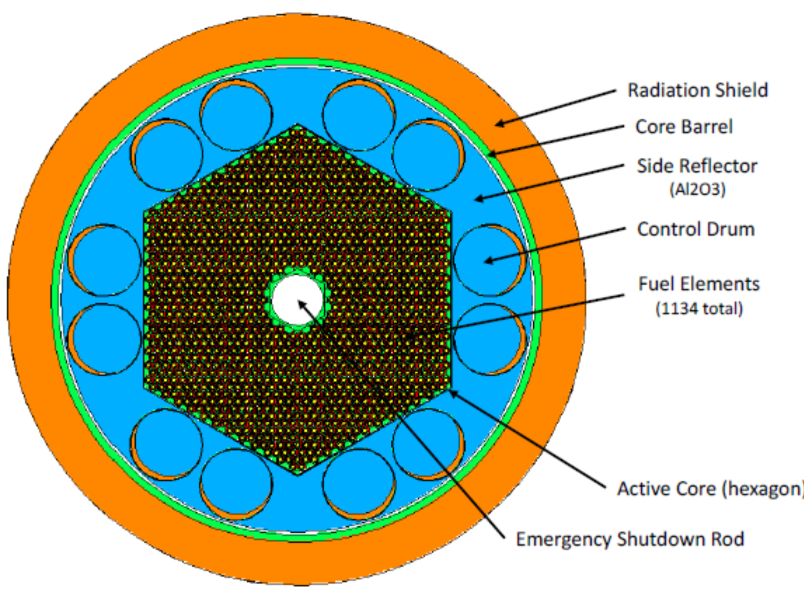

Figure 2-9. Cross-section of Design A Core Layout [24]

INL core alternative Design B uses cylindrical heat pipes and fuel pins. The heat pipes and fuel pins are arranged in six wedges to form a hexagonal grid as in the LANL design (see Figure 2-10). In Design B, however, the space between the elements is filled with liquid sodium rather than stainless steel. Fuel pins and heat pipes are manufactured and inspected separately and do not need to be integrated as they do in Design A.

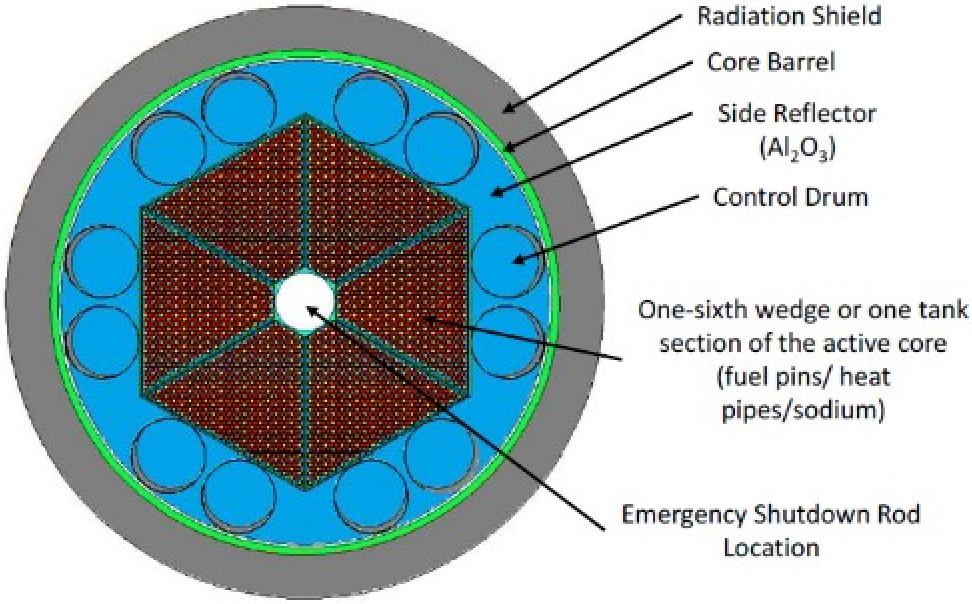

Figure 2-10. Cross-section of Design B Core Layout [24]

Each wedge of the Design B core is a double-walled stainless steel tank, which fully contains the fuel pins while allowing the heat pipes to emerge through welded holes (see Figure 2-11). Design B contains the same number of fuel elements and heat pipes as the LANL Megapower design albeit with slightly larger diameter pellets and a larger pitch between them. Bonding the heat pipes and fuel elements with liquid sodium rather than stainless steel increases heat transfer which allows for smaller temperature gradients both during normal operation and transients such as a heat pipe failure. There is also likely to be less stress on the core structures (a collection of thin plates) as they maintain more uniform temperatures than the monolith. INL did not present an analysis of the performance of the tank concept during seismic loading.

The LANL Megapower design is considered a representative micro-reactor for the purposes of this report for the following reasons:

- Significant design information is available from both LANL and INL.
- Thermal, structural, and neutronic analyses as well as a PIRT have already been performed.

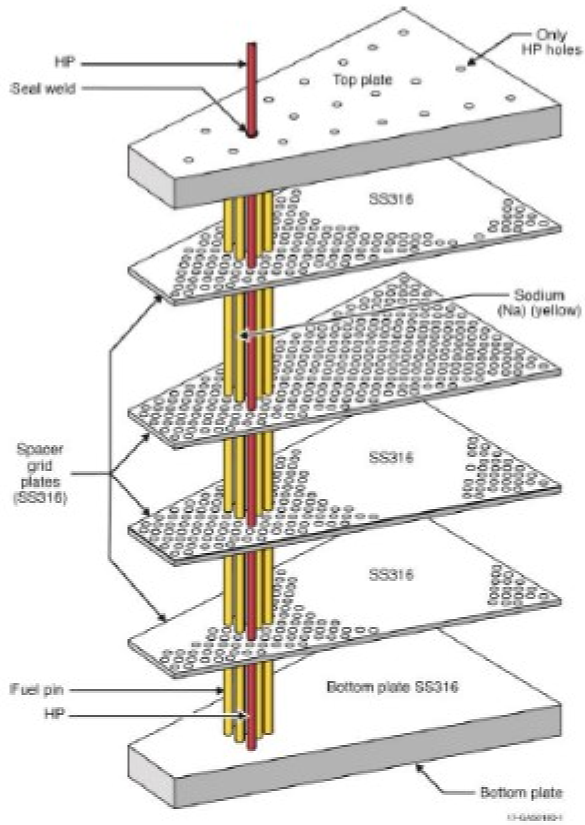

Figure 2-11. Design B Fuel Pins and Heat Pipes in Inner Tank [24]

2.3. Oklo

The Oklo reactor design is a compact fast spectrum reactor of approximately 2 MW_{th} [25], which is intended to be transportable and deployable to remote locations. It uses HALEU and is expected to have a 20-year refueling interval [26]. A potential general core layout is found in an Oklo patent filing [27] as shown in Figure 2-12. The conceptual design of the hexagonal lattice structure is shown in Figure 2-13. The Oklo design uses a hexagonal fuel element with a void for a heat pipe in the center. INL produced fuel performance analyses and prototype fuel elements for Oklo, describing the fuel element design as "innovative but unorthodox" [21]. This fuel design bears some similarity to the INL Megapower alternative Design A (see Section 2.2, Figure 2-8), albeit with a different proposed fuel material.

The Oklo design uses a U-10Zr metallic fuel form with stainless steel cladding, which was qualified as a driver fuel in both the Experimental Breeder Reactor II (EBR-II) and the Fast Flux Test Facility [28]. The elements will be kept in a stable geometry by one or more grid plates [29]. The Oklo fuel design is intended to have low burnup and a fuel and cladding temperature regime covered by existing fuel performance databases [30]. The Oklo reactor is oriented vertically with heat pipes extending above the core to the PHX [31] as shown in Figure 2-14.

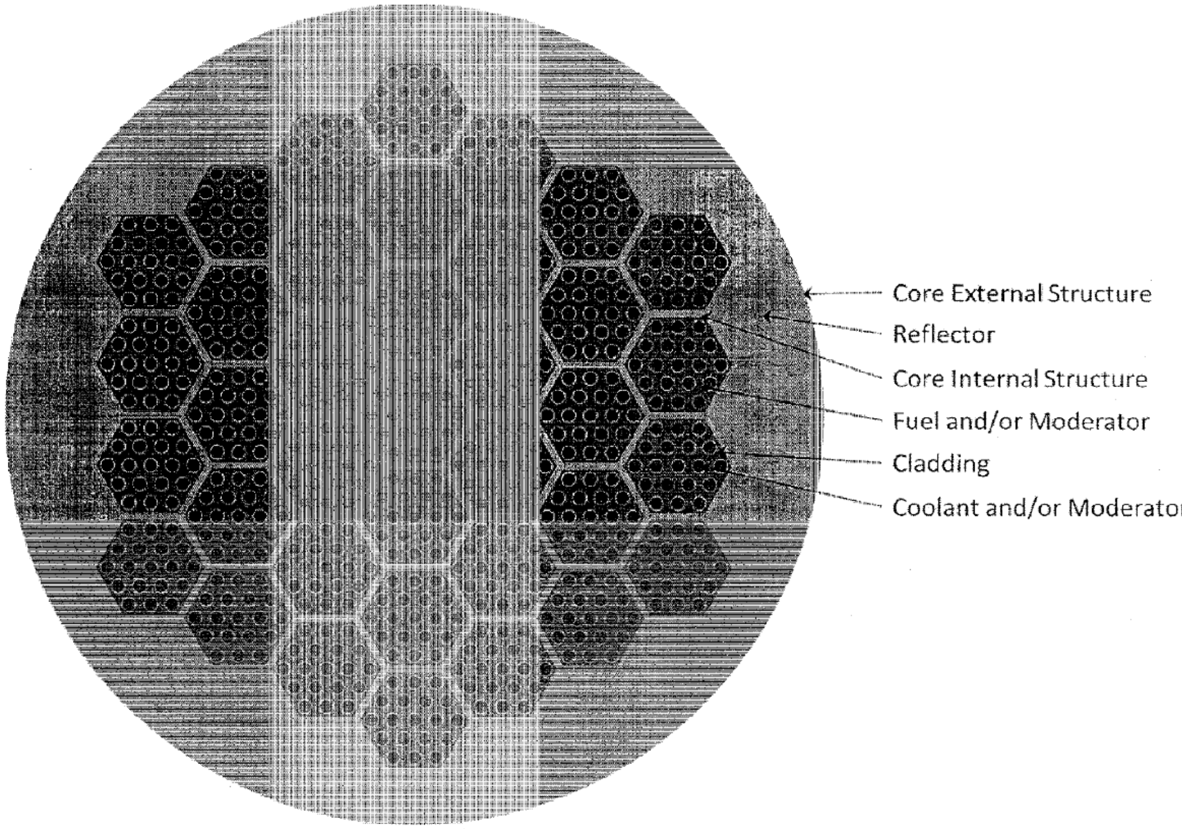

Figure 2-12. Potential Oklo Core Layout [27]

Oklo submitted a Principal Design Criteria (PDC) report for NRC Staff review, which was later withdrawn. Oklo has also submitted a report documenting a pilot application of the DG-1353 framework [29]. The NRC has issued comments for both the PDC [32] and DG-1353 pilot reports [33]. The NRC Staff comments provide further insight into the Oklo micro-reactor safety approach.

The heat pipes are sealed, independent heat transport devices. Oklo states the passive air-cooling system will remove the decay heat [29]. Volume 1 (Draft) of the NRC non-LWR regulatory strategy [34] describing the NRC approach for computer code modeling of design basis events (DBEs), provides a description of the Oklo decay heat removal system (DHRS). This system is called the reactor cavity cooling system (RCCS) and removes heat from the reactor enclosure without any requirements for penetrations from external heat removal systems.

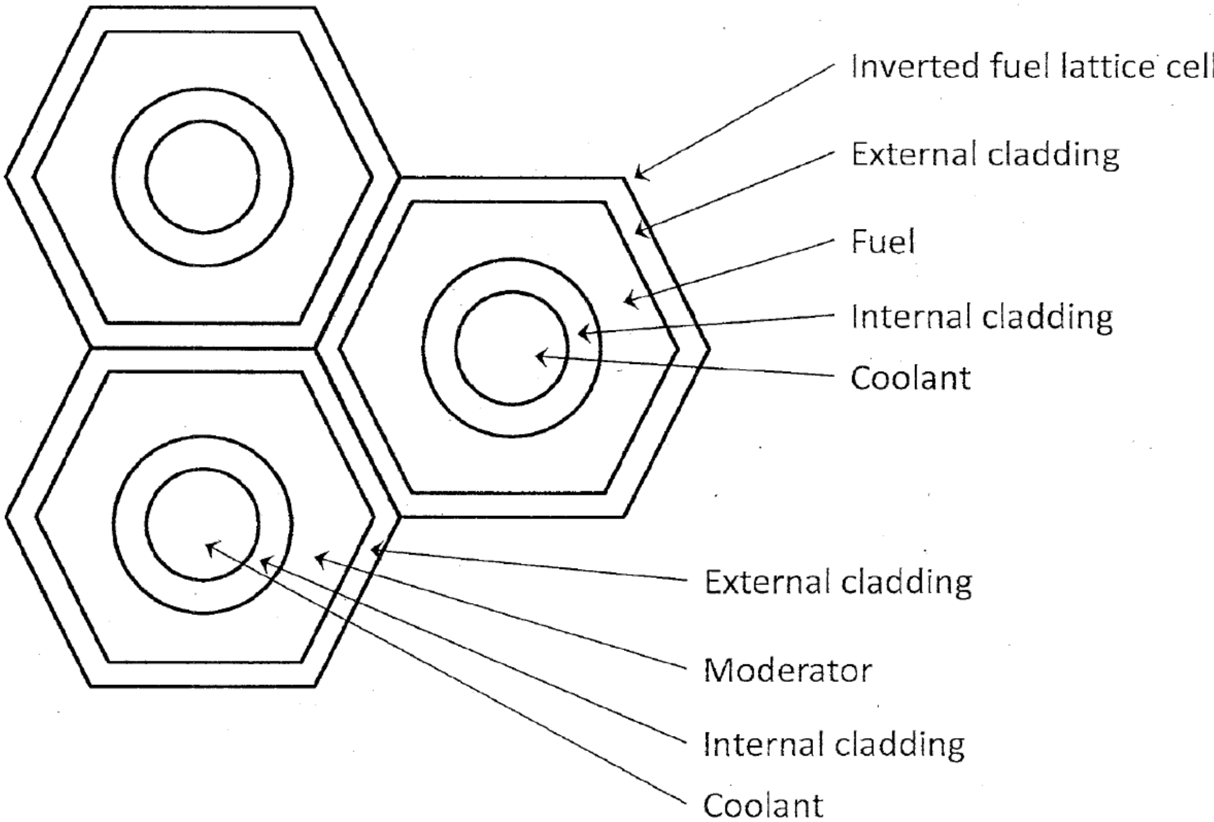

Figure 2-13. Potential Oklo Lattice Design [27]

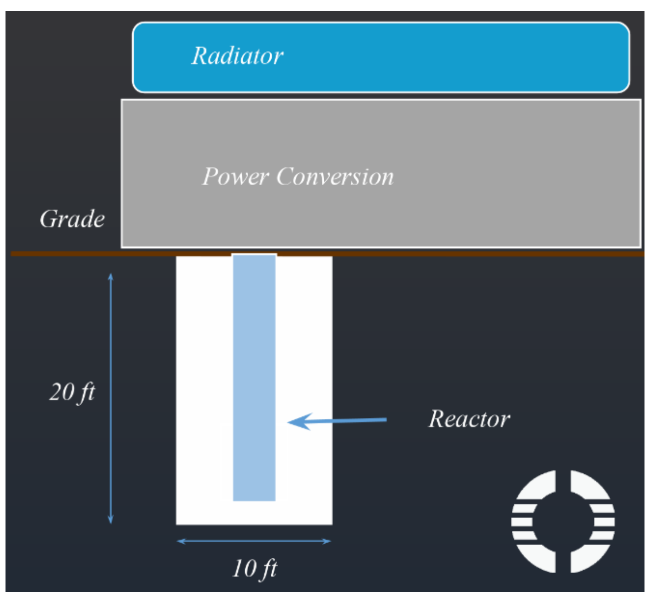

Figure 2-14. Oklo Reactor Layout [35]

2.4. eVinci

The eVinci micro-reactor design (see Figure 2-15) is being developed at Westinghouse (in collaboration with LANL) with power configurations ranging from 0.2 to 15 MW_e and the potential for autonomous operation [36] [37]. The system is intended for factory fabrication and a lifetime of 10 years after which the entire package would be returned to the factory for either refueling or storage. The reactor uses HALEU uranium dioxide fuel with 19.75% enrichment and operates in the epithermal spectrum. The moderator is a metal hydride that dissociates at high temperatures to release hydrogen into the moderator channels. The passive creation of hydrogen reduces the overall reactivity to reduce power [36].

The reactor core is depicted in cross-section in Figure 2-16. Fuel, moderator, and heat pipes are arranged in a hexagonal pattern in a steel monolith. It is unknown whether the integral fuel elements and heat pipes or drop-in elements are used [21]. Control drums are embedded within the radial reflector section. A central rod is used to shut down and start up the reactor. The eVinci designers take some credit for load-following behavior but primary power control is achieved via the rotating control drums.

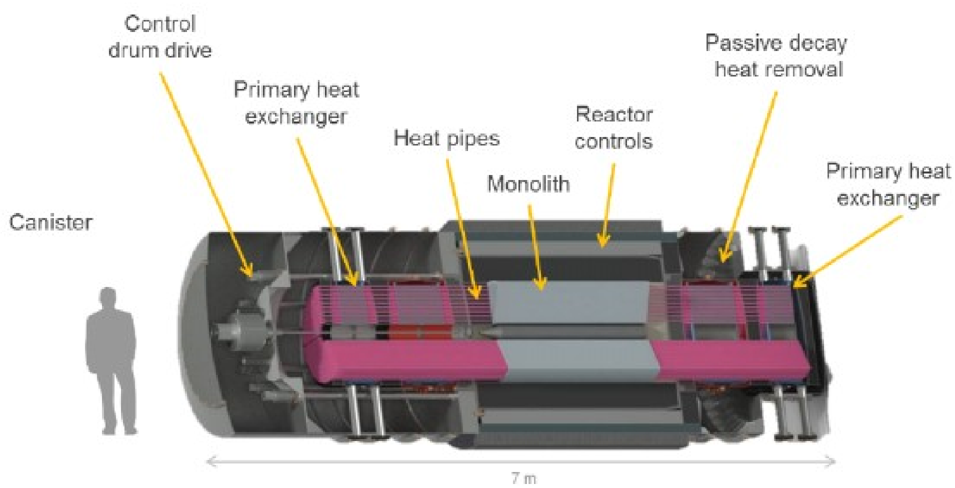

Figure 2-15. eVinci System Package [36]

The eVinci reactor concept utilizes sodium as the heat pipe working fluid [38]. The heat pipes protrude from both ends of the core, allowing for two PHXs. Each PHX consists of an annular tube over the condenser section of the heat pipes at its end of the package. Inlet and outlet plenums for each PHX will penetrate the stainless-steel package. Power conversion is achieved by using a supercritical carbon dioxide (sCO_2) Brayton turbine at 600°C or by using Stirling engines.

NOTE: A Brayton turbine is an energy conversion device utilizing a Brayton cycle. This cycle has the following stages. In the first stage, compression of the working fluid occurs adiabatically (i.e., at heat does not enter or leave the system). In the second stage, heat addition occurs under isobaric conditions (constant pressure). In the third stage, the working fluid expand adiabatically, while in the fourth stage heat rejection occurs under isobaric conditions. In this process of compression and expansion, the gaseous working fluid is able to drive a turbine, converting the thermal energy added in the second stage of the process to mechanical energy in the fourth stage.

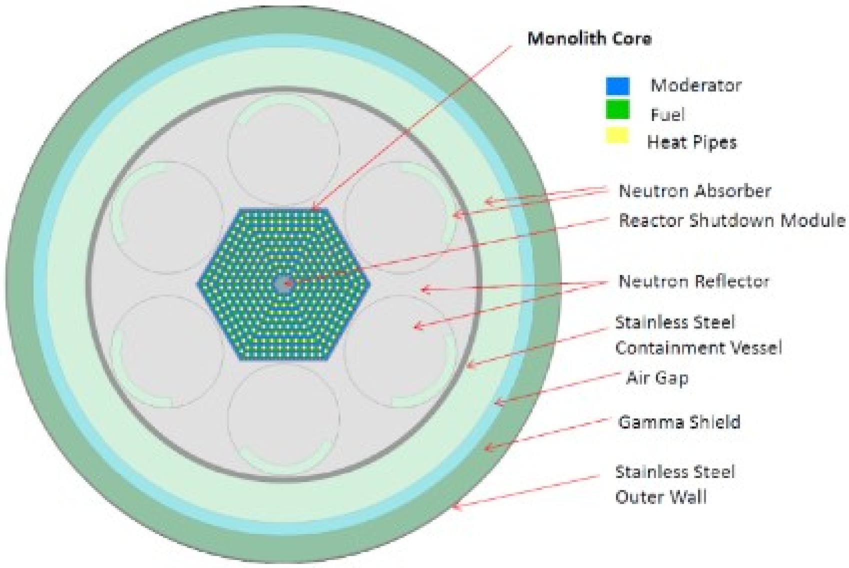

Figure 2-16. eVinci Reactor Cross-Section [36]

2.5. General Micro-Reactor Concepts and Feature Matrix

The micro-reactors reviewed in this section share a number of key characteristics but also have important differences. The Kilopower design is not considered further as it is not expected to be deployed for terrestrial applications. To simplify the scope of the following discussion, this report considers only micro-reactor concepts that could be utilized for terrestrial applications. Furthermore, the primary focus of this report is on commercial applications. The unique aspects of deploying micro-reactors for military applications are not explicitly addressed by this report, although overall methods discussed for evaluating levels of safety could be generically applicable. Although there are not many designs under consideration, it may be helpful to group them in terms of the arrangement of the fuel, which is likely to have a significant influence on the safety analyses.

The designs are divided into two general concepts. The first, labeled the Monolithic concept, is inspired by the eVinci design and uses information from the Megapower design where public information on eVinci is lacking. The other concept, labeled the Assembly concept, is inspired by the Oklo design, using the INL Megapower alternative Design A to supplement Oklo design information not publicly available.

The parameters which are expected to be important to licensing and safety analysis for the representative of each concept are compared in Table 2-2. The overall system layout of eVinci (see Figure 2-15) is considered prototypical for heat pipe micro-reactors and will be used to represent relative locations of systems in both concepts. The primary differences between the Monolithic and Assembly concepts exist in the configuration of the core and the heat pipes.

Table 2-2. Selected Parameters for Micro-Reactor Concepts

Parameter	Monolithic	Assembly
Thermal Power	5 MW	2 MW
Fuel	UO_2	U-10Zr
Neutron Spectrum	Epithermal	Fast
U-235 Enrichment	HALEU	HALEU
Fuel Cladding	Stainless Steel (SS), drop-in or monolith	SS
Number of Fuel Elements	2,112	703
Heat Pipe Working Fluid	K	Na
Heat Pipe Evaporator Wall Material	SS, drop-in or monolith	SS
Heat Pipe Condenser Wall Material	SS	SS
Heat Pipe Temperature	675°C	675°C
Number of Heat Pipes	1,224	703
Intended Operating Period	10 years	12 years
Reactivity Control	B_4C drums	B_4C drums

2.5.1. Monolithic Concept

The Monolithic micro-reactor concept utilizes a monolithic metal structure to support the fuel and evaporator sections of the heat pipes and is best represented by eVinci (see 2.3). Note that the eVinci design proposes an epithermal neutron spectrum while Megapower was to operate in the fast neutron spectrum. The Monolithic concept has an intended lifespan of ten years before the entire package (see Figure 2-15) is returned to a factory for refueling or storage.

2.5.1.1. Core, Fuel, and Reactivity Control

The Monolithic concept uses UO2 fuel enriched to 19.75% and operates in the epithermal spectrum at 675°C. A helium bond gas is used at a pressure of 20 atmospheres [23]. A metal hydride neutron moderator is used. The moderator material is intended to contribute to passive reactivity control as it will dissociate at higher temperatures, releasing hydrogen to reduce the overall reactivity. The operating reactivity control is provided by rotating B4C control drums in the reflector region of the core. For a sense of scale, Megapower uses 12 control drums each with reactivity worth of approximately $1.10, compared to approximately $3.00 excess core reactivity at beginning of cycle [24].

The available literature does not establish if fuel elements and heat pipes for eVinci will be manufactured separately and bonded to the monolith or if the material of the monolith will act as the walls of the fuel elements and/or the evaporator sections of the heat pipes [21]. This decision has implications for any thermal/structural analysis of the core as well as DID. This analysis considers both options where there are likely to be substantive differences.

If the fuel element walls are integral to the monolith, then this implies that fuel pellets are added to holes in the monolith with bonded caps to either end of the channel. Because the walls of the monolith act as the fuel cladding, a failure of the monolith may lead to release of radionuclides to containment (or confinement, see Section 2.5.1.5) or to other fuel elements or heat pipes with failed monolith walls. If a monolith wall failure leads to radionuclides entering a heat pipe, they may be transported to the condenser section within the PHX.

Conversely, the insertion of pre-fabricated fuel elements into holes in the monolith presents its own challenges. This method nominally adds another layer of material which may factor into DID as well as heat transfer and structural calculations. This method may be preferably for economic reasons as pre-fabricated elements are likely to be easier to inspect and replace if non-conforming [24]. However, the interactions between the cladding and the monolith walls may be more complex to evaluate.

2.5.1.2. Primary Coolant System

The potassium heat pipes will operate at sub-atmospheric pressure [23]. The open literature does not establish if fuel elements and heat pipes will be manufactured separately and bonded to the monolith or if the material of the monolith will act as the walls of the fuel elements and/or the evaporator sections of the heat pipes [21]. This decision has implications for any thermal/structural analysis of the core as well as DID. This analysis considers both options where there are likely to be substantive differences.

If the heat pipe evaporator walls are integral to the monolith, then the heat pipe condenser wall sections are manufactured separately and bonded to either end of the monolith channel after the wick and working fluid are loaded. The failure of the monolith may lead to release of heat pipe working fluid to containment (or confinement, see Section 2.5.1.5) or to other fuel elements or heat pipes with failed monolith walls.

Conversely, the insertion of prefabricated heat pipes into holes in the monolith presents its own challenges. This method may be preferably for economic reasons as prefabricated heat pipes are likely to be easier to inspect and replace if non-conforming [24]. However, the interactions between the heat pipe walls and the monolith walls may be more complex to evaluate.

The total volume of primary coolant is divided into thousands of individually-contained heat removal devices, which makes a loss of coolant accident (LOCA) in a micro-reactor different from that of a reactor with a single large volume of coolant. The loss of a single heat pipe as well as the potential for a cascade of failing heat pipes could be an important LBE for micro-reactors using heat pipes. The loss of heat removal from the condensing section of the heat pipes is also likely to be of importance.

2.5.1.3. Power Conversion

Multiple power conversion schemes have been proposed for micro-reactors, including an sCO_2 Brayton cycle, Stirling engines, and/or direct heating. A steam-based system is generally not considered due to its lower thermal efficiency (e.g., versus sCO_2) and the potential for encountering potassium or sodium in the event of a PHX leak.

The Monolithic concept is assumed to use an sCO_2 power conversion system that is arranged in such a way that activation of the system is minimized, and a turbine missile could only damage the lines coming from the PHX.

2.5.1.4. Decay Heat Removal

The Monolithic concept uses decay heat exchangers (DHXs) between the core and PHXs on either side of the core (see Figure 2-15). The transfer of heat into the DHXs depends on heat being conducted from the fuel through the monolith and then convected through the heat pipes. The ultimate method of decay heat removal is not specified in the open literature for Megapower or eVinci, but it is described as passive. Oklo has proposed a natural circulation air cooling system that constantly removes a decay-heat level of energy from the reactor enclosure [29] similar to that shown in Figure 2-17. This configuration is assumed for the Monolithic concept. The section of the HPs that travel through the DHX transfers heat to an inert gas which in turn transfers heat to the reactor enclosure walls. The heat transfer from the vessel depends on the establishment of sufficient natural circulation of air through the RCCS.

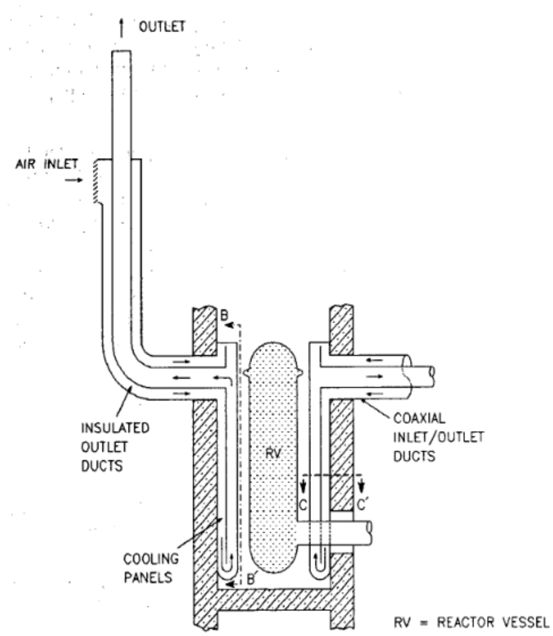

Figure 2-17. MHTGR Reactor Cavity Cooling System [39]

2.5.1.5. Containment/Confinement

Micro-reactors (and non-LWRs in general) present a challenge to the traditional LWR approach of a pressure-retaining containment structure. Many non-LWRs are designed such that expected transients do not generate high-pressure conditions. There have been proposals to develop performance criteria for a system, called a functional containment, that would not necessarily retain pressure at all phases of operation or an accident, but would adequately retain radionuclides for the same degree of protection to the public and the environment as a traditional containment. This is discussed extensively in SECY-18-0096 [40], which was a 2018 Staff Requirements Memorandum (SRM) [41] reflecting the Commission decision to approve SECY-18-0096. SECY-18-0096 summarizes the concept of functional containment as follows:

> "Non-LWR technologies have operating conditions, coolants, and fuel forms that differ from LWRs. These differences may allow or possibly require different approaches to fulfilling the safety function of limiting the release of radioactive materials. This has led to describing a "functional containment" as a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment."

It should be emphasized that this position represents a significant evolution of safety philosophy from LWRs. It reflects the range of additional barriers that are present in many advanced non-LWR concepts capable of preventing or mitigating the release of radionuclides to the environment.

The Commission approved the methodology proposed by the staff in SECY-18-0096 [40], in the 2018 Staff Requirements Memorandum [41]:

> "The Commission has approved the staffs proposed methodology for establishing functional containment performance criteria for non-light-water-reactors (non-LWRs). The staff should continue to keep the Commission informed as it develops the licensing framework for non-L WRs and should notify the Commission if future policy issues arise as this work progresses."

The design of the reactor enclosure is not well-established at this point. However, it is reasonable to expect that reactor designers will consider a functional containment. One means by which the adequacy of a functional containment design could be demonstrated is through the development of a mechanistic source term (MST) [32]. The envelope of the functional containment is assumed to be the package shown in Figure 2-15, with penetrations for instrumentation and control, the shell side fluid of the PHXs, and the shell side fluid of the DHXs.

2.5.2. Assembly Concept

In the Assembly concept, the heat pipes and fuel elements are manufactured separately and are combined in a grid to form the core. This is best represented by the Oklo design (see Section 2.3) or the INL Megapower Design A (see Section 2.2). The Assembly concept has an intended lifespan of twelve years before the entire package (see Figure 2-15 for the general size and shape) is returned to a factory for refueling or long-term storage.

Considerably less public information exists for designs which fall under the Assembly concept versus the Monolithic concept (see Section 2.5.1) and so the analysis is more dependent on the design and operational assumptions made in this section.

The Assembly concept is assumed to be vertically oriented with a single PHX above the core as shown in Figure 2-14. This differs from the Monolithic concept which is generally assumed to be horizontally oriented with PHXs on either side of the core (see Figure 2-15).

2.5.2.1. Core, Fuel, and Reactivity Control

The Assembly concept uses metallic U-10Zr fuel enriched to the HALEU range (assumed 19.75%) and operating in the fast spectrum at a temperature of 675°C. The core is represented by Figure 2-12. Operating reactivity control is provided by rotating B_4C control drums in the reflector region of the core. For a sense of scale, Megapower uses 12 control drums each with reactivity worth of approximately $1.10, compared to approximately $3.00 excess core reactivity at beginning of cycle [24].

Review of available concepts highlight that fuel elements may be manufactured in hexagonal cross-sections with stainless steel cladding and a central circular void as seen in Figure 2-8 or Figure 2-13. The circular void could be filled with a heat pipe before the entire unit is added to the core assembly through the use of a grid system. The core assembly would be sub-critical without the reflectors and without the control drums being turned sufficiently with the reflector sides towards the core.

2.5.2.2. Primary Coolant System

The assembly concept uses sodium heat pipes operating at sub-atmospheric pressure [29]. The total volume of primary coolant is divided into hundreds of individually-contained heat removal devices, which makes a LOCA in a micro-reactor different from that of a reactor with a single large volume of coolant. The loss of a single heat pipe as well as the potential for a cascade of failing heat pipes could be an important LBE for micro-reactors using heat pipes. The loss of heat removal from the condensing section of the heat pipes is also likely to be of importance.

2.5.2.3. Power Conversion

Multiple power conversion schemes have been proposed for micro-reactors, including an sCO_2 Brayton cycle, Stirling engines, and/or direct heating. A steam-based system is generally not considered due to its lower thermal efficiency (vs sCO_2) and the potential for encountering potassium or sodium in the event of a PHX leak.

The Assembly concept is assumed to use an sCO_2 power conversion system that is arranged in such a way that activation of the system is minimized, and a turbine missile could only damage the lines coming from the PHX.

2.5.2.4. Decay Heat Removal

The Assembly concept is assumed to use an RCCS as diagrammed in Figure 2-17. Oklo has proposed a natural circulation air cooling system that constantly removes a decay-heat level of energy from the reactor enclosure [29]. The Assembly concept could use this method to ensure decay heat removal. The section of the heat pipes that travel through the DHX transfers heat to an inert gas, which in turn transfers heat to the reactor enclosure walls. The heat transfer from the vessel depends on the establishment of sufficient natural circulation of air through the RCCS.

2.5.2.5. Containment/Confinement

Micro-reactors (and non-LWRs in general) present a challenge to the traditional LWR approach of a pressure-retaining containment structure. Many non-LWRs are designed such that expected transients do not generate high-pressure conditions. There have been proposals to develop performance criteria for a system, called a functional containment, that would not necessarily retain pressure at all phases of operation or an accident, but would adequately retain radionuclides for the same degree of protection to the public and the environment as a traditional containment. This was discussed extensively in SECY-18-0096 [40], which was approved by the Commission in a 2018 SRM [41]. As noted above, SECY-18-0096 defines a concept of functional containment that represents a significant evolution of safety philosophy from LWRs. It reflects the range of additional barriers that are present in many advanced non-LWR concepts capable of preventing or mitigating the release of radionuclides to the environment.

The design of the reactor enclosure is not well-established at this point. However, it is reasonable to expect that reactor designers will consider a functional containment. One means by which the adequacy of a functional containment design could be demonstrated is through the development of a mechanistic source term (MST) [32]. The envelope of the functional containment is assumed to be one half of the package shown in Figure 2-15, with penetrations for instrumentation and control and the shell side fluid of the PHX.

2.5.3. Manufacture, Transportation, and Final Construction

Both micro-reactor concepts may have features beyond their physical design that will be of interest in the licensing process. A primary manufacturing concern is that both micro-reactor concepts use HALEU in novel fuel arrangements. Processes must be developed to manufacture the fuel and other core components in a reliable and inspectable way [42].

All micro-reactors reviewed here have proposed to manufacture the reactor enclosure (see Figure 2-15) and its contents at a controlled facility and transport it, fueled, to its operating site. When compared to assembling the reactor on-site, this approach may reduce uncertainties in schedule and allow for easier inspection and re-work of non-conforming components. On the other hand, transportation of the enclosure to the operating site as an assembled unit is likely to complicate requirements for packaging and routing of materials.

At the operating site, the reactor enclosure may be installed below-grade with common construction equipment and tied into local infrastructure including the power conversion system, DHRS, and a control room[2]. Final testing and inspection of the completed design at the site would be required before power operation begins. Note that for micro-reactor concepts, manufacture of the nuclear reactor off-site is a possibility. This would allow vendors to achieve greater economies of scale and simplify the construction process. In this situation, the final inspection and testing at site would require evaluation of the installation as well as performance of commissioning tests. The manufacture of the nuclear system at an off-site facility would require additional testing and inspection prior to shipment of the micro-reactor loaded with fresh fuel.

[2] Because of the simplicity of micro-reactor concepts, there has been some consideration of operation without an on-site control room.

3. LICENSING MODERNIZATION PROJECT

This section presents the major tasks of the LMP [10], discussing some reactor applications of the LMP, and assessing how tasks may apply or require additional consideration for a micro-reactor. This section is organized into the following components of the LMP.

- Section 3.1 presents the LMP approach to selecting and evaluating LBEs, including deterministic approaches for utilizing PRA insights to select and conservatively evaluate design basis accidents (DBAs).

- Section 3.2 presents the LMP approach to Structures, Systems, and Components (SSC) safety classification and performance criteria specification.

- Section 3.3 presents the LMP approach to evaluating the DID adequacy.

This section has been developed using the September 2018 draft issue of the LMP [9]. The process followed here is not intended to be a complete implementation of the LMP for developing a risk-informed, performance-based safety basis to be submitted for regulatory review. Rather, this section utilizes the available draft LMP for the purpose of developing insights into how components of the LMP may or may not require an evolution of safety/risk assessment methods.

The objective of the LMP is to foster the development of technology-inclusive, risk-informed, and performance-based regulatory guidance for licensing non-LWRs. It still requires NRC consideration, with the NRC responsible for independently developing regulatory guidance that can achieve the overall objective of being technology-inclusive, risk-informed and performance-based. Currently the NRC has issued a draft guidance document [43] for public comment that has incorporated some evaluation of the LMP.

Conceptually, there are no fundamental technical issues with the LMP that would restrict micro-reactor designers from using it to formulate a micro-reactor licensing application. In some cases, however, the methods to perform a PRA must be enhanced in order to treat design characteristics that are unique relative to operating LWRs. A key example is how external events impact safety functions provided by passive systems. Some of the possible issues that would be faced by application of the LMP to micro-reactors are discussed further for the various elements of the LMP. Another relevant issue is how DID should be evaluated for designs that are low-power, relatively simple (in terms of technological complexity and the number of interacting or interfacing systems) and provide safety function utilizing passive systems that generally are relatively robust. For such designs, there may be a more limited set of layers of defense

3.1. Selection of Licensing Basis Events

The LMP (NEI 18-04) [9] provides a technology-inclusive approach to develop LBEs, classify SSCs, perform SSC special treatments, and assess DID adequacy. It is an example of a risk-informed application. How it is implemented in the context of a micro-reactor design may require some additional considerations related to methodology for evaluating reliability and risk for a system that places significant reliance on inherent mechanisms to achieve key safety functions.

Figure 3-1 illustrates the LMP process, where the initial list of LBEs is determined. The LMP approach leverages the design-specific PRA to identify and assign frequencies to events. This report utilizes reference material relevant to micro-reactors to aid in evaluating methodological challenges in implementing aspects of the LMP related to the selection of LBEs. This includes the recently released eVinci micro-reactor LMP demonstration [38]. In areas where a lack of knowledge exists, engineering judgment is utilized to assign event frequencies in order to provide an illustrative classification of events. The discussion that follows considers the micro-reactor concepts discussed in Section 2, with focus implicitly on HPR concepts.

The uniqueness of micro-reactors is such that using other reactor information or experience could lead to an incorrect characterization of the risk profile, biasing the LMP process. As a result, application of operating experience from other areas judged to be similar to micro-reactor operation should be incorporated following a pre-established technical methodology. This is necessary to promote incorporation of similar, but non-prototypic, operating experience in a repeatable manner (i.e., limits analyst-to-analyst variability) justified by an articulated technical basis. In the case where engineering judgement is utilized, this should be supported by a reviewed technical justification.

The process diagrammed in Figure 3-1 is used under the LMP to identify and evaluate LBEs[3]. A micro-reactor with existing design development work and an as-designed PRA may be able to leverage that work at Step 4 to begin classifying an existing list of LBEs from the PRA. This was the case with the LMP demonstrations associated with the High-Temperature Gas-Cooled Pebble Bed Reactor (HTGC-PBR) [44] and PRISM [45]. In both demonstrations, the initial list of LBEs was identified from sequences in limited scope PRAs (primarily restricted to the evaluation of risk associated with internal events[4]). These event sequences and associated frequencies could be directly grouped into anticipated operational occurrences (AOOs), DBEs, and beyond design basis events (BDBEs). In some cases, event sequences with similar progression and dependence on similar SSCs were grouped together to reduce the number of events considered.

Table 3-1 shows the criteria used under the LMP to classify grouped LBEs, AOOs, DBEs, and BDBEs. They are all defined by their assessed frequencies per plant-year given expected behavior from all SSCs. The DBAs are prescribed such that safety-related SSCs are sufficient to maintain dose limits during a DBE. Data sources are summarized in Table 3-2.

NEI 18-04 LMP guidance states "… for normal operations, including AOOs, the NRC regulations are, for the most part, generic and can be applied to an advanced non-LWR plant." The relevant regulations are either 10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities) or 10 CFR Part 52 (Licenses, Certifications, and Approvals for Nuclear Power Plants). While this statement may be applicable to most non-LWR designs (e.g., molten salt reactors (MSRs), high-

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[3] It is important to communicate the footnote attached to event sequences with respect to in LBEs in Nuclear Energy Institute (NEI) 18-04. Paraphrasing the footnote on page 2 of NEI 18-04, "… Licensing Basis Events are defined in terms of event sequences comprised of an Initiating Event, the plant response to the Initiating Event (which includes a sequence of successes and failures of mitigating systems) and a well-defined end state."

[4] It should be noted that for many advanced, non-LWR concepts, the risk contribution from internal events (e.g., arising due to the random failure of components) is likely relatively low. These reactor concepts have generally been developed utilizing the operational experience acquired from the current operating LWR fleet. This experience has led to design concepts where safety functions are generally achieved utilizing simpler engineered systems, in many cases systems that rely on naturally occurring processes to achieve the required function (e.g., passive heat removal systems where coolant flow is realized through natural circulation processes obviating the need for an active engineered means of delivering motive force to the coolant).

temperature gas reactors, or sodium fast reactors), the relative simplicity of micro-reactors together with the reliance on inherent safety systems may lead to a limited number of AOOs being identified.

No matter the applicability of the NRC regulations, the LBEs identified in the PRA are important events that have the potential to release radioactivity to the public. Additionally, the LBE process needs to be developed in a manner that facilitates the determination of risk-significant LBEs and SSCs and the evaluation of DID adequacy.

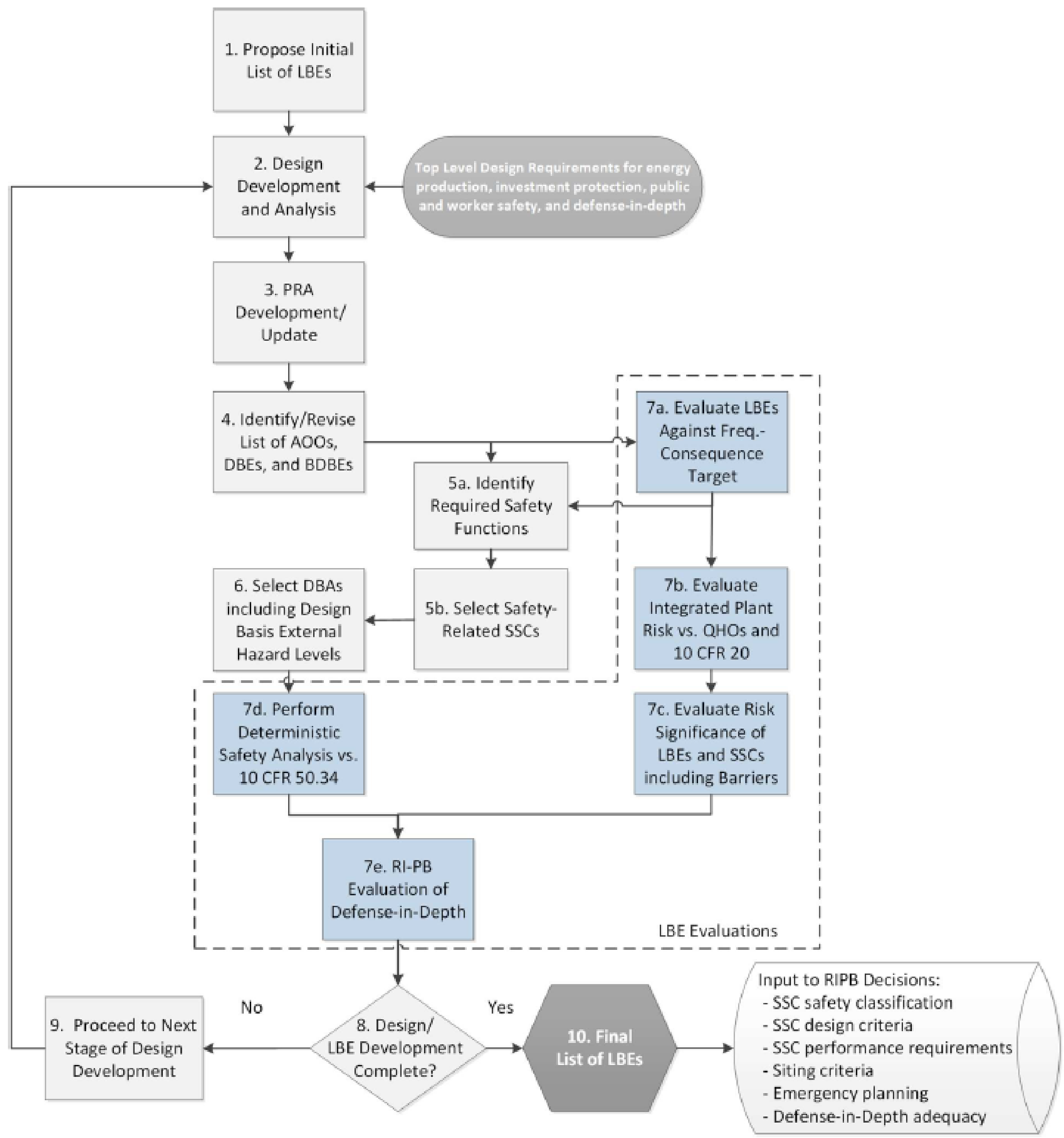

Figure 3-1. LMP Process for Selecting and Evaluating Licensing Basis Events [9]

Table 3-1. Paraphrased LMP LBE Classifications [9]

Licensing Basis Event	Mean Frequency Range (/plant-year)	Description
Anticipated Operational Occurrence	$P \geq 10^{-2}$	Expected to occur one or more times during the life of a nuclear power plant. Takes into account the expected response of all SSCs within the plant, regardless of safety classification.
Design Basis Event	$10^{-2} > P \geq 10^{-4}$	Not expected to occur in the life of a single nuclear power plant. Takes into account the expected response of all SSCs within the plant, regardless of safety classification.
Beyond Design Basis Event	$10^{-4} > P \geq 5 * 10^{-7}$	Rare event sequences that are not expected to occur in the life of a nuclear power plant. Takes into account the expected response of all SSCs within the plant, regardless of safety classification.
Design Basis Accident	N/A	Postulated event sequences that are used to set design criteria and performance objectives for the design of Safety Related SSCs. Assume that only Safety Related SSCs are available to mitigate DBE consequences to within the 10 CFR 50.34 dose limits.

Table 3-2. Micro-Reactor LBE Identification Data Sources

Source	Justification
eVinci LMP Demonstration	The eVinci concept developed by Westinghouse has been used as a pilot application of the LMP for micro-reactor concepts [37].
Oklo DG-1353 Pilot [29]	This is one of the micro-reactor concepts to which the LMP has been applied. The Oklo reactor is the inspiration for the Assembly concept in this analysis.
Megapower PIRT [23]	The Megapower reactor is the inspiration for the Monolithic concept in this analysis.
PRISM LMP Demonstration [45] PRISM Preliminary Safety Information Document [46]	PRISM utilizes passive heat removal systems and inherent reactivity feedback to a similar degree as micro-reactors. Fuel and coolant similarities may reveal design-specific events.
HTGC-PBR LMP Demonstration [44]	HTGC-PBR utilizes passive heat removal systems and inherent reactivity feedback to a similar degree as micro-reactors. Inclusion of this source serves to identify “generic” reactor events.

3.1.1. Task 1: Propose Initial List of LBEs

Similar to assessing initiating events in LWRs, the initial list of LBEs, which includes postulating initiating events and event sequences, are deterministically selected and may be supported by qualitative risk insights. For advanced non-LWRs with no operating experience and in the pre-conceptual design phase, analysis techniques such as failure modes and effects analyses (FMEA), hazard and operability studies, or Master Logic Diagrams may be used for the initial selection of events for micro-reactors.

For the LWR operating fleet, there are many hundreds of years of operating experience to leverage for identifying initiating events. Without plant operating experience, micro-reactor designers must leverage one or more of the analysis techniques described above. Adoption of existing LBE demonstrations should generally be avoided. Micro-reactors are significantly different from large Generation IV designs (e.g., PRISM [45]), and lack the operating experience that supports some non-LWR concepts (e.g., EBR-II operating experience). An inadequate specification of LBEs would likely lead to incorrect risk insights as part of the LMP process.

When formulating the list of LBEs, designers should consider how LBEs affect Fundamental Safety Functions (FSFs) for nuclear reactors. Assessing safety in any reactor is generally expected to be based on the FSFs defined by the International Atomic Energy Agency (IAEA) in Specific Safety Requirements SSR-2/1 [47]. Specifically, SSR-2/1 states under Requirement 4:

> *"Fundamental safety functions Fulfilment of the following fundamental safety functions for a nuclear power plant shall be ensured for all plant states: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases."*

These are also known as the "three C's"—control, cool and confine. Typically, the initial list of LBEs would be developed for a micro-reactor design utilizing these FSFs to define challenges to a design's safety function.

In the case of LWR designs, the formulation of the above FSFs provides a foundation from which to develop design-specific events that challenge safety functions. For example, NUREG-0800 SRP 15.0 [48] categorizes AOOs and postulated accidents according to one of seven types.

- Increase in heat removal by the secondary system
- Decrease in heat removal by the secondary system
- Decrease in reactor coolant system flow rate
- Reactivity and power distribution anomalies
- Increase in reactor coolant inventory
- Radioactive release from a subsystem or component

These types of events represent altered conditions with respect to controlling, cooling and confining. Event scenarios within each group will potentially have different effects on the plant that challenge different safety limits.

As noted in NUREG-0800 SRP 15.0 [48], AOOs are those events that are expected to occur one or more times during the operating life of the nuclear power plant. This ranges from events of moderate frequency (expected to occur several times during the operating life of the nuclear power plant) to infrequent events (may occur during the operating life of the nuclear power plant). In many cases, the design of the nuclear power plant incorporates systems that prevent an AOO from escalating to a situation where a more severe challenge to one or more barriers confining fission products occurs. This is typically the case for a postulated accident, which is not expected to occur during the operating life of a nuclear power plant.

Micro-reactor LBEs can be developed based on consideration of the FSFs specified in SSR-2/1 [47]. The eVinci pilot of the LMP [38] identified a few initiating events unique to the reactor concept. These initiating events are specific to a single unit operating at power, considering only the reactor core as a source of radioactive material that could be released to the environment. The initiating events identified are:

- Spurious reactor trip
- Power Conversion Subsystem failures causing an initiating event
- PHX tube rupture
- Multiple (greater than three) heat pipe seal ruptures

The approach followed to identify these initiating events considered available lists of pressurized water reactor (PWR) initiating events, screening each event for applicability to the eVinci design. The PWR initiating events considered can be found in NUREG/CR-3682 [49], NUREG/CR-5750 [50], and NUREG/CR-6928 [51]. In addition, this eVinci LMP demonstration [38], utilized an FMEA to identify events that are unique to the eVinci Micro-Reactor. The initiating event frequencies were developed using a combination of approaches including nuclear industry data, engineering judgement, assumed failure rates, and quantification of system fault trees. Where failure rate assumptions were made, these were reviewed by the design team.

This approach ensures consistency across the evaluation of different 50th percentile initiating event frequencies. However, it is subject to uncertainty that could affect the assessment of relative risk amongst different initiating event classes. The sources of these uncertainties could lead to more significant uncertainties for specific initiating event class frequencies. For example, initiating events unique to heat pipe design, such as multiple heat pipe seal ruptures, could have significantly larger uncertainty due to the comparable lack of experience with manufacturing practices and micro-reactor operating conditions.

While treating uncertainties through sensitivity analysis can aid in understanding the impact of uncertainties on overall risk, evaluating how different sources of uncertainty across event sequences or event sequence classes contribute to risk is valuable from the perspective of managing risk. For example, consider a situation in which multiple heat pipe failure initiating event frequencies have relatively large uncertainty compared to frequencies for initiating events in other classes. A risk management program for a new reactor should adapt its inspection program in light of this to a) gain knowledge that would enable uncertainty reduction with operational experience, and b) detect the potential for this initiating event and thereby control the impact of this uncertainty on realized consequences to public health and safety.

As noted above, the development of LBEs for a new reactor with limited operating experience should follow from FSFs. The eVinci micro-reactor LMP demonstration [38] developed a functional event tree utilizing three top events corresponding to the IAEA SSR-2/1 [47] FSFs identified above.

- Reactivity control

  Subsystem eVinci relies on three approaches to control reactivity [38]: (a) Control Drum subsystem, (b) Emergency Shutdown Subsystem, and (c) passive release of hydrogen from the moderator (H2).

- Heat removal

  eVinci has two heat removal pathways [38]: a) heat removal from the reactor by the secondary heat removal system (Secondary), and b) thermal conduction from the core block to the canister with natural convection removing heat from the canister wall to an air duct system discharging air into the environment (SVS).

- Confinement of radioactive material

  The Canister Containment Subsystem (CCS) in the eVinci design provides the overall confinement function.

A generic functional event tree for the eVinci design is reproduced in Figure 3-2. This functional event tree considers a condition in which a pre-existing heat pipe failure could exist with the reactor maintained in operation. This pre-existing heat pipe failure is an initial top event in the functional event tree shown in Figure 3-2.

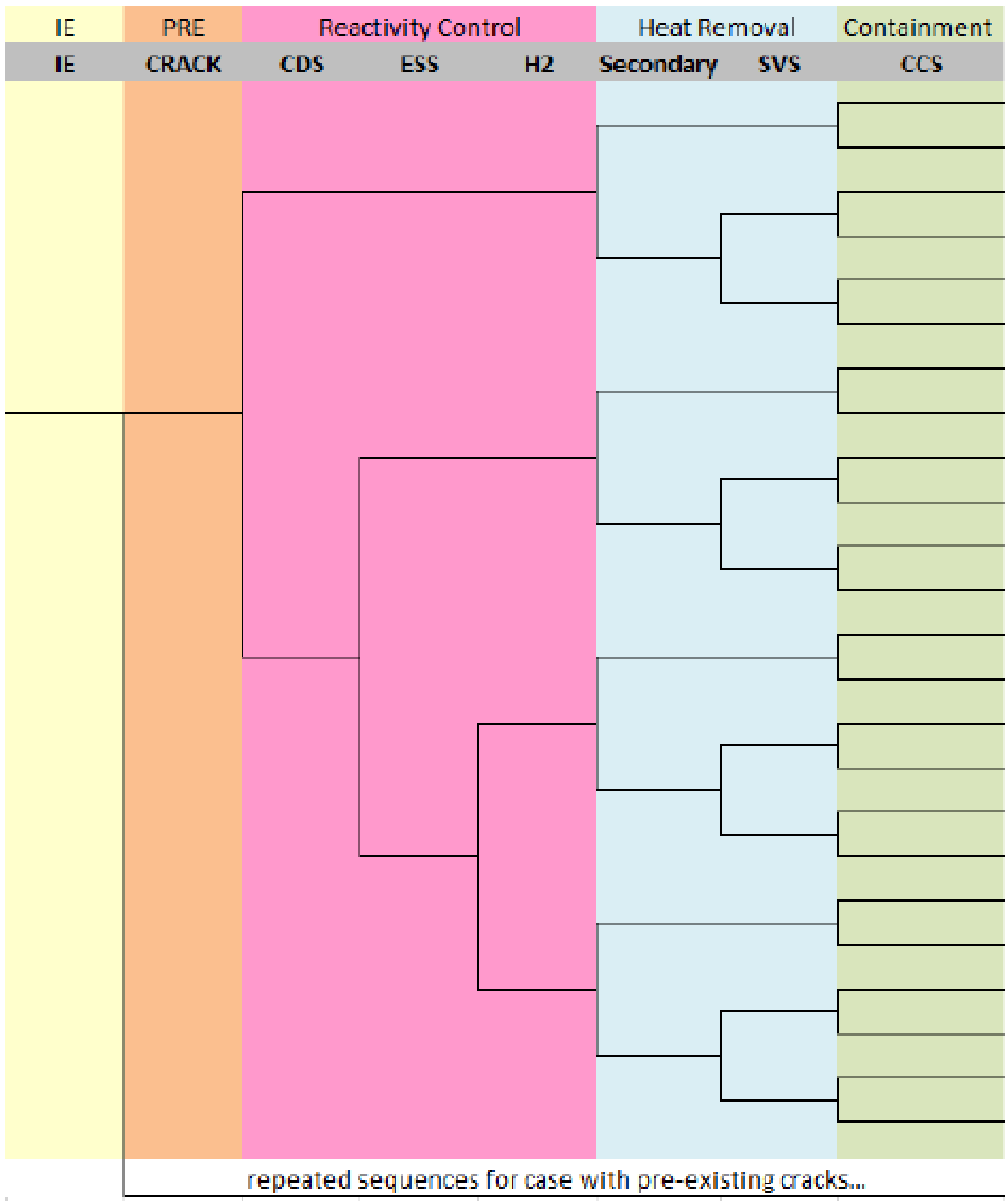

Figure 3-2. eVinci Micro-Reactor Generic Functional Event Tree [38]

3.1.2. Task 2: Design Development and Analysis

Design analysis is performed and iterated upon for all phases of the design (i.e., pre-conceptual, conceptual, preliminary, and final design phase). Task 2 of LBE process includes defining each element of the safety design approach, identifying the design features to meet the top-level requirements for energy production and investment protection, and analyses to develop sufficient understanding to perform a PRA and the deterministic safety analyses. Logically, effort under Task 2 also identifies safety-related SSCs that are needed for the deterministic safety analysis.

3.1.3. Task 3: PRA Development/Update

The PRA is developed for a design to a level of detail consistent with the various phases of the design. The level of detail required should be sufficient to develop risk insights that can inform the design and guide subsequent phases of the design. The PRA should be updated at each phase of the design to reflect evolution of the design. There is no requirement as part of the LMP for a reactor designer to develop a PRA from design inception. There are alternate ways to develop the safety design philosophy at very initial stages of a design to ensure that the design will initiate from a robust basis for risk management. However, when PRAs are developed to inform the design from an early stage, they are expected to be of limited scope and details consistent with the relatively immature state of the design.

In order to develop a robust PRA, it is essential that significant upfront effort be placed in developing the potential failure modes for the design. Not extensively discussed in the LMP [9] is how to evaluate failure modes that emerge for simpler engineered systems (i.e., exploiting inherent safety) exposed to external perturbations from the environment, as typically evaluated within external hazard PRAs for currently operating LWRs. This remains the most important methodological challenge for evaluating risk from a micro-reactor concept.

NOTE: Additional events that could result in the release radionuclides to the environment arise due to malevolent/intentional acts. Such events are currently beyond the scope of PRAs for both LWRs and non-LWRs. These events are handled, from the perspective of reactor licensing, under 10 CFR Part 73.

For the micro-reactor concept, Table 3-3 provides a summary of the initiating event groups. Included with this summary is an indication of likely event classes for these event groupings (i.e., AOO, DBE or BDBE). The event groups identified in Table 3-3 include power control initiating events (IE group Transient Overpower (TOP)), heat removal initiating events (IE groups BOP, PHX, DHX, HP and COR). Table 3-4 lists example power control initiating events. Table 3-5 lists examples of heat removal initiating events from the BOP, PHX, DHX, HP and COR initiating event groups.

Table 3-3. Micro-Reactor Initiating Event Groups

Relevant Goal	IE Group	Description	AOO	DBE	BDBE
Power Control	TOP	Transient Overpower		X	X
Heat Removal	BOP	Balance of Plant (BOP) Transients	X	X	
Heat Removal	PHX	PHX Leaks		X	
Heat Removal	DHX	DHX Leaks		X	
Heat Removal	HP	Loss of Heat Pipe Functionality	X	X	
Heat Removal	COR	Loss of Core Heat Removal			X

Accepted techniques exist for identifying failure modes of engineered systems[5]; however, their application to identifying failure modes for inherent safety systems requires careful attention. It is important that the evaluation does not exclude potential disruption of physical processes, upon which inherent safety systems rely. While the likelihood of such disruption is very low under external conditions typical of operation, the potential for different external conditions should not be precluded.[6] It is conceivable that failure modes only emerge when considering perturbations to the system arising from its interaction with the external environment.

Table 3-4. Power Control Initiating Events

Initiating Event Name	Initiating Event Description	Plant Response	Frequency (1/yr)	LBE
TOP1	Medium reactivity insertion due to misalignment of a single bank of control drums	Sufficient to trigger scram, but design will tolerate without fuel damage despite scram failure. Source [46]	1×10^{-4}	DBE
TOP2	Large reactivity insertion due to misalignment of multiple banks of control drums	Scram failure leading to failure of a few pins before inherent reactivity feedback reduces power. Source [46]	1×10^{-4}	DBE
TOP3	Extreme reactivity insertion due to misalignment of all control drums	Scram failure leading to significant fuel damage before inherent reactivity feedback reduces power. Source [46]	1×10^{-6}	BDBE

The quantitative characterization of the likelihood of such failure modes, however, may have large uncertainty. This presents a methodological challenge for micro-reactor PRAs beyond the scope of this report. Given the overall simplicity of the micro-reactor design, compensatory measures to mitigate disruption of an inherent safety system could provide significant benefit to overall risk management in light of uncertainties that are difficult to characterize.

[5] Examples of methods include FMEA and Process Hazard Analysis (PHA). The System Theoretic Process Analysis (STPA) has been found to provide a means of identifying failure modes of engineered systems that arise due to interaction of multiple SSCs. The STPA method is often useful at identifying failure modes that are not readily identified through detailed evaluation of individual SSCs.

[6] A robust identification of failure modes must incorporate not just potential random failures of SSCs, but also the impact of external perturbations to the reactor system that could lead to degraded or completely failed SSC functioning. For reactor systems relying on inherent safety, the risk profile of the plant is likely to be dominated by these types of interactions between the engineered system and its external environment. These external perturbations reflect not just external events (e.g., seismic events) but should also incorporate to some measure malevolent acts.

Table 3-5. Heat Removal Initiating Events

Initiating Event Name	Initiating Event Description	Plant Reaction	Frequency (1/yr)	LBE
BOP1	BOP trip with bypass heat removal functional	Near-design level of heat removal. Sufficient to trigger scram, but design will tolerate without fuel damage despite scram failure [29].	9.98×10^{-1}	AOO
BOP2	BOP trip with bypass heat removal non-functional	Only decay-heat level of heat removal available. Immediate scram required to avoid fuel damage [29].	2×10^{-3}	DBE
PHX1	Leakage of PHX shell side to cavity or environment	Only decay-heat level of heat removal available. Immediate scram required to avoid fuel damage [52].	8.8×10^{-3}	DBE
DHX1	Leakage of DHX shell side to cavity or environment	Decay heat removal unavailable. Sufficient to trigger scram, but design will tolerate without fuel damage if turbine bypass heat removal available [52].	8.8×10^{-3}	DBE
DHX2	DHX air side blockage	Decay heat removal unavailable. Sufficient to trigger scram, but design will tolerate without fuel damage if turbine bypass heat removal available [46].	$\sim 10^{-8}$	Treated as BDBE[7]
Heat Pipe 1 (HP1)	Small leakage through tubes	Small loss of primary and decay heat removal from core. Not sufficient to trigger scram. Source [52]	4.4×10^{-2}	AOO
HP2	Medium leakage through tubes	Medium loss of primary and decay heat removal from core. Sufficient to trigger scram, but design will tolerate without fuel damage despite scram failure. Source [52]	8.8×10^{-3}	DBE
HP3	Large leakage through tubes	Large loss of primary and decay heat removal from core. Immediate scram may not be sufficient to avoid fuel damage. Source [52]	2.6×10^{-3}	DBE
COR1	Core structural failure sufficient to decouple fuel from HPs	Large loss of primary and decay heat removal from core. Immediate scram may not be sufficient to avoid fuel damage. Source [46]	$\sim 10^{-11}$	Treated as BDBE

[7] It is assumed that detailed design and operational procedures will ensure that this event is very rare. It is assumed to be a BDBE as a result.

3.1.4. Task 4: Identify/Revise List of AOOs, DBEs, and BDBEs

The development of a PRA is required to identify the different groupings of LBEs. As listed above in Table 3-1, the LBEs are classified into different groups based on their evaluated frequencies. A fundamental challenge for micro-reactors is developing this event classification in a manner that accounts for significant uncertainties in initiating event frequencies, as described above under Task 3.

As noted in the LMP [9] in relation to this task, additional emphasis may be required in the area of ensuring adequate DID in the design. Currently the LMP [9] notes that rare events (those with 95th percentile frequencies less than 5×10^{-7}/plant-year) should be used to confirm that the design is not subject to cliff-edge effects and that adequate DID (Task 7e) has been built into the safety approach. This could include modifications to the design or enhancement of programmatic controls; it does not typically result in modifications to the LBE classification unless the design changes.

3.1.5. Task 5a: Identify Required Safety Functions

Under this task, PRA Safety Functions (PSFs) are identified. These are those functions that are necessary to prevent or mitigate the release from any radiological source treated within the PRA. A subset of PSFs will be Required Safety Functions (RSFs), which are those functions required to maintain either DBEs or BDBEs within their respective Frequency-Consequence (F-C) targets.

As noted above, uncertainty in the PRA model may play a significant role in evaluating how different safety functions contribute to the overall safety of the plant. In cases where high uncertainties exist, in either the evaluation of frequencies or consequences, it will be important for analytical methods to be applied that assess the extent to which certain safety function classifications may be sensitive to uncertainty. In cases where inherent safety systems are primarily relied upon to achieve the overall safety function, it is conceivable that the design may not be able to meet either DBE or BDBE F-C targets in light of uncertainty.

Evaluating the role of uncertainty in the identification of RSFs is an important insight that can be developed from the PRA. Incorporation into a DID approach may not adequately highlight the sensitivity of risk-informed decision-making to uncertainty. Insights at this stage, that account for the impact of uncertainty, should be considered in identifying RSFs for designs relying on inherent safety as a primary line of defense.

3.1.6. Task 5b: Select/Revise Safety-Related SSCs

Safety-related SSCs are identified as those which are required to:

- Achieve an RSF for each DBE and
- maintain BDBEs within the F-C target.

Following the LMP [9], safety related SSCs are

- selected by the designer from SSCs identified to perform RSFs necessary to mitigate the consequences of DBEs within the LBE F-C target. In addition, safety-related SSCs are the only SSCs credited to mitigate consequences of a DBA to meet dose limits of 10 CFR 50.34. Analyses evaluating DBA consequences are performed with conservative assumptions.

- identified by designers and relied upon to perform RSFs for certain BDBEs. These are BDBEs having consequences greater than the 10 CFR 50.34 dose limits. The selected SSCs are required to prevent the frequencies of these events from increasing to the point that the event becomes a DBE. In this manner, the selected SSCs ensure that the F-C target is met for this group of BDBEs.

In the case of inherent safety systems, the safety-related SSCs may need to be modified to incorporate operational limits necessary to maintain physical processes within bounds required to achieve their RSFs.

3.1.7. Task 6: Select Deterministic DBAs and Design Basis External Hazard Levels

DBAs are defined from DBEs by crediting only safety-related SSCs in performance of RSFs. All non-safety-related SSCs that could perform an RSF are assumed unavailable. Analysis of these DBAs is performed in preparation of a conservative deterministic safety analysis. As part of this process, the design basis external hazard levels are selected.

A challenge to demonstrating micro-reactor safety will be associated with significant reliance on inherent safety systems. It is possible that this reliance on inherent safety may lead to limited diversity in achieving a particular safety function. The concept of single failure is difficult to apply in this case. However, should conservative analysis demonstrate a high-confidence in performance of a safety function by an inherent safety system, the single failure criterion should be considered met through simplicity and diversity in the operational band of an inherent safety system. Specifically, if inherent safety system operation is robust under external perturbations considered within the design basis, which in many cases may be dominated by external hazards, this should serve as a surrogate for the single failure criterion when considering active systems.

3.1.8. Task 7: Perform LBE Evaluations

3.1.8.1. Task 7a: Evaluate LBEs Against F-C Target

At this stage, evaluation of LBEs against their F-C target is conducted. The evaluation should consider the impact of uncertainties in event frequency on meeting the F-C target. In situations where uncertainties lead to an event frequency uncertainty band that overlaps two LBE categories, it is generally required to evaluate the event against F-C targets for both categories. This is intended to ensure that uncertainties in frequency estimates do not impact whether or not a design can meet the F-C target.

The large uncertainty range for event frequencies that could occur in a micro-reactor PRA may require additional emphasis on DID strategies.

3.1.8.2. Task 7b: Evaluate Integrated Plant Risk against Quantitative Health Objectives and 10 CFR 20

Evaluation of integrated risk (i.e., across all LBEs) against the following cumulative risk targets is performed.

- The total frequency of a site boundary dose exceeding 100 mrem should not exceed one/plant-year for all LBEs.

- The average individual risk of early fatality within one mile of the exclusion area boundary (EAB) does not exceed 5×10^{-7}/plant-year for all LBEs.

- The average individual risk of latent cancer fatalities within ten miles of the EAB does not exceed 2×10^{-6}/plant-year for all LBEs.

At this stage, the evaluation of a range of design alternatives and enhanced programmatic elements may be valuable for micro-reactor designs. Such an evaluation could identify whether reasonable, cost-effective alternatives exist that could ameliorate any sensitivity of integrated plant risk to PRA model uncertainties.

3.1.8.3. Task 7c: Evaluate Risk Significance of LBEs and SSCs Including Barriers

LBEs are classified as risk-significant if the LBE site boundary dose exceeds 2.5 mrem over 30 days and the frequency of the dose is within 1% of the F-C target. SSCs are classified as risk-significant if an SSC is required to maintain any LBE with the F-C target, or if the total frequency of LBEs with the SSC failed within 1% of any of the three cumulative risk targets (Task 7b).

In situations where uncertainty in the evaluation of LBE risk is relatively high, as could be the case should a micro-reactor risk profile be dominated by rarer external hazards or malevolent acts, some reconsideration of how to apply this LMP guidance [9] may be warranted. It may be worthwhile to consider a sensitivity in which 95th percentile risk estimates are considered against the above criteria for LBE and SSC risk-significance. Should uncertainties not be a significant contributor, then the above conclusions related to risk-significance should not be altered. However, should uncertainties be relatively high, risk-significance may be altered. While this could be accommodated through alternate approaches, such as DID, it may prove simpler to accommodate uncertainty in the PRA model through enhancing the robustness of the design.

3.1.8.4. Task 7d: Perform Deterministic Safety Analyses Against 10 CFR 50.34

At this stage, a deterministic safety analysis is performed that utilizes conservative assumptions for DBAs. While excessive conservatism can be a challenge, particularly for complex, large-scale power reactors, the simplicity of a micro-reactor design may provide designers with sufficient margin to utilize conservative assumptions to ensure that inherent safety systems have a sufficiently broad envelope to accommodate uncertainties in external challenges to their function.

3.1.8.5. Task 7e: Risk-Informed, Performance-Based Evaluation of DID

Under this task, risk-significant sources of uncertainty in the PRA model (in frequency and consequence estimates) are evaluated to identify the robustness of the plant safety basis. In particular, this task represents a fundamental effort to ensure that the analytical evaluations of risk have not biased safety decision-making in a manner where the vulnerabilities have not been adequately addressed in the design.

While DID is a valuable means of accommodating uncertainties in a PRA model, and assuring that the overall plant safety strategy is robust in light of uncertainties that cannot be eliminated, it may be more beneficial for a designer to impose additional conservatisms to realize enhanced robustness in the design. In this manner, the PRA model results, discussed above, may be made more insensitive to uncertainties in inherent safety system operation.

3.1.9. Task 8: Decide on Completion of Design/LBE Development

At this stage, the insights developed through the above stages can be incorporated into a decision regarding whether the design should be frozen or progress to a subsequent design stage. If the design is judged to be adequately robust, then a freezing of the design may be appropriate. In the event that additional improvements to the design are necessary, then consideration of design changes, operational improvements, or programmatic enhancements should be developed to inform the subsequent design phase.

For micro-reactors, given the immaturity of the technology, a number of multiple steps may be required at this point. For example, the design could be frozen to support development of a prototype reactor that can be tested to further inform the development of the design. This could potentially occur multiple times, depending on the maturity level of the design. It may not be likely for the design to be developed without construction of a prototype.

3.1.10. Task 9: Proceed to Next Stage of Design Development

A design may progress iteratively through multiple stages to refine its safety-in-design. A design based on proven technology could largely proceed through design stages as noted in the LMP [9], relying on small-scale or separate-effect testing. As noted above, however, for designs that represent significant departures from the current technology, it may be warranted to perform one or more prototype stages to inform design with full-scale or integral-effect data. This may be the case for micro-reactors, given that many concepts represent a significant departure from the technological experience base in water-moderated nuclear reactor technology. In this case, this task would be preceded by a prototype development phase.

3.1.11. Task 10: Finalize List of LBEs and Safety-Related SSCs

At this stage, sufficient data and maturity in the design has accumulated to finalize the list of LBEs and safety-related SSCs. For micro-reactors, this stage may not be reached until one or more prototype development phases have occurred.

3.2. Safety Classification and Performance Criteria for Structures, Systems, and Components

After identifying and categorizing LBEs, a set of safety functions are determined for which success will keep releases within the limits established by the F-C targets in an occurrence of any identified LBE.

The LMP approach to classifying SSCs is shown in Figure 3-3 and has some overlap with the diagram for LBE evaluation in Figure 3-1. SSCs modeled in the PRA may be classified as safety-related, non-safety related with special treatment (NSRST), or non-safety related with no special treatment depending on their importance to meeting F-C targets in DBAs, DBEs, and BDBEs or for meeting DID adequacy.

PRISM identified safety functions by considering an ultimate goal ("maintain control of radionuclide release") and then evaluating sub-goals that are necessary to achieve it. The PRISM results are shown in Figure 3-4 and the process is performed for the micro-reactor concepts in Figure 3-5. Note that certain functions present for the PRISM analysis, such as "control radiation from processes", are not present for micro-reactors due to differences in plant configuration and operation. This is similar to the process followed and the basic safety functions determined for the HTGC-PBR [44].

A comparative study may be performed in which different sets of safety functions are credited against the set of DBEs and BDBEs. Those sets that meet the F-C targets for all LBEs are eligible to be the declared RSFs. Both the PRISM and HTGC-PBR demonstrations determined that core heat removal and reactivity control were sufficient as RSFs and they are expected to be sufficient RSFs for the LBEs for both micro-reactor concepts[8]. This corresponds to Step 5a in Figure 3-1.

The discussion in this section draws on examples from the PRISM and HTGC-PBR LMP demonstrations to guide the discussion. Neither are micro-reactors, and thus should not be generally considered to have specific applicability to the micro-reactor concepts. The discussion in this section focuses on the detailed guidance provided in the LMP [9] related to SSC evaluation (see Section 4 of the LMP).

[8] This set of two RSFs is not generically applicable to advanced reactors. It is conceivable that for a liquid-fuel molten salt reactor, an RSF involving containment of primary system coolant may be identified.

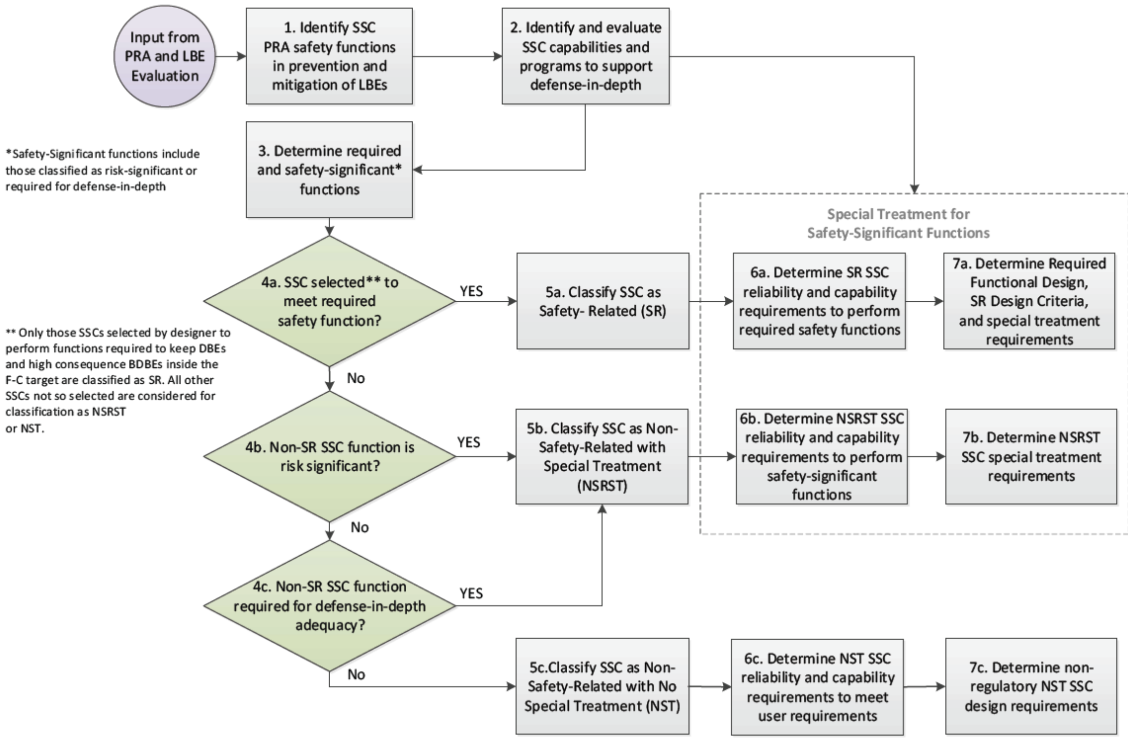

Figure 3-3. LMP SSC Function Safety Classification Process [9]

3.2.1. Task 1: Identify SSC Functions in the Prevention and Mitigation of LBEs

Task 1 in Figure 3-3 (Task 5a in Figure 3-1) requires the identification of safety functions, which are generally defined as those functions responsible for the prevention and mitigation of an unplanned radiological release from any source within the plant. The PRISM and HTGC-PBR demonstrations both identified safety functions using a method that starts with the goal of controlling radionuclide release and identifying the basis requirements that support this goal. This process is depicted in Figure 3-4 for the PRISM design, and an example for micro-reactors is presented in Figure 3-5. While the term "safety function" includes all functions that contribute to the goal of controlling release, RSFs are PSFs that are required to maintain DBEs or BDBEs inside their respective F-C target.

PRISM identified four basic safety functions from their PRA and evaluated sets of the functions against F-C targets [45]:

- Reactivity Control
- Core Flow
- Heat Removal from primary sodium
- Confinement

The PRISM safety functions are shown in Figure 3-4. The evaluation of external hazards and other plant operating states could significantly alter the margin to the F-C target shown in Figure 3-6.

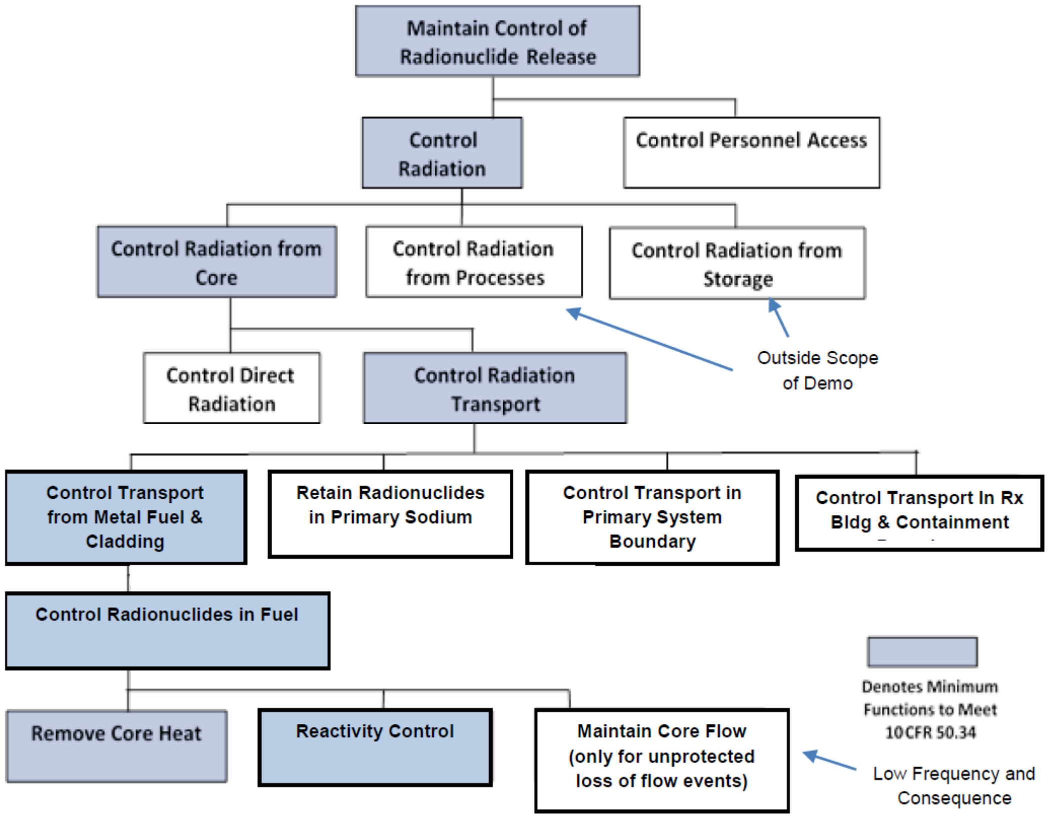

Figure 3-4. PRISM Safety Functions and Required Safety Functions [45]

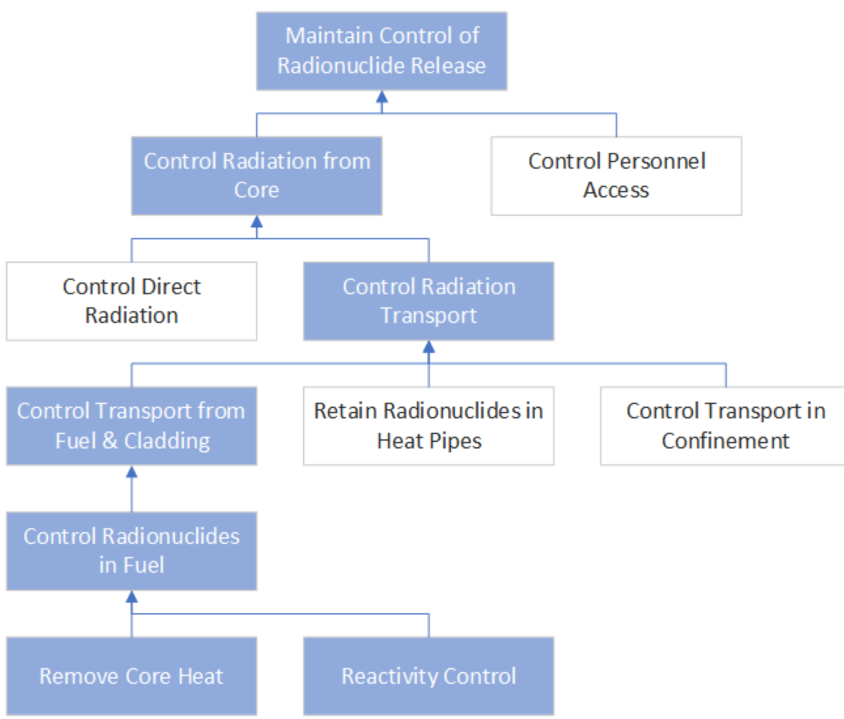

Figure 3-5. Micro-Reactor Safety Functions

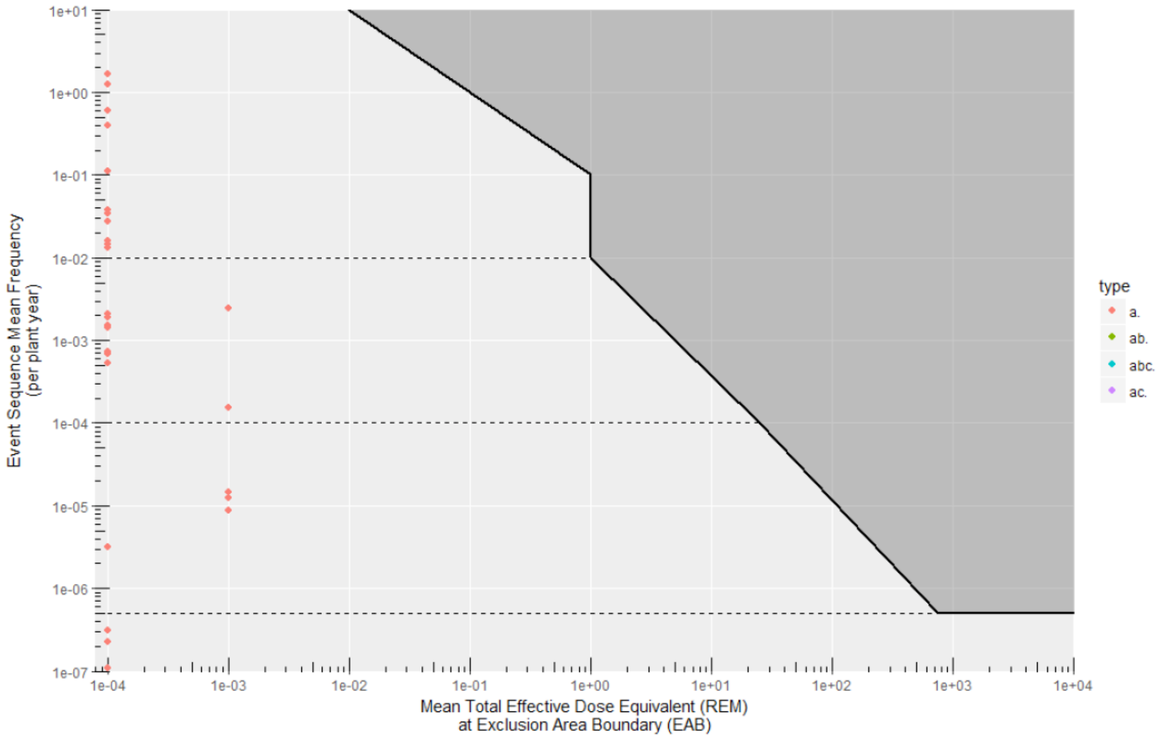

Figure 3-6. PRISM LBE F-C Chart based on Full Power Internal Events PRA [45]

Similar results are obtained for the HTGC-PBR. An example F-C target along with LBEs is shown in Figure 3-7 for the HTGC-PBR, though this is intended to only serve as illustration since it is not directly applicable to micro-reactors. The F-C chart presented in Figure 3-7 is an example addressing a relatively limited-scope PRA, addressing internal initiating events at full power. This F-C chart (Figure 3-7) illustrates the role that an SSC can play in performing an RSF. If the SSC contributes to reducing the consequences (i.e., the Total Effective Dose Equivalent at the EAB) associated with an event sequence, it performs a mitigation function. However, if the SSC contributes to reducing the mean frequency of an event sequence, without altering the consequence from the event, it performs a prevention function.

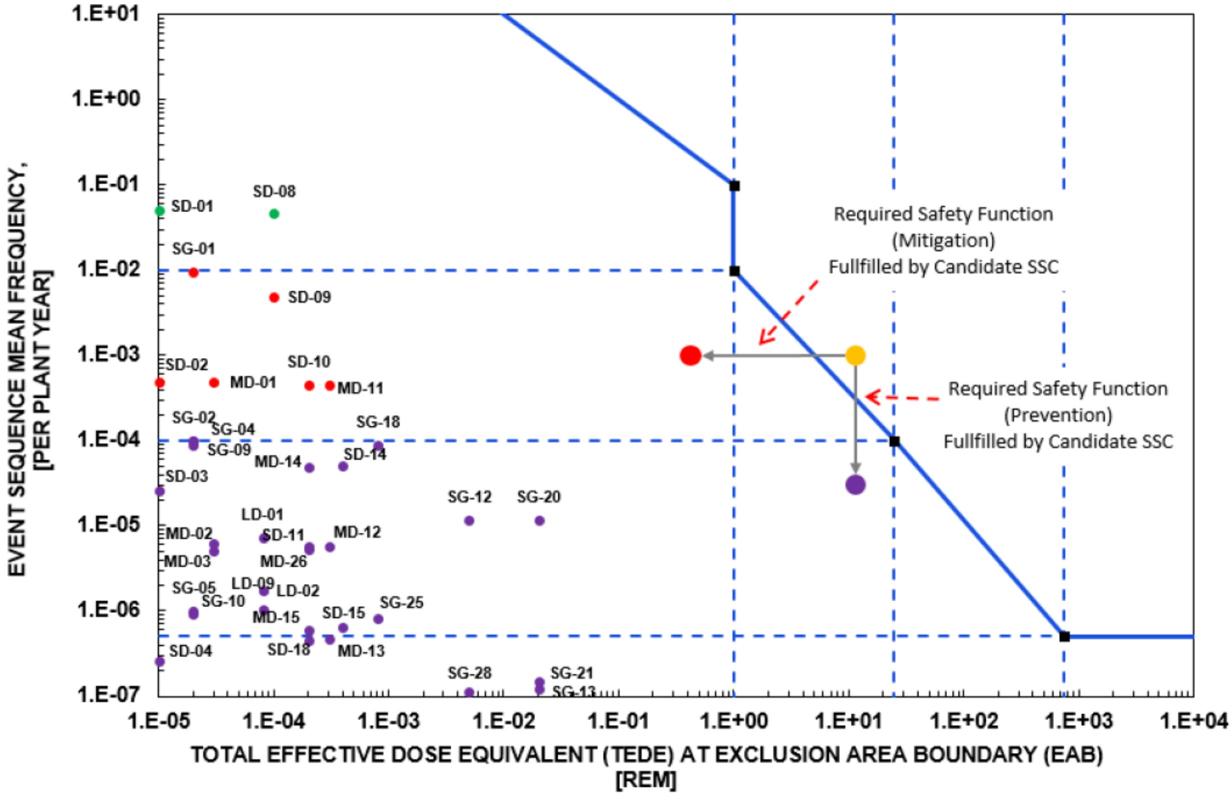

Figure 3-7. HTGC-PBR F-C Map based on Limited-Scope PRA [44]

The success of a given SSC to achieve a safety function may move an event sequence down in frequency (prevention) or down in consequence (mitigation), ideally to an area under the F-C target curve.

A micro-reactor will have a similar set of safety functions as itemized above. In the case of a heat pipe based design, however, the core flow would more specifically refer to energy transport from fissile material. This is achieved through the evaporation and condensation of working fluid in the heat pipe. A micro-reactor is expected to exhibit a relatively robust margin to the F-C target (see Figure 3-8). The eVinci LMP demonstration [38] developed a F-C chart that identified a similar isolation of high-consequence scenarios to very low frequency, as shown by the PRISM (Figure 3-6) and HTGC-PBR (Figure 3-7) LMP demonstrations. The eVinci baseline estimates are reproduced from the eVinci LMP demonstration [38] in Figure 3-8. The F-C chart shown for the eVinci Micro-Reactor LMP demonstration [38] evaluates off-site consequences without crediting environmental dispersion. This is equivalent to a dose receptor point within 1 m of the release point.

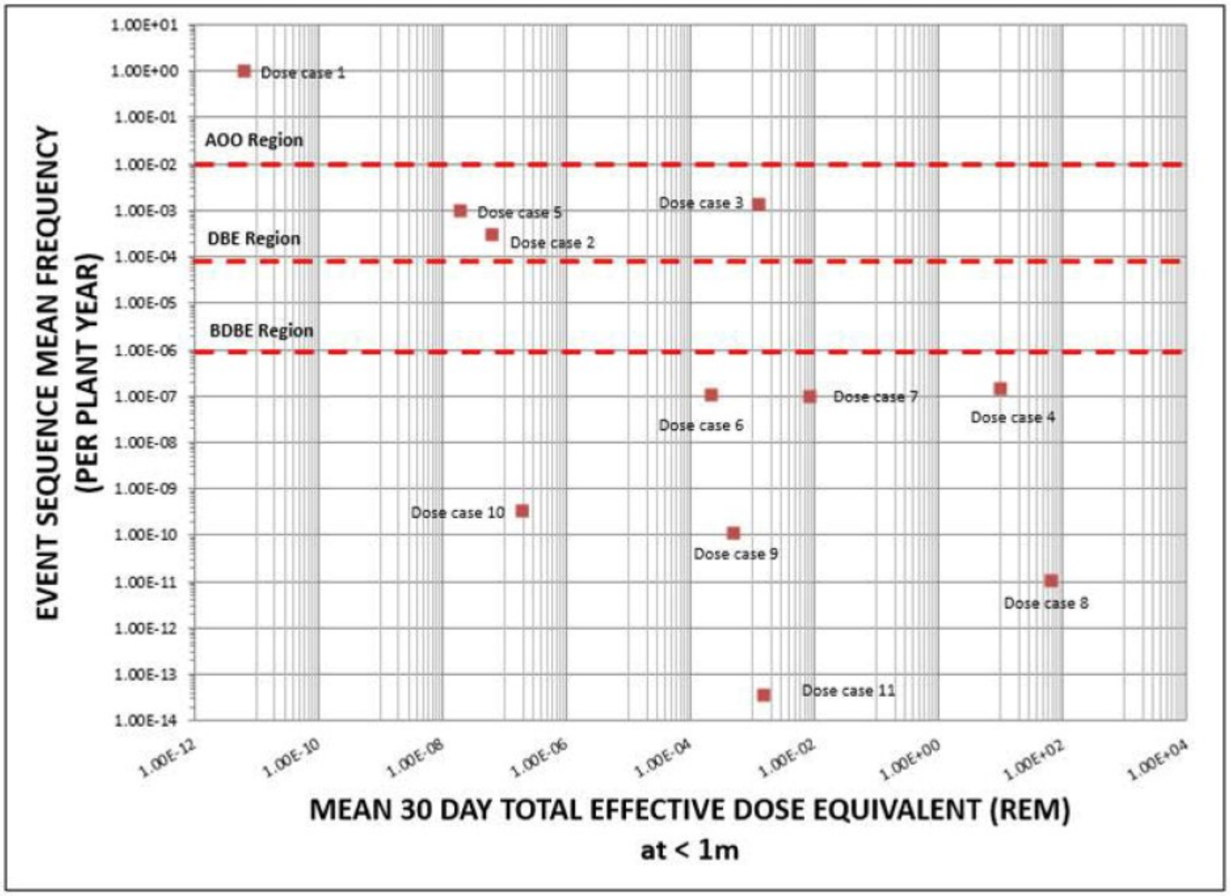

Figure 3-8. eVinci F-C Chart for Baseline PRA [38]

Despite margin to the F-C target, the role of different SSCs in achieving safety functions can be evaluated. The eVinci LMP demonstration provides an example of the role that the CCS plays in ameliorating the consequences of an event. Figure 3-9 shows the impact of the CCS on mitigating off-site consequences.

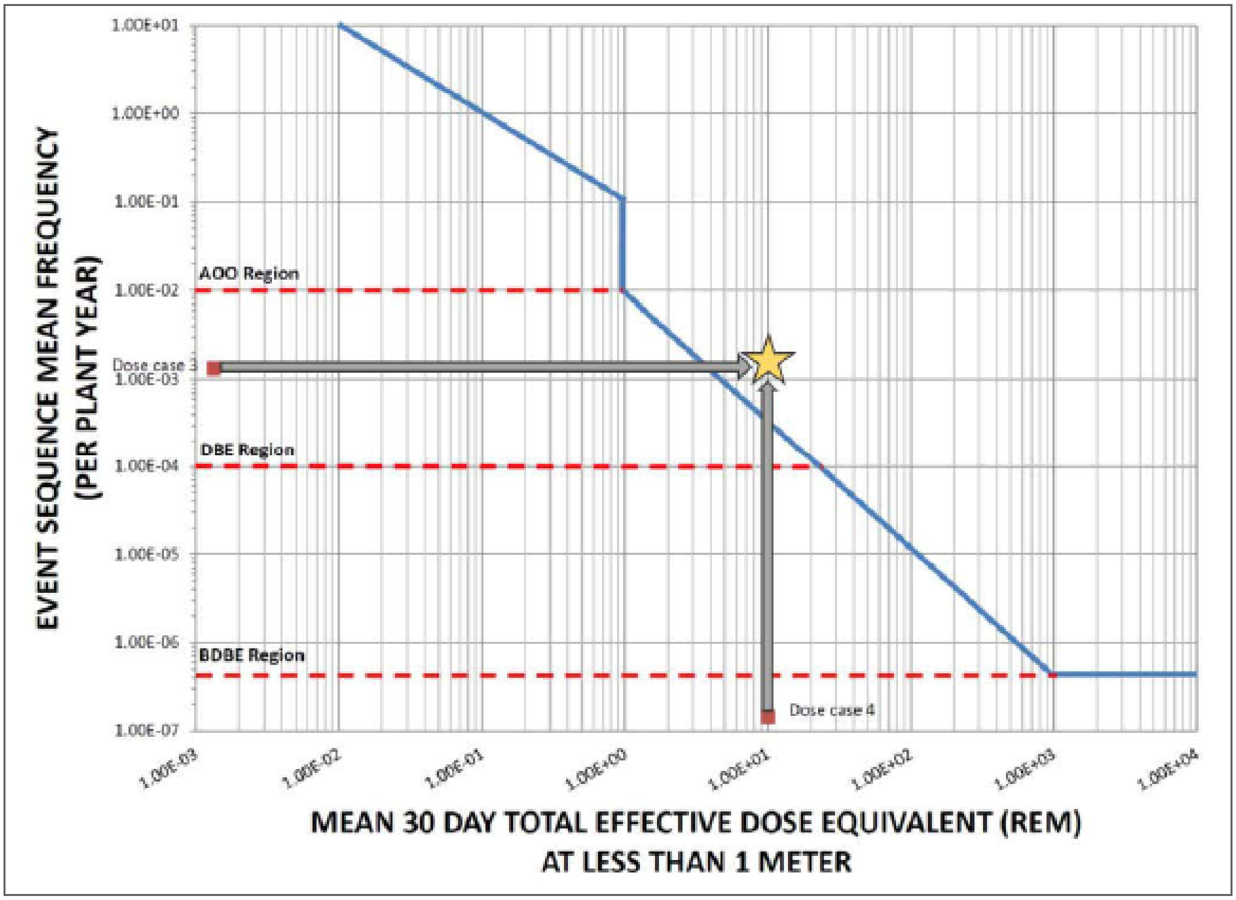

Figure 3-9. eVinci Illustration of the Role of the Canister Containment Subsystem in Mitigating Off-Site Consequences [38]

3.2.2. Task 2: Identify and Evaluate SSC Capabilities and Programs to Support DID

SSCs that are required to prevent or mitigate events may also be critical to supporting the evaluation of DID adequacy. These SSCs are safety-significant, and the robustness of the SSCs in achieving the safety functions depend on a detailed evaluation of how external events may impact SSC performance.

3.2.3. Task 3: Determine the Required and Safety-Significant Functions

Figure 3-10 illustrates the interrelationship between SSCs, highlighting the relationship between risk-significant and safety-significant SSCs. Safety-significant SSCs form a larger set than risk-significant SSCs. Safety-significant SSCs include those necessary to meet a combination of risk and DID criteria.

From this evaluation, PRISM proposed Reactivity Control and Heat Removal as their RSFs. Task 3 from Figure 3-3 selects a set of SSCs which fulfill the RSFs to be classified as safety-related. Numerous SSCs may be available to satisfy each RSF but the set chosen to be classified as safety-related must cover all DBEs and BDBEs and be capable of mitigating their consequences to within the F-C targets. SSCs necessary to support safety-related SSCs must also be considered safety-related. The final PRISM list of safety-related SSCs is [45]:

- Digital instrumentation and control systems
- Control rods and drives and associated operator actions
- Electromagnetic pump supply breakers and associated operator actions
- Power equipment
- Reactor vessel and internals
- Reactor Vessel Air Cooling System
- Supporting structures

A DBA analysis demonstrated that this set of SSCs can mitigate all considered DBAs to beneath the F-C targets. Note that both active (e.g., control rods) and passive (e.g., supporting structures) SSCs were considered.

3.2.4. Tasks 4 and 5: Evaluate and Classify SSC Functions

These two tasks evaluate and classify SSC functions.

Tasks 4A and 5A: Task 4A utilizes DBEs and high-consequence BDBEs to identify SSCs available to perform RSFs. A combination of SSCs is selected to perform each RSF such that all DBEs and high-consequence BDBEs are covered. The SSCs are classified as safety-related in Task 5A. Other SSCs are classified as non-safety-related.

Tasks 4B and 5B: Non-safety-related SSCs are evaluated to identify risk-significance (i.e., those SSCs necessary to ensure that LBEs meet the F-C target or is significant in relation to an LBE cumulative risk metric). When an SSC is risk-significant, but not safety-related, it is classified as a NSRST SSC.

Tasks 4C and 5C: SSCs that are neither safety-related nor risk-significant are evaluated in Task 4C to assess whether they should be subject to special treatment. Those SSCs that support functions necessary to achieve DID adequacy are also classified as NSRST SSCs.

3.2.5. Task 6: SSC Reliability and Capability Requirements

This task identifies requirements for reliability and capability of SSCs modeled in the PRA. Requirements for safety-significant SSCs establish specific design and special treatment requirements (Task 7). SSCs that are non-special treatment have reliability and capability requirements that fall under non-regulatory owner design requirements.

SSCs that are not considered safety-related according to the LMP may still be subject to special treatment due to having significant importance to risk metrics or DID adequacy [9].

Oklo's LMP pilot did not consider any active SSCs to be safety-related and the results of their DBA analysis met the F-C target [29]. Oklo then identified that there were no RSFs. The NRC later questioned whether the safety contributions of passive SSCs were included in this analysis and whether any passive SSCs should be considered safety-related [33].

This represents a fundamental challenge for extending PRA methods to address designs based primarily on inherent safety systems. While these systems are likely quite unlikely to experience random failure, a thorough failure mode analysis must be conducted to assess how external perturbations could arise due to, for example, external events.

3.2.6. Task 7: Determine SSC Specific Design Criteria and Special Treatment Requirements

Safety-related SSCs satisfy design criteria that are defined at a functional level. The Required Functional Design Criteria (RFDCs) are necessary to satisfy RSFs. The RFDCs are used to derive Safety-Related Design Criteria (SRDCs) for SSCs necessary to ultimately perform the RSFs.

Note that non-safety-related-special treatment SSCs do not satisfy SRDCs. However, reliability and capability requirements apply to these special treatment SSCs as part of the integrated decision-making process that is necessary to establish DID adequacy.

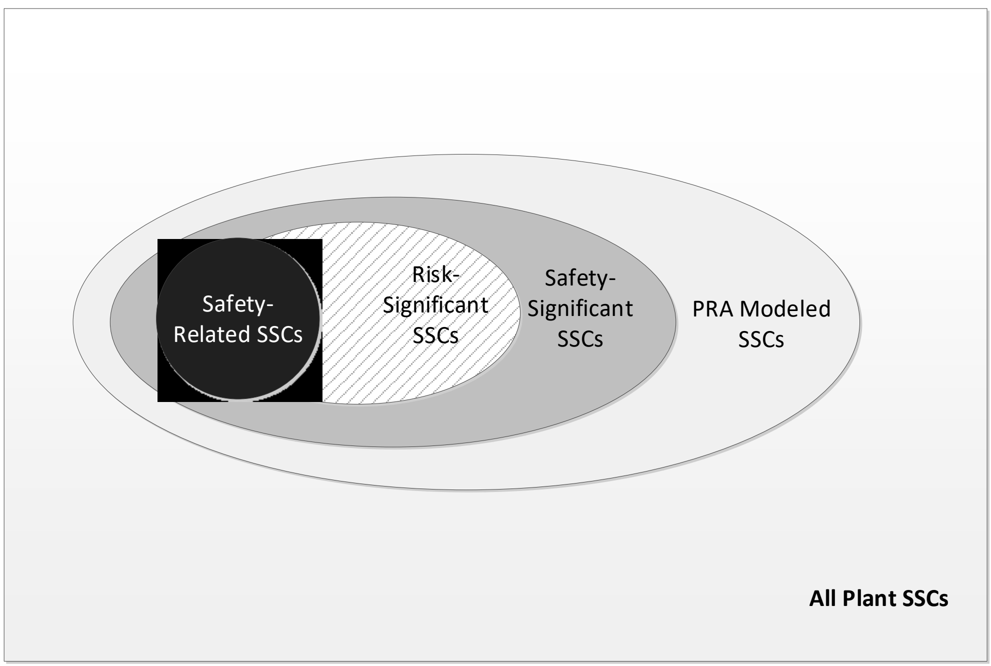

Figure 3-10. Illustration of Risk-Significant and Safety-Significant SSCs [9]

3.3. Evaluation of Defense-in-Depth Adequacy

DID is a multi-layered approach where each layer acts independently from the others with the objective of preventing and/or mitigating facility accidents that release radiation into the environment. The DID approach seeks to ensure that no single layer of protection is exclusively relied upon to prevent and/or mitigate accidents that release radiation or hazardous materials to the environment.

Figure 3-11 depicts the LMP DID evaluation process. This process requires iteration in several areas as marked by the triangles. The DID adequacy evaluation is carried out by a group called the Integrated Decision-Making Panel (IDP), formed by the reactor designer. The IDP for advanced non-LWR design is comprised of those responsible for the necessary deterministic and probabilistic evaluation of the design, operations, and maintenance program [9].

Following the diagramed tasks in Figure 3-11, the initial design capabilities are established with considerations made for owner requirements, regulatory and licensing considerations, and associated safety, operating, and maintenance practices. F-C targets are established following regulatory objectives, as well as defining SSC safety functions for use in the PRA. After the scope of the PRA (i.e., hazards, initiating events, and event sequences) has been determined, the risk analysis is completed.

The LMP adopts layers of defense as part of a risk-informed, performance-based approach to evaluating plant capabilities and programs as shown in Figure 3-11. These layers of defense are similar to the IAEA levels of defense.

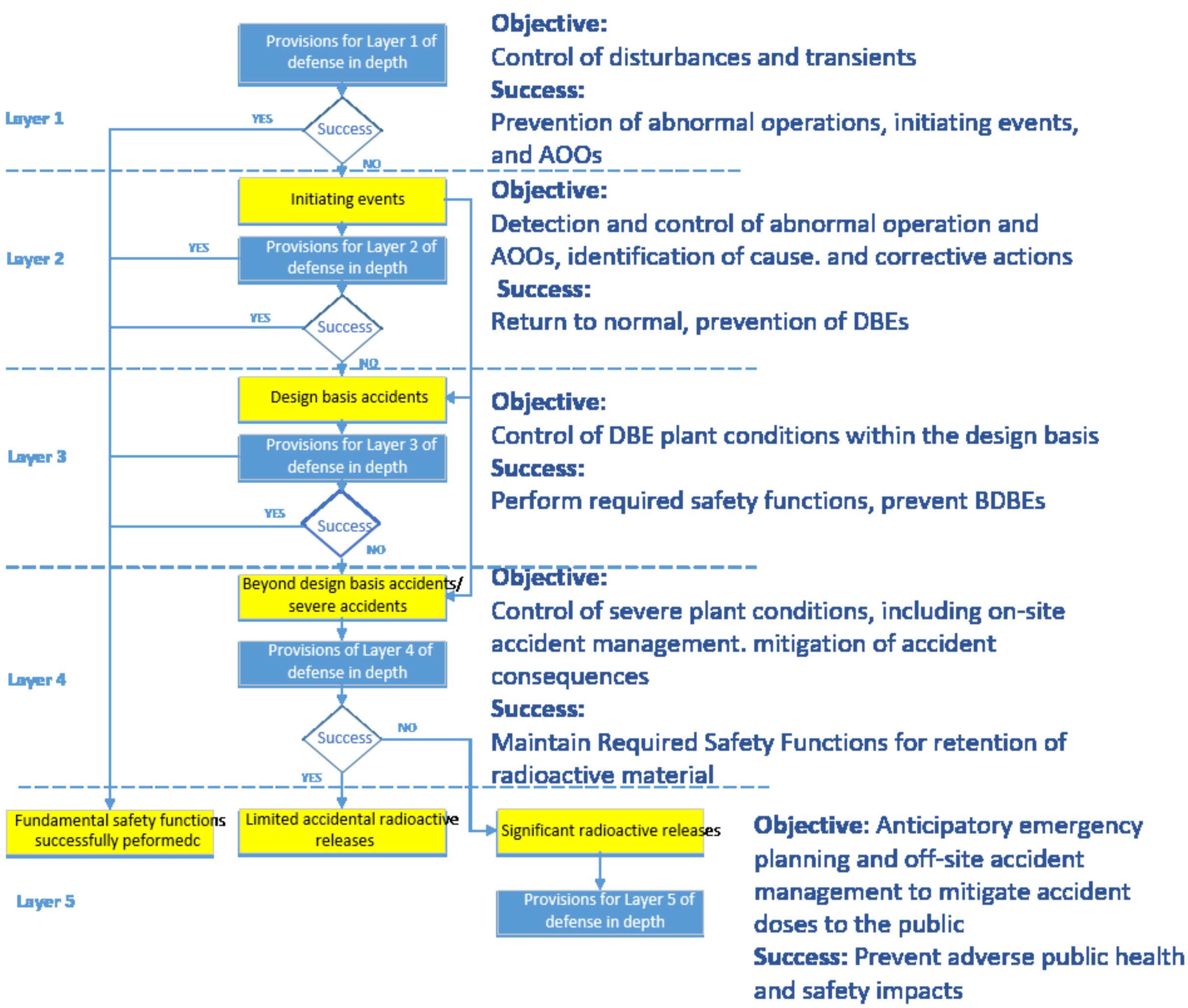

Figure 3-11. LMP Evaluation of LBEs in terms of Layers of Defense [9]

The evaluation of DID adequacy follows the subsequent set of tasks, as illustrated in Figure 3-12. As noted in the LMP [9], a number of these tasks are performed as part of the efforts to select LBEs and/or perform safety classification of SSCs. In the subsequent discussion, how the iterative process of evaluating DID adequacy relies on other components of the LMP is specified.

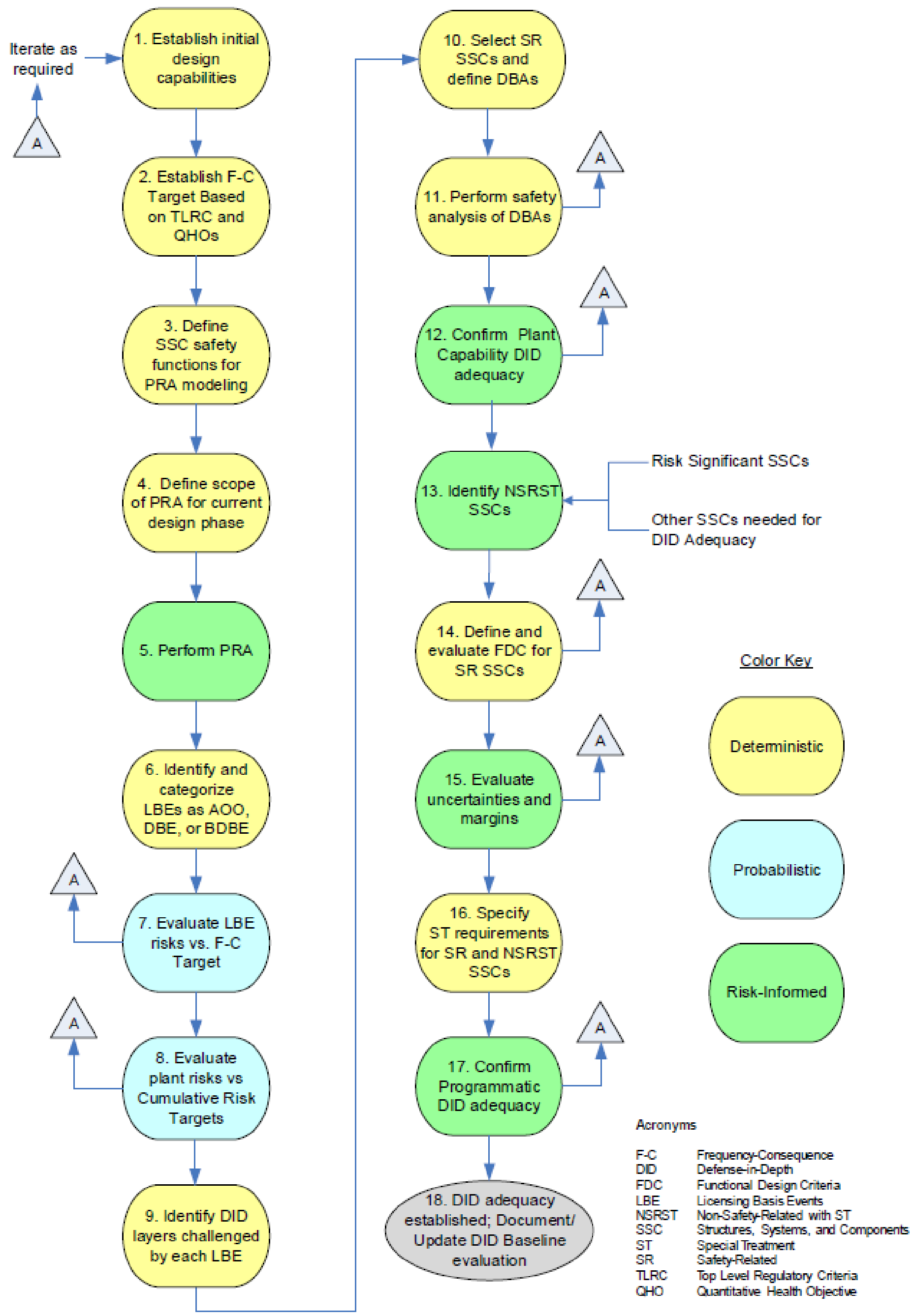

Figure 3-12. LMP Integrated Process for Incorporation and Evaluation of DID [9]

3.3.1. Task 1: Establish Initial Design Capabilities

At an early stage of design, a range of stakeholders contribute to formulating the capabilities of a design. These capabilities often contribute to DID. At the same time, formulation of the safety design philosophy for a new reactor may not always align in a classical sense with DID. For micro-reactors, the significant reliance on inherent safety systems may not support multiple layers being available to achieve different safety functions. One example is heat transport away from fissile material.

The robustness of these functions, however, does not satisfy DID. An important example of a challenge to DID is that inherent safety functions may be robust against random failure (i.e., the challenge presented by an internal events PRA), but not capable of maintaining the safety function subsequent to an external event (e.g., a seismic event damaging the structural integrity of a heat pipe manifold).

3.3.2. Task 2: Establish F-C Target Based on Regulatory and Quantitative Health Objectives

At this stage, the F-C target is established in order to support subsequent evaluations of DID adequacy. This adequacy evaluation focuses on LBEs that have the highest levels of risk-significance.

3.3.3. Task 3: Define SSC Safety Functions for PRA Modeling

Reactor-specific PSFs are identified at this stage. These PSFs encompass a range of FSFs that are common to all reactor technologies. The identification of SSCs to be modeled in the PRA to support the PSFs enables an assessment of DID. A reasonable sense of the adequacy of DID can be obtained at this stage.

Designs such as the micro-reactor may not provide a clear demonstration of DID in functions such as fission energy transport (i.e., core heat removal) at this stage of the adequacy evaluation. A primary or singular reliance on inherent safety presents a challenge to DID adequacy. These designs may be susceptible to challenges from difficult to characterize external hazards. The lack of complementary systems that realize safety functions through either passive or active means is the principal challenge for the micro-reactor.

While micro-reactors are low-power with robust inherent safety systems, the perceived level of safety primarily applies to internal initiating events under conditions without nearby population. Should micro-reactors be deployed in higher density population zones, with smaller EABs, the potential consequences from an event may be significantly greater than assumed by extrapolating insights from LWR consequence evaluations.

3.3.4. Task 4: Define Scope of PRA for Current Design Phase

The scope of a PRA for a micro-reactor should consider a relatively extensive range of hazards, and initiating events at the earliest stages of a design. Significant design changes from the initial concept may be required to address external events, despite the relative robustness of inherent safety systems to random failures.

3.3.5. Task 5: Perform PRA

The conduct of the PRA is essential to developing risk insights at different phases of the design. Numerical estimates are often less useful during the design, since risk insights can be essential to establishing a robust design that can accommodate a range of hazards and initiating events. Given the robustness of inherent safety systems, numerical estimates during the earlier design phases may bias a reactor designer to an assessment of high safety margins. These safety margins could be degraded as the scope of the PRA is extended, particularly if the design does not have sufficient levels of DID present at an early stage. A particularly important challenge for non-LWR designs relying on passive safety systems is the extent to which perceived reliability is reduced as the scope of the PRA is extended to consider external events.

3.3.6. Task 6: Identify and Categorize LBEs as AOOs, DBEs, or BDBEs

This task is completed under the task in Section 3.1.4. Of relevance to micro-reactors, or any other design that relies on passive safety systems (i.e., systems that are robust to random failure), is the vulnerability to external events or malicious hazards. Frequency of occurrences for these types of initiating events are generally hard to characterize and subject to large uncertainties. Such uncertainties can potentially lead to an LBE being more frequent over the life of a new technology than indicated by historical data or expert judgment. While the LMP provides discussion of the impact of LBE uncertainties, it is critical that DID can ameliorate uncertainties in initiating event frequency. Robust DID ensures that a plant has multiple means of protecting one or more barriers to fission product release to the environment. DID is also central to ensuring that a plant is less susceptible to single-point vulnerabilities (or cliff edges) in a system that emerge from difficult to characterize initiating challenges to plant safety.

3.3.7. Task 7: Evaluate LBE Risks versus F-C Target

This task is completed under the task in Section 3.1.8.1. The evaluation of LBE risks provides a means by which designers can focus on events that contribute the most to overall plant risk. As noted above, a micro-reactor may be particularly sensitive to external events or malicious hazards. These events are relatively difficult to quantitatively characterize, with initiating event frequencies subject to significant uncertainties (in some cases an initiating event may not be recognized as possible at the time of plant design). DID has traditionally accommodated uncertainties, ensuring that a design is robust to even the least well-characterized challenges to plant performance and safety.

For designs relying significantly on passive safety systems, the margin to exceeding the F-C target for many internal LBEs (i.e., arising from random failure of plant SSCs) is likely to be significant. This was identified in the eVinci LMP demonstration [37]. However, uncertainties in characterizing risk from external events or malicious hazards may be (and typically are) quite large. From a DID perspective, the magnitude of uncertainty in risk quantification is also an important consideration when focusing attention.

DID should also be targeted to areas where risk quantification uncertainties are large. A primary goal of risk management is establishing enhanced levels of safety through focusing design and operations in a manner to address the most likely sources of risk to public health and safety. The process of estimating risk involves acknowledgment or quantification of uncertainty in the future state of a plant due to either operation of internal SSCs or emergent challenges through the plant's interaction with environmental, socio-political, or other technological systems. The quantitative estimation of a probability (or frequency) of occurrence of some consequence, as a result of an accidental future plant state being realized, represents an evaluation of the uncertainty that arises from stochastic sources (i.e., the inability to *completely* control the future state of a technological system).

The median frequency estimate of an accident consequence provides a quantitative level of uncertainty in the future state of a technological system. This frequency does not capture how uncertainty in the estimate, or the uncertainty in the form of the probabilistic model itself, impacts the quantitative characterization of risk—the uncertainty in the frequency estimate itself. From a Bayesian perspective, the median frequency is derived from the *prior* estimate of the probability of the frequency of occurrence for an accidental consequence. Uncertainties in the median estimate are captured by the distribution of occurrence frequencies described by the prior occurrence frequency distribution. An implicit assumption in conducting PRAs is that characterization of the prior occurrence frequency distribution has incorporated sufficient experience or data that any subsequent update in experience or data will not lead to a *posterior* occurrence frequency distribution that is significantly different from the *prior* occurrence frequency distribution. It is not clear that, for difficult-to-characterize external events or malicious hazards, the prior occurrence frequency distribution is supported by sufficient experience to assume it has approximately converged to the *posterior* occurrence frequency distribution.

This source of uncertainty (often considered to be a model form uncertainty in modeling and simulation applications) is critical to consider when approximate convergence of the prior occurrence frequency distribution to the posterior occurrence frequency distribution cannot be demonstrated. In light of the potentially significant shift in the risk profile from internal events for a design relying on passive safety systems, DID remains a critical means to ameliorate the impact of significant uncertainty in characterizing the occurrence frequency distribution itself.

3.3.8. Task 8: Evaluate Plant Risks versus Cumulative Risk Targets

This activity is performed as part of LBE evaluation Task 7b (see Section 3.1.8.2).

3.3.9. Task 9: Identify DID Layers Challenged by each LBE

Figure 3-11 presents the layers of defense process used to evaluate each LBE. More attention is given to risk-significant LBEs, identifying and evaluating the DID attributes supporting capabilities for each layer. As part of this process, dependencies among the layers are also minimized. For designs relying significantly on passive safety, it is critical to ensure that the design is not sensitive to uncertainties in the probabilistic model utilized to characterize the risk. It is worthwhile at this stage to ensure that DID adequacy can be demonstrated as ameliorating this type of "model form" uncertainty that challenges the robustness of direct application of PRA results.

3.3.10. Task 10: Select Safety-Related SSCs and Define DBAs

This task is performed as part of the characterization of LBEs, as discussed in Section 3.1.8.4 above.

3.3.11. Task 11: Perform Safety Analysis of DBAs

This task is performed as part of the characterization of LBEs, as discussed in Section 3.1.8.4 above.

3.3.12. Task 12: Confirm Plant Capability DID Adequacy

This task evaluates the adequacy of the plant capabilities for DID. As part of this adequacy evaluation, LBEs are evaluated to ensure that risk targets are met so as to not exclusively rely on a single design element, program, or DID attribute.

3.3.13. Task 13: Identify Non-Safety-Related with Special Treatment SSCs

This task is performed as part of SSC safety classification (see Section 3.2.2).

3.3.14. Task 14: Define and Evaluate Required Functional Design Criteria for Safety-Related SSCs

This task is performed as part of SSC safety classification (see Section 3.2.3).

3.3.15. Task 15: Evaluate Uncertainties and Margins

This task is intended to address many of the issues around uncertainty in risk quantification discussed above. The LMP directs that a design's layers of defense should be utilized to compensate for "residual unknowns" [9]. This includes offsite response.

For designs relying on passive safety, this stage of the DID evaluation process may provide inadequate flexibility to address uncertainties that affect the PRA. The design philosophy by this stage may be too well-defined to allow sufficient adjustment to ensure DID can ameliorate uncertainties in the risk quantification. Practically, this may lead to significant DID iteration or a premature exit from this DID adequacy evaluation. It is critical that the susceptibility of the design to uncertainty in risk quantification be critically evaluated as early as possible, particularly for designs relying heavily on passive safety systems.

3.3.16. Task 16: Specify Special Treatment Requirements for Safety-Related and Non-Safety-Related Special Treatment SSCs

A consideration in setting SSC performance requirements is the need to assure DID adequacy is realized not just in the design, but also in the as-built and as-operated and maintained plant. It is essential that these performance requirements be maintained through throughout the life of the plant.

3.3.17. Task 17: Confirm Programmatic DID Adequacy

Programmatic measures supporting DID adequacy is based on the performance requirements established in Task 16. As noted in the LMP [9], the programmatic measures are evaluated utilizing SSC risk-significance, how SSCs contribute to different layers of defense, and special treatment effectiveness in providing additional confidence in risk-significant SSC performance.

3.3.18. Task 18: DID Adequacy Established—Document/Update DID Baseline Evaluation

The evaluation of DID adequacy continues until risk-significant vulnerabilities are no longer identified. It is important to note that potential compensatory actions may not make a practical, significant improvement to the LBE risk profiles or risk-significant reductions in the level of uncertainty in characterizing the LBE frequencies and consequences. Documentation of either no risk-significant vulnerabilities or no practical compensatory actions are essential to demonstrate the adequacy of DID.

For designs relying on passive safety, it is important that documentation be provided demonstrating how DID has been utilized to reduce uncertainty in the probabilistic model form used to characterize risk.

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4. TECHNICAL ISSUES ASSOCIATED WITH APPLYING THE LICENSING MODERNIZATION PROJECT TO MICRO-REACTORS

The fundamental technical challenge posed by micro-reactors is the availability of methods by which the level of safety achieved by a reactor design can be evaluated. This section is structured in terms of the following technical issues that are critical to establishing and supporting methodologies for quantitatively evaluating the level of safety within the framework of PRA and risk-informed decision-making methods established for the operating fleet of LWRs.

- Methodologies to evaluate risk
    - Selection of the spectrum of events that could result in a reactor entering states posing elevated risk to public health and safety
    - Quantitative evaluation of the potential for the spectrum of events to elevate risk to public health and safety from operation of a micro-reactor
- Available operational experience against which to assess the completeness of a priori evaluations of the potential and quantitative likelihood of a micro-reactor entering into states that could pose risk to public health and safety
- The evaluation prior to and during operation that the reactor is being maintained within a state that does not elevate the likelihood that it could enter a state adverse to public health and safety

Since risk-informed decision-making is becoming an increasingly important part of non-LWR safety evaluation, the ability to quantify risk to public health and safety is essential. It is necessary to resolve the technical or methodological challenges to evaluating risk for reactor designs relying heavily or exclusively on inherent safety. Doing so enables a natural integration into an evolving risk-informed, performance-based decision-making framework. The focus of the following discussion is on providing the framework to achieve this.

4.1. Licensing Basis Event Selection

Section 3.1 provides a discussion of the identification of LBEs for a micro-reactor. The safety functions provided by any nuclear reactor design are

- Control – maintain power generation within a pre-defined range
- Cool – provide functions to remove heat generated within fuel
- Contain – prevent transport of radioactive material outside of the facility

Each of these FSFs are discussed in the following sections.

4.1.1. Reactor Power Control

This first safety function is provided by a combination of an active safety system and an inherent physical effect.

- B_4C control drums are an active means to regulate the amount of negative reactivity insertion. At the beginning of a cycle, a HPR such as Megapower may have as much as $3.00 excess reactivity. With 12 control drums, each having a reactivity worth of about $1.10, there is a total of $13.20 of negative reactivity. There is thus substantial negative reactivity available to accommodate the initial core excess reactivity. A misalignment of control drums, however, could result in a TOP.

- Inherent negative reactivity feedback is also available to arrest transient over-power scenarios. In the event of a core power transient, sufficient additional heat generation will cause a temperature excursion of the metal hydride moderator. At elevated temperatures, hydrogen is released from the metal hydride. This has the effect of inserting negative reactivity into the reactor core.[9]

It is reasonable to assume that the design basis arrests transient over-power events before damage to the fuel or heat pipe monolith occur. With fuel and reactor structural integrity maintained, transport of radionuclides into either the reactor enclosure or the condenser section of the PHX would be prevented. Without more detailed understanding of the reactor response to reactivity transients, it is conceivable that some of the rarer, but higher reactivity insertion events, could lead to a challenge to fuel and reactor structural integrity. In this case, the end state of these more severe over-power transient event scenarios could result in radiological release to the environment. As a result, measures necessary to maintain heat removal from the fuel may not be sufficient to prevent radiological release from the core. Releases may be mitigated by provision of cooling in the long-term (reducing thermally-driven diffusion of fission products out of the damaged fuel). Furthermore, additional confinement of fission products transported from the fuel into heat pipes and interfacing systems would also serve to mitigate the extent of radiological release to the environment. It should be noted that it has been postulated that the end state of a positive reactivity excursion event in a Canada Deuterium Uranium (CANDU) reactor could be potentially less severe should the Emergency Core Cooling System (ECCS) remain available to deliver water to the rubblized core material [53]. However, it is generally difficult to establish with high confidence the coolable state of core debris after a reactivity excursion event. CANDU reactor PRAs generally consider that positive reactive excursion events result in large fission product release to the environment.

Transient over-power initiating events typically involve additional questions regarding plant response such as

- Is the reactivity transient arrested prior to failures of the fuel or reactor structure?
- If the fuel or reactor structural integrity is lost, are fission products released?
  - Are fission products released into the condenser section of the PHX?
  - Are fission products released into the reactor enclosure?
- If fission products are released from the reactor, can the fission product release go to the environment?

[9] The Oklo core exhibits negative temperature feedback coefficients of reactivity. Under transient conditions, these feedbacks contribute to the return of core reactivity to a safe condition. The negative reactivity feedbacks in the Oklo reactor include the fuel thermal expansion coefficient of reactivity, the fuel Doppler coefficient of reactivity, and the structural material thermal expansion coefficient of reactivity.

- Does fission product release into the environment occur via a release pathway from the PHX condenser section into the environment?
- Does fission product release into the environment occur via a release pathway from the reactor enclosure into the environment?

The previous questions illustrate aspects of plant response that are typically relevant to assessing the plant behavior following an initiating over-power event. The event progression has some similarities to higher power water-moderated and non-LWR reactor concepts. Because the heat pipe also functions as a fission product barrier, its performance requires additional considerations. Such evaluations, however, have been performed in the past for CANDU reactors. In particular, CANDU analyses exist for the pressure tube and calandria tube response in loss-of-coolant, loss-of-reactivity control or loss-of-power regulation accidents [53]. Beyond typical assessments of the CANDU design basis, the multiple-fuel channel design of a CANDU reactor has necessitated the definition of a broader range of radiological release categories in PRAs. This is necessary to capture the fact that radiological release scenarios can occur with damaged fuel limited to one fuel channel (or in rarer cases a few fuel channels). This potential must be considered for a number of HPR concepts.

4.1.2. Reactor Heat Removal

As with traditional, water-moderated reactor designs, removal of heat generated within fuel is performed through transfer of energy to a working fluid. Energy accumulated in this working fluid is transported to a heat sink (e.g., the PHX or DHX in a micro-reactor) through motion of the fluid. In the case of a HPR, the motion of the working fluid occurs within each individual heat pipe as vapor generated in the heated evaporation region flows towards the cooled condenser region and capillary forces promote the return flow of condensed fluid back to the evaporation region. Further discussion of the physical processes governing the behavior of a heat pipe is provided in Section 4.2.1.

Under normal operating conditions, heat transfer in an HPR thus relies on,

- Integrity of the array of heat pipes containing the working fluid (heat pipe integrity)
- Integrity of the (possibly high-elevation) condenser region of each pipe (condenser integrity)
- Maintenance of sufficient heat rejection from the condenser section into the PHX or DHX heat sinks (loss of heat removal function)

The heat transfer mechanisms can be challenged in several different ways.

- Heat Pipe integrity

  Loss of structural integrity of one or more heat pipes can occur through different means. Failure of a single heat pipe could potentially propagate and fail multiple heat pipes. Alternatively, failure of multiple heat pipes could arise due to a common mode failure. Such a common mode failure would occur as a result of, for example, defects introduced during manufacturing.

  This type of event is similar to a LOCA in a water-moderated reactor. Relative to a CANDU reactor, the failure of a single heat pipe would be analogous to the failure of a single fuel channel. As in the case of CANDU reactors, where multiple, simultaneous fuel channel failures is an incredible event, the simultaneous failure of multiple heat pipes is likely rare.

Unlike CANDU reactors, however, the consequential failure of multiple heat pipes following an initiating single heat pipe failure cannot be ruled out. In CANDU reactors, safety research has established the potential for a single fuel channel failure to induce additional fuel channel failure to be incredible. Only in the case of a reactivity excursion event (like that caused by a large break LOCA) will multiple fuel channel failures occur, as established by safety research. However, these failures occur because of the energy deposited in each fuel channel during the rapid reactivity excursion and not due to mechanical loading of fuel channels in the vicinity of a failed channel.

- Condenser integrity

  The integrity of the condenser region of a heat pipe can be challenged for several reasons. The loss of integrity of the condenser is similar to failure of steam generator tubes in water-moderated reactors. As in the case of water-moderated reactors, failure of the primary-secondary boundary is most likely to occur with a small failure area (e.g., a single steam generator tube leaks or failures). Gross failure of the condenser structure should be much less likely. Past consideration of gross primary-secondary boundary failure has been considered as part of the safety evaluation for Russian-designed water-moderated plants[10] due to the horizontal steam generator design. In that reactor design, the collectors in the horizontal steam generators could experience a gross structural failure. It is not possible to rule out a very large failure area. Thus, Water-water Energetic Reactors (VVERs) consider a primary-to-secondary side (PRISE) scenario as part of their safety basis.

  Depending on the response of the PHX or DHX to a single heat pipe condenser failure, the possibility of this leading to a gross loss of heat removal for multiple heat pipes should be evaluated. This is an additional consideration. Unlike water-moderated reactors, injection of water to the secondary side is not a viable option to restore secondary side heat removal in the event of condenser integrity loss.

- Loss of heat sink

  Failure of the function of the PHX occurs with a loss of condenser function. Energy will accumulate in the working fluid inside multiple heat pipes. With reduced heat rejection to the working fluid in heat pipes, the fuel and structure of the micro-reactor monolith heat up. Damage to the fuel or reactor monolith could occur as a result of unmitigated heatup. This event is similar to a loss of heat sink in water-moderated reactors.

  Depending on the micro-reactor concept, a loss of the PHX can be mitigated through a backup heat removal system. Micro-reactor concepts, such as the Oklo micro-reactor, provide a DHX with capacity sufficient to reject power generated within the reactor following shutdown. The Oklo micro-reactor is designed around a concept where heat generated within the reactor is dissipated into the reactor enclosure. At the time of issue for this report revision, the Oklo DHX design relies on natural air circulation within the reactor

[10] These Russian-designed nuclear power plants are called in English Water-Water Energetic Reactor. In Russian, this LWR design is called *vodo-vodyanoi energetichesky reactor*, leading to the more commonly used acronym VVER.

enclosure to passively transfer the energy from the enclosure atmosphere into the environment [29].

This passive heat removal function will likely be provided across multiple micro-reactor designs because of the inherent safety it provides. Traditional PRA methods are not applicable to evaluating the reliability of passive safety measures. These methods have generally been developed to assess the reliability of active systems subject to the random failure of system components required for overall system function.

When system performance of a safety function is based on an inherent or passive feature (i.e., a naturally occurring physical process), evaluation of system function requires application of different methodologies. As an example, the NuScale Final Safety Analysis Report (FSAR) evaluated in Chapter 19 the reliability of the passive ECCS and DHRS. The methodology to evaluate passive system reliability adopted in the NuScale FSAR was structured into multiple stages.

- Evaluate thermal hydraulic response of the passive safety system functioning by natural processes that involve low forces driving natural circulation.
- Identify modeling uncertainties that impact the thermal hydraulic response of the system and establishment of low driving force circulatory flows.
- Evaluate system response with consideration of modeling uncertainties to characterize conditions where the passive system does not maintain the plant within acceptable limits.
- Integrate passive system function into accident progression development through at least a branch reflecting successful operation and a branch reflecting unsuccessful or degraded operation.

NOTE: More generally, it may be necessary to classify system operation into a number of different plant operating states (POSs), ranging from operational to non-operational. Should the plant response be sensitive to operation of a system in different modes of degraded operation, it would be necessary to integrate these modes into accident progression evaluations as is normally done in Level 2 PRA event trees for LWRs. In the case of LWR severe accident progression (Level 2 PRA), the development of an Accident Progression Event Tree (APET) normally requires specific phenomenological top events be categorized into multiple outcomes (i.e., event tree branches). For example, in LWR Level 2 PRA, multiple modes of containment failure arise and impact the magnitude and/or timing of radiological release to the environment. It is normally necessary to define multiple event tree branches to be able to distinguish the impact of containment failure mode on radiological release. The development of Level 2 PRA for non-LWR concepts, however, is not necessary since the transition into Level 2 PRA occurs only when core damage occurs. To capture a broader range of reactor concepts, the LMP [9] is structured such that non-LWR PRAs evaluate risk only in terms of radiological release to the environment. This ensures that the LMP encompasses PRAs for reactor concepts without the LWR solid-fuel reactor core that, upon damage, can release substantial amounts of radioactive material. In the case of, for example, molten fuel systems there is no reactor core to be damaged and radiological release from the fuel or reactor system does not have a traditional correlate with core damage.

While the above discussion applies to evaluating thermal-hydraulic performance of a system, passive SSC response applies more broadly across the other basic safety functions. Further discussion of inherent safety is provided in Section 4.2.

4.1.3. Containment of Radioactive Material

Reactor designs traditionally include several passive or active barriers to radionuclide transport from the nuclear fuel to the environment. These controls on the radiological hazard constitute a consequence mitigation strategy aimed at minimizing the risk to public health should fission products escape their initial confinement (e.g., the fuel elements). Any obstacle that impedes fission product transport to the environment can be credited as part of the mitigation strategy. Note that off-site measures, evacuation planning, and radionuclide hold-up to allow for radioactive decay are also aspects of the consequence mitigation strategy. Regardless of the micro-reactor concept in question, a mitigation strategy for radionuclide release is a necessary design feature. The robustness of the mitigation strategy may depend on variables such as total radionuclide inventory and siting characteristics.

Mitigation interfaces with the LMP framework occur at several points:

- Mitigation is performed by SSCs
- Mitigation factors into the classification of SSCs
- Mitigation relies on at least one (almost certainly two or more) DID layers
- Mitigation helps to control the radiological hazard such that LBEs meet F-C criteria

For a given LBE, SSCs comprising the mitigation strategy intervene in multiple and diverse ways. When screening LBEs according to F-C criteria:

- the physics of active and passive barriers must be assessed for their radionuclide retention capability, and
- the release timing must be considered.

There are technical issues pursuant to mitigation strategies in the context of the LMP framework as applied to micro-reactors. They include but are not limited to:

- Technical/analytical challenges and data needs for analysis of fission product barriers
    - Metal-fuel behavior (radionuclide release) for assembly concept micro-reactors
    - Radionuclide migration through and release from fuel elements of both monolith and assembly concept micro-reactors (e.g., the metal fuel and heat pipes)
    - Thermal/mechanical behavior of core structures in both monolith and assembly concept micro-reactors, which differ between the monolith and assembly concepts
    - Fission product chemistry (e.g., interaction with Na/K should heat pipe in-leakage occur)
- Paradigm shift for atmospheric radionuclide transport modeling

  - Traditional far-field atmospheric transport models (e.g., puff release and Gaussian plume) may be inappropriate for evaluating dose (consequences) if site boundaries are close
  - Alternatives include near-field atmospheric transport models
- Potential need for ground radionuclide transport modeling
  - Micro-reactors may be installed below-grade using a functional containment
  - Possible need to consider consequences in terms of under-ground radionuclide transport

4.1.4. Generalized Accident Progression Event Tree for Micro-Reactors

The above discussion identifies the basic safety functions of any nuclear power reactor with consideration of these functions in the context of a generic micro-reactor. Since PRAs are developed to express how these different safety functions are implemented in the context of a nuclear power plant design or facility, it is possible to develop a generic APET appropriate to a generic micro-reactor.

NOTE: In LWR PRAs, an APET is typically developed to represent inherent plant response in the event of loss of installed (active) safety functions. The loss of these installed safety functions, in the context of an LWR PRA, is what drives a plant to core damage. The subsequent accident progression, upon accounting for the function of active containment safety functions (e.g., containment sprays) with a containment bridging event tree, generally is represented by an APET. An APET is also referred to as a Containment Event Tree (CET). The APET is generally what characterizes the LWR Level 2 PRA, in which inherent plant response to core damage is evaluated with the end goal of assessing the range of radioactive release scenarios that could occur.

The concept of a Level 1 PRA event tree for micro-reactors that assesses the range of scenarios of active safety system failures that could result in core damage, is not directly applicable. Since inherent safety features in a micro-reactor represent the inherent response of the plant to a range of initiating events, the development of generic accident evolution commences with the introduction of an APET. The micro-reactor APET tracks the inherent response of the plant to an initiating event.

As noted above, a micro-reactor achieves the different safety functions as follows,

- Engineered reactivity control system (e.g., control drums) are available to increase or decrease reactivity. Following an initiating event, an engineered reactivity control may be necessary to control any unsafe increase in core reactivity. There are two possible outcomes considered for this safety function or “top event”:
  - Reactivity controlled
  - Reactivity not controlled
- Inherent reactivity feedback to maintain the core reactivity within a safe band. As noted above, micro-reactors will likely possess inherent negative reactivity feedback mechanisms that prevent the reactor from reaching unsafe power levels. While such inherent negative reactivity feedbacks are generally highly reliable based on past experience with nuclear

reactor technology, two possible outcomes are considered for this top event as part of formulating a generic micro-reactor APET:

  - Reactivity controlled
  - Reactivity not controlled

- Core heat removal could be provided by engineered means, depending on the specific scenario and the required amount of heat removal. For example, should an upset event occur in which the reactor power remains elevated, then the primary heat removal mechanism may rely on engineered secondary systems (i.e., the required amount of heat removal may be too large to achieve through passive convection or radiative heat dissipation alone). The potential presence of this type of safety measure is assumed in the development of this generic micro-reactor APET. However, it is not credited and all event scenarios assume that only primary and secondary passive heat removal measures are potentially available.

- Core heat removal via a primary passive heat removal measure. The micro-reactor design concepts discussed above incorporate a PHX as part of the power conversion function. This method of core heat removal is assumed to passively reject heat to the power conversion block. Thus, passive functioning of this heat exchanger is assumed to accommodate heat removal at operating power levels. Failure of this PHX (e.g., due to structural damage promoting sCO_2 leaks) results in a loss of primary heat removal. The function of this PHX is assumed to realize two modes, although degraded modes of operation would be realistic:

  - Core temperature controlled
  - Core temperature not controlled

- Core heat removal via a secondary passive heat removal measure. This is similar to the primary heat removal measure considered above. However, it is only capable of removing heat from the core at decay levels. The function of this secondary heat exchanger is assumed to realize two modes, although degraded modes of operation would be realistic:

  - Core temperature controlled
  - Core temperature not controlled

- The fuel structure integrity may become challenged as a result of either reactivity insertion or uncontrolled temperature increase. In this simplified evaluation, the fuel structure is assumed to be either perfectly intact or failed. In reality, depending on the nature of the reactivity and/or temperature excursions, the fuel structure may experience a spectrum of fuel failures.

  - Structural integrity maintained
  - Structural integrity lost

- The core structure integrity (in particular the heat pipes) may become challenged as a result of either reactor insertion or uncontrolled temperature increase. The core structure is assumed to be either perfectly intact or failed, although a graded failure spectrum would be more realistic.

  - Structural integrity maintained

- Structural integrity lost

Figure 4-1 presents a generic APET to predict how a micro-reactor responds to the different modes of operation of the different safety features noted above. A fundamental challenge to evaluating the likelihood of a range of event scenarios that could occur for a micro-reactor 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 a micro-reactor.

In addition to the evaluation of "passive reliability", the generic event tree presented below Figure 4-1 expresses the operation of passive safety features in terms of binary states. This captures successful and unsuccessful functioning of the passive safety feature. In reality, the performance of passive measures would experience a graded set of degradation, ranging from intended function to complete loss of function. A more sophisticated approach with multiple outcomes could be used to characterize ranges of responses. For example, in the case of a feature providing heat removal, degraded operation would correspond to the reduction of magnitude of heat removal. The degradation performance could include failures that increase the resistance to heat transfer from the core to the heat sink. In the case of structures, a graded set of failures would correspond to, for example, increasing failure areas that allow for transport of fission products out of the fuel and the core monolith/assembly.

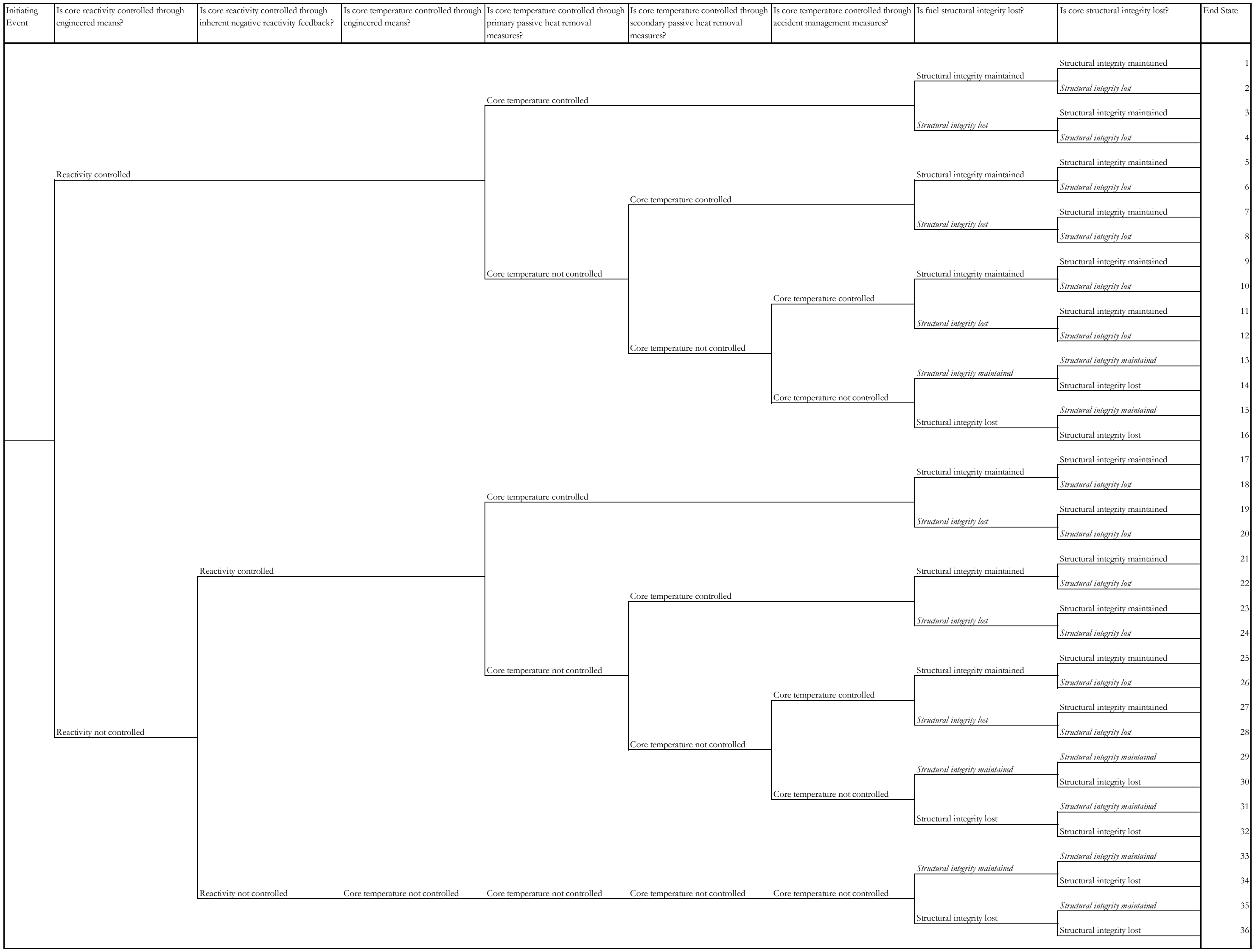


Figure 4-1. Structure of Generic Micro-Reactor Event Tree

4.2. Inherent Safety

Typically, internal event PRAs consider the failure of SSCs due to random processes with consideration of some common mode contribution across similar SSCs. In the case of a micro-reactor where inherent safety measures are part of the design, the random failure of a safety function is considerably less likely or plausible compared with traditional power reactors. For such inherent features, to achieve a safety function, an SSC is not required to change state; the safety function is achieved through natural processes governed by the physical laws. Thus, it is much more difficult (although not impossible) to postulate modes or conditions whereby such a safety function will not perform as intended.

With the development of Generation III+ reactor technology, such as the NuScale, AP1000, or the Economic Simplified Boiling Water Reactor, work has been performed to develop methodologies that attempt to assess the "reliability" of such inherent safety functions. As an example of a recent submission to the NRC, Chapter 19 of the NuScale FSAR [54] introduces some consideration of reliability for passive safety systems. It utilizes guidance provided by a range of past studies, such as Electric Power Research Institute (EPRI) [55], [56] and IAEA [57]. As noted in the FSAR, this involves consideration of the various factors that could lead to degradation of a passive safety function.

For example, the NuScale FSAR considers the functioning of the DHRS and ECCS, both of which rely on natural circulation to remove heat from the reactor core. Since natural circulation processes are typically established through small density gradients, they can be sensitive to perturbations that reduce these density gradients. The NuScale FSAR followed an approach to evaluate potential degradation of the density gradients to interrupt natural circulation as a result of thermal-hydraulic uncertainties. The evaluation of the thermal-hydraulic uncertainties in developing these passive safety systems led to the conclusion that DHRS passive function could fail with a probability of 4×10^{-6}, while ECCS passive safety function could be lost with a probability of 1×10^{-7}. These results indicate the high reliability that can be achieved through passive safety functions; that is, passive safety functions are robust against loss due to thermal-hydraulic failure with consideration of uncertainties.

Not explicitly considered in past studies, but of importance to designs largely based on inherent safety, is the role of other types of initiating event classes on the inherent safety function. For example, consider advanced High-Temperature Gas-Cooled Reactors (HTGRs) under loss-of-flow conditions. This type of design typically relies on natural circulation and radiation to transport decay heat away from the core and limit the extent of fuel heat up. Assisting this transport of energy out of the reactor vessel and into the reactor enclosure are a series of fins, enhancing the dissipation of energy into the reactor enclosure. Events, such as seismic events exceeding the design basis (as defined in risk-informed application framework of the LMP [9]), pose the potential to damage fins attached to the reactor vessel; they may either be dislodged or the severe motion could impair the contact between fins and the reactor vessel wall. The effect of this would be to introduce additional heat transfer resistance from the reactor vessel to the reactor enclosure. This degradation of heat dissipation from the reactor core into the reactor enclosure could be sufficient to induce more severe reactor core heat up. Due to the sensitivity of HTGR fuel structure integrity to the extent of reactor core heat up, this postulated hazard from a seismic event could lead to more severe radiological consequences (and risk) relative to that identified for a loss-of-forced-flow condition, arising from SSC failure characterized as part of an internal events PRA.

In this sense, the consideration of passive safety functions across a broader range of initiating events is critical to understand how the risk profile for non-LWR concepts changes. The mode by which a passive safety function fails may in fact be fundamentally different from random failure of the physical process when considering different types of perturbations that arise due to external events. In the case of the above HTGR example, random impairment of reactor core heat removal may be strongly influenced by uncertainty in the effective heat transfer resistance from the reactor vessel to the reactor enclosure. In the case of an external event (i.e., a severe seismic event), dislocation of heat shields on the reactor may systematically increase the heat transfer resistance in a manner that results in a high probability for natural circulation driven heat removal becoming insufficient to prevent core heat up and a challenge to fuel integrity.

4.2.1. Limits to Inherent Safety for Heat Pipe Reactors

The following discussion illustrates the ways in which, despite the significant robustness of heat transfer and associated inherent safety, heat transfer mechanisms can either break down or experience limits to their effectiveness. The heat pipe is used to provide a concrete example for discussion owing to its frequent consideration as a means of establishing transport of fission and decay energy in micro-reactors. The evaluation of such limitations of inherent functions is a foundational component of beginning to assess the limitations of inherent safety, identifying modes in which inherent safety functions could be degraded or cease to perform their intended function.

HPRs are intended to provide a means by which heat transfer from nuclear fuel is achieved in an efficient and reliable manner. In this sense, a HPR achieves a level of inherent safety through the exploitation of natural heat transfer processes. By doing so, it does not rely on active components to achieve the movement of energy away from its point of generation to a final heat sink.

Heat transfer processes, however, exhibit a number of limitations to the amount of heat that can be transported from source to sink. As an example, consider the boiling crisis encountered in water-moderated reactors. The boiling of liquid water (either light or heavy water) provides significant heat removal due to the large amount of latent heat required to produce the phase transition from liquid to vapor. This enables very large heat fluxes from nuclear fuel into the working fluid to be removed efficiently through boiling. However, in excess of a critical heat flux, a boiling crisis ensues; the heated surface passes into a dryout regime. In a post-critical heat flux, dryout regime, a vapor film forms on the surface, serving to significantly increase the resistance of the heat transfer into the liquid.

In the case of a HPR, the following are fundamental heat transfer limitations that occur as a function of the heat pipe design and heat transfer load. An illustration of these heat transfer limitations is provided in Figure 4-2. Depending on how conditions have evolved to achieve a particular operating temperature, the amount of heat transfer through a heat pipe could be limited by one of the following mechanisms. Despite the passive nature of heat transfer in a heat pipe, these physical limits must be considered when evaluating the overall effectiveness or magnitude of heat transfer [58].

- Continuum flow limit

  Vapor flow in the free molecular or rarefied regime can occur in heat pipes that are very small or operated at low temperatures. Heat transport in this regime is limited until a continuum vapor state can be reached.

- Frozen startup limit

  Working fluid may be depleted due to refreezing in the adiabatic or condensation zones. As a result, vapor generated in the evaporation zone is depleted, and this may progress to the point that the evaporation zone experiences dry out.

- Viscous limit

  The vapor pressure in the condenser may be reduced to zero under conditions where viscous forces dominate the flow of vapor in the heat pipe. This typically occurs at low temperatures (i.e., outside the normal operating range of the heat pipe). This condition is relevant to heat pipes operating with liquid metal working fluids.

- Sonic limit

  Heat pipes can experience vapor velocities that are either sub-sonic or sonic. The sonic limit can occur during startup or steady state operation and is applicable to heat pipes with liquid metal working fluids. Heat transfer is limited as long as vapor flow remains choked.

- Entrainment limit

  At high vapor velocities, shear forces at the vapor-liquid interface can entrain liquid from the wick surface into the vapor stream. In reducing the condensate return to the evaporator section, this process limits the extent of heat removal possible.

- Capillary limit

  The capillary wick structure has an inherent limitation to support circulation. Heat transfer is limited in this regime by the extent to which circulation can occur for a given capillary flow structure.

- Condenser limit

  Ultimately, as the operating temperature increases, a heat pipe may also experience limitations on heat transfer due to the ability of the condenser section to be cooled. There are a number of factors that can influence the condensation rate, including the presence of non-condensable gases that serve as a resistance to mass transport to the condenser surface.

- Boiling limit

  When radial heat fluxes or the heat pipe wall temperature become too high, the liquid returned in the wick may experience boiling. The reduction in condensate return under this situation serves as a limitation to heat transfer.

A critical factor influencing various heat transfer limitations described above is the ability of condensate to return to the evaporator region through the capillary wick structure.

In addition to the limits imposed under steady state conditions, the time-scale over which a heat pipe can respond to sudden heat load changes can potentially lead to failures.

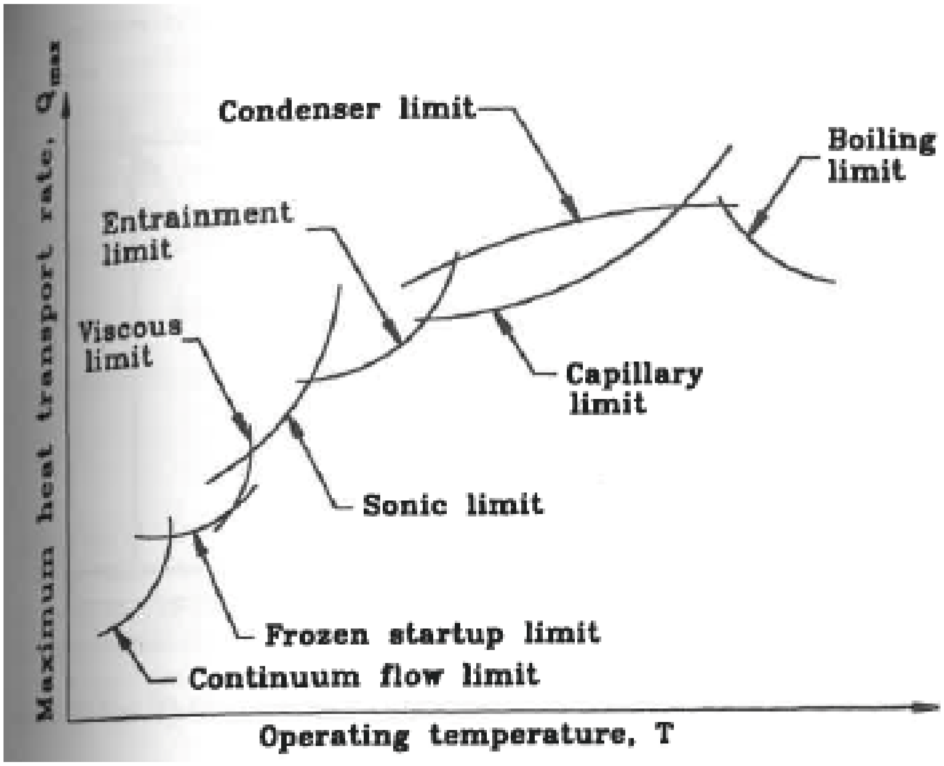


Figure 4-2. Illustration of Heat Pipe Heat Transfer Limitations as a Function of Operating Temperature [58]

4.2.2. Generalized Framework for Evaluating Inherent Safety

4.2.2.1. Identification of Plant Evolution Scenarios

Evaluation of passive safety system reliability has been introduced above in the context of the NuScale approach to quantifying the reliability of the ECCS and DHRS. The NuScale application focused on the reliability of passive systems providing core cooling. Inherent system safety for this application was dominated by thermal-hydraulic phenomena. An overall methodology for evaluating passive system reliability, however, is not specific to thermal-hydraulic phenomena. As noted above, there are inherent processes relevant to micro-reactor reactivity control that involve a coupling of chemical, thermal-hydraulic, and neutronic phenomena. A generalized method for evaluating passive system behavior under a range of uncertainties should be advanced by a vendor in establishing the licensing basis. While a specific methodology is not necessary, it is useful to provide the characteristics that any approach should incorporate into a passive system reliability evaluation.

- Step 1: Identification of factors influencing initial and boundary conditions for passive system.
- Step 2: Characterization of variability attached to initial and boundary conditions (i.e., technical basis for uncertainty distributions).
    - This type of variability is termed aleatory because it reflects uncertainty that can never be eliminated with more knowledge.

- Step 3: Identification of relevant physical phenomena, processes and parameters affecting the behavior of a passive system.

- Step 4: Characterization of variability in physical modeling parameters used in the quantitative evaluation of passive system response

    - This type of variability is termed epistemic because it reflects uncertainty in the modeling of physical processes arising from limitations in knowledge.
    - With enhanced knowledge these uncertainties could be reduced.

These components of methods to evaluate passive system reliability are focused on establishing the range of factors, and associated uncertainties, that affect prediction of passive system performance. This type of assessment is necessary to establish all the possible realizations of system response that could occur in light of aleatory and epistemic uncertainty. The passive system reliability is then derived as the fraction of realizations in which the system response maintains the plant within safety limits.

The approach and necessity for providing technical bases can also be illustrated through a schematic demonstration of the quantitative evaluation process. This starts through the development of a set of dynamical equations that can describe how physical and chemical processes govern the temporal evolution of the state of the plant. Assume that a system evolves as a result of time t in response to some external perturbation $\boldsymbol{f}_{ext}(t)$ from an initial state $\boldsymbol{s}_p(t_0)$ at time t_0. The system evolves according to a set of dynamical processes from the initial state $\boldsymbol{s}_p(t_0)$ to a future state $\boldsymbol{s}_p(t)$ that can be represented by an evolution operation $\mathcal{S}_p$ for the passive system,

$$\boldsymbol{s}_p(t) \to \boldsymbol{s}_p(t_0) = \mathcal{S}_p[\boldsymbol{m}(t-t_0)]\left(\boldsymbol{s}_p(t)\middle|\boldsymbol{s}_p(t_0); \boldsymbol{f}_{ext}(t-t_0), \boldsymbol{b}_{int}(t-t_0)\right) \tag{4-1}$$

The dynamical evolution of a system expressed by the operation $\mathcal{S}_p$ could, for example, be governed by a set of partial differential equations that represent how the state of a system evolves. In the case of single-phase fluid, the state of the system is normally expressed in terms of the conserved quantities of mass, momentum and energy of the fluid. These evolve according to the Navier-Stokes equations, and the solution of these partial differential equations would express the evolution operation $\mathcal{S}_p$. The model for dynamical evolution of the system can be a much more general and complicated set of coupled differential-algebraic equations, as is the case with many systems-level codes that simulate nuclear power plant response under severe accident conditions. The expression of the evolution operation $\mathcal{S}_p$ in these cases is much more complicated, involving the numerical solution of many coupled equations in a computer code with many hundreds of thousands of lines of executable statements. The NRC MELCOR computer code is an example of a detailed systems-level numerical solution method that expresses the evolution operation $\mathcal{S}_p$.

In this discussion, the details of how any specific dynamical system is modeled and its future state is predicted and abstracted into the operation $\mathcal{S}_p$. This abstraction is desired to illustrate the technical issues necessary to assess passive system reliability without explicitly advocating for any particular existing methodology.

The ways in which a system can evolve over an evolution time $t - t_0$ to a future state $\boldsymbol{s}_p(t)$ vary based on the,

- initial state of the system $\boldsymbol{s}_p^0 \equiv \boldsymbol{s}_p(t_0)$,
- external perturbations $\boldsymbol{f}_{ext}(t - t_0)$ to which it is subjected over the evolution time,
- interactions (or boundary conditions) $\boldsymbol{\ell}_{int}(t - t_0)$ with interfacing systems over the time of evolution, and
- parameters defining the model of system dynamical evolution $\boldsymbol{m}(t - t_0)$, where the time variability has been inserted to characterize the potential for the evolving state of the dynamical system to alter parameters (e.g., transition from sub-cooled boiling heat transfer to nucleate boiling heat transfer) or for new sets of physical processes to become relevant (e.g., the occurrence of molten core concrete interaction in LWRs when a severe accident progresses to ex-vessel damage).

Each of these factors influence the evolution of a passive system to some future state $\boldsymbol{s}_p(t)$ but may not be purely deterministic. That is, either they are inherently random (i.e., stochastic) and can only be known when realized; or they are incompletely known given the state-of-knowledge.

Thus, the ways in which a passive system can evolve to some future state, in response to an external perturbation, must be considered in a probabilistic manner. The evolution equation (4-1) provides a deterministic representation of a dynamical system evolution. To capture the ways in which the dynamical system could evolve given uncertainty/variability, it is necessary to extend this deterministic evolution equation to integrate variability and establish a probabilistic distribution of the future passive system state.

Assuming that each source of uncertainty/variability influencing the dynamical evolution of the passive system is independent, then it is possible to specify probability distributions for each source. For example, uncertainty in model parameters relevant to the dynamical evolution of the system would be characterized by a probability distribution $\mathcal{P}(\boldsymbol{\mathcal{M}}(t - t_0))$ over the random variable $\boldsymbol{\mathcal{M}}(t - t_0)$. Any given realization of the dynamical system evolution, in particular the future state of the system $\boldsymbol{s}_p(t)$, would reflect a sample or realization of a specific set of dynamical system model parameters $\boldsymbol{m}$. If the evolution of the system is only influenced by model parameter uncertainty, then any future state realization $[i]$ of the system $\boldsymbol{s}_p^{[i]}(t)$ would reflect a single sample from the distribution of model parameters; that is, $\boldsymbol{m}^{[i]}(t - t_0) \leftarrow \mathcal{P}(\boldsymbol{\mathcal{M}}(t - t_0))$.

NOTE: Not directly addressed in this discussion are sources of variability that enter due to the manner in which the predictive model for the dynamical system is numerically solved. For example, additional uncertainties can arise based on the discretization of the spatial or temporal domains. These sources of variability in predictive results are generally resolved through verification and validation exercises that support application guidance. A modeler normally applies computer code models with best practice guidance around, for example, appropriate levels of discretization refinement such that solution uncertainties are appropriately low and consistent with the level of uncertainty established through code validation.

While this mathematical formulation is presented in an abstract manner, it is intended to provide a generalized framework into which a range of analytical methodologies can be introduced. The overall output of this type of probabilistic methodology is to identify the range of plant evolution scenarios that can be realized. That is, the set of plant states that are possible given uncertainty in the perturbations to which a plant could be subjected, as well as how a plant may evolve in response to a given perturbation. Figure 4-3 provides a schematic illustration of the outcome of this type of probabilistic analysis, which provides a set of "paths" along which the plant state can evolve. These paths are depicted in Figure 4-3 using a "horse tail" plot, where each path is overlaid on top of each other to illustrate the range of temporal evolution of a particular parameter characterizing system response.

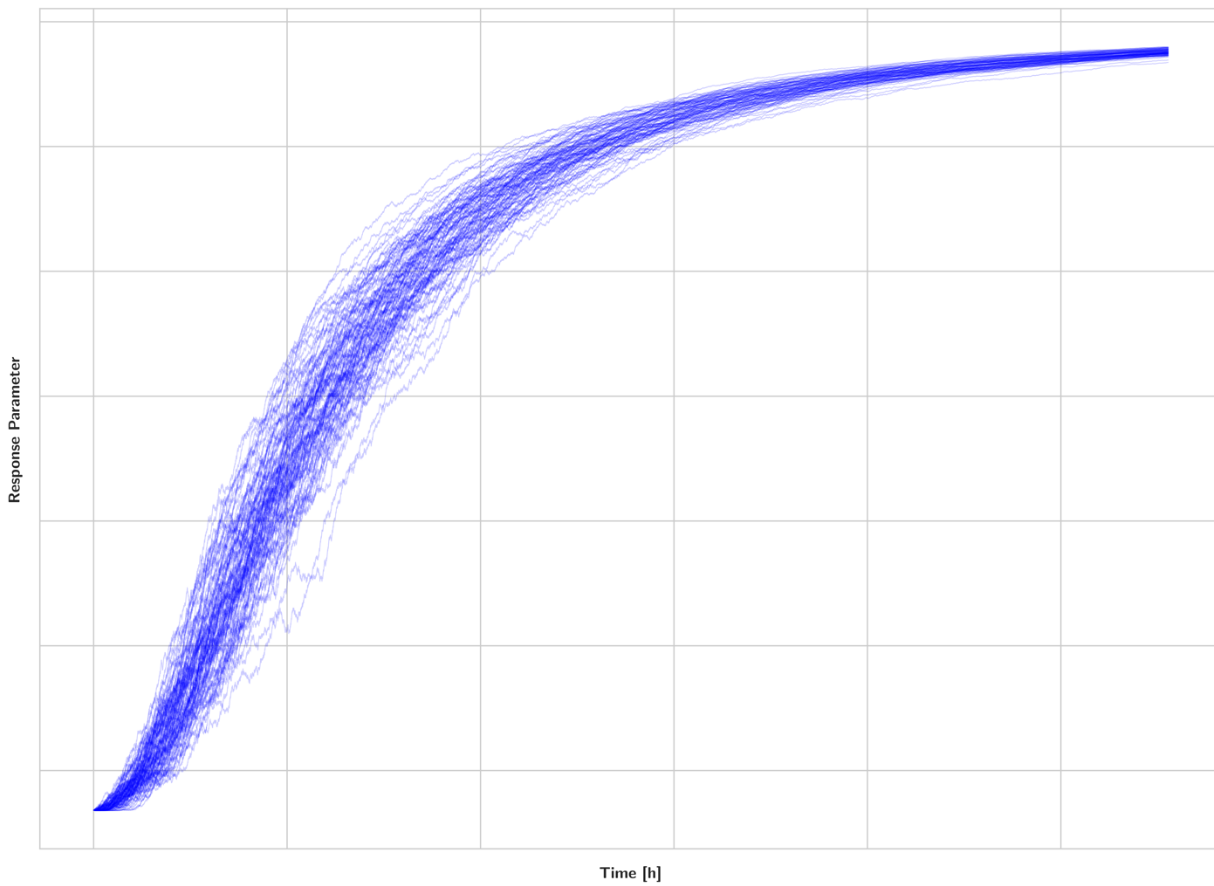


Figure 4-3. Illustration of Evolution Paths for Plant subjected to Initiating Perturbation

4.2.2.2. Incorporation of Passive SSC Reliability into Risk-Informed Decision-Making Framework

Figure 4-1 showed the overall structure of a generic APET for a micro-reactor. This event tree is not intended to be design-specific, and some considerations are introduced into the event tree that may not be present in some micro-reactor design. For example, the presence of engineered core heat removal measures may not be present in some of the proposed micro-reactor design concepts. Instead, the micro-reactor design concepts currently being advanced rely on a range of inherent safety measures.

Given the range of system evolution realizations that can occur in light of uncertainty influencing plant evolution, it is next necessary to identify specific safety metrics that enable characterization of the functioning of a credited safety measure. For example, the APET shown in Figure 4-1 credits a primary passive core heat removal function to control core temperatures. This could be provided by the PHX used in the power conversion function of a micro-reactor, as discussed above in Section 2. A range of safety metrics could be defined that allow characterization of the mode of operation of this primary heat core removal function. For example, the heat removal achieved by the system has excessive capacity to remove all heat generated within the reactor core. Alternatively, a derived metric such as the temperature in the core monolith remains below that at which thermally-induced stresses would exceed a failure limit.

Evaluation of the performance of the system requires correlating the range of system states realized to one or more metrics utilized for characterizing safety function performance. For example, assuming that core monolith temperature is a derived success metric for performance of the core heat removal function, the likelihood of function failure would be the probability that peak core temperature realizations exceeded the temperature at which monolith structure failure would be possible. Figure 4-4 shows example probability distributions for the peak core node temperature distribution (black line) and the distribution of monolith structural failure temperatures (red line). In Figure 4-4, it is assumed that a single failure temperature of 1000 K does not exist; rather, there is some probability that failure of the structure can occur at temperatures lower than 1000 K, and some probability that structure failure can occur at temperatures above 1000 K. The probability that structural failure occurs across the range of realizations is then the area under the core node temperature distributions, starting at temperatures for which the failure temperature distribution rises above zero. This total failure probability appears as the orange region in Figure 4-4 under the peak core node temperature distribution.

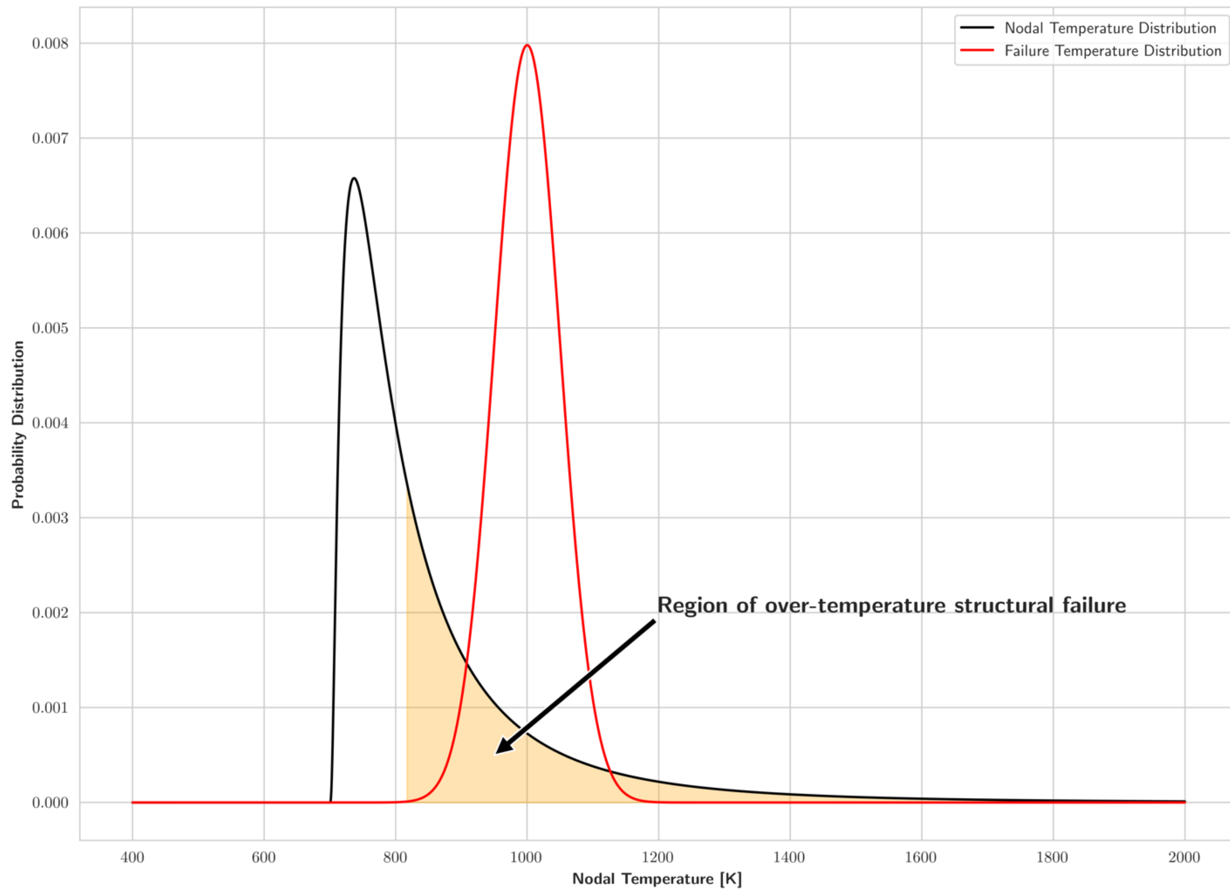


Figure 4-4. Evaluation of Probability of Exceeding Failure Threshold—Heat Pipe Monolith Over-Temperature Example

The probability calculated from the area under the curve given by the orange region in Figure 4-4, expresses the reliability of the core heat removal function in light of uncertainty. This probability $P_{hr,fail}$ can be used in an event tree like a normal probability for a top event branch. This is shown in Figure 4-5.

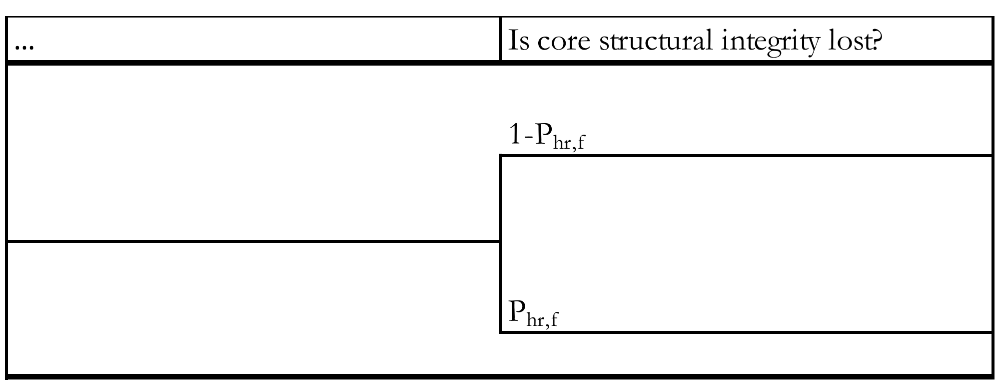


Figure 4-5. Illustrative Incorporation of Passive Reliability into Traditional PRA Event Tree

4.2.3. Example: Structural Failure under Thermal Loading Transient

The following discussion provides a perspective on evaluation of the response of the passive monolith structure subjected to (1) heating by internal heat generation (from reactor fuel), with (2) heat removal by an interface with a heat exchanger. The example is not intended to be a specific example of a hazard encountered in micro-reactors. Rather, it is intended to provide an example of the application of concepts necessary for evaluating the "reliability" of passive SSCs. The chosen example is simplified to illustrate the fundamental components of evaluating passive SSC reliability without extensive detail to interfere with the overall presentation.

The example below considers how the monolith structure could heat up under an imbalance between internal heat generation and rejection to a secondary-side heat sink. The driving parameter of influence in this example is the effective heat transfer to the secondary-side. The assessment of conditions that could cause this imbalance is an essential step in understanding types of hazards that could give rise to elevated risk for such a passive system as a micro-reactor. For example, are there seismic events that could damage either the primary or secondary inherent heat removal functions sufficiently to increase the effective resistance to heat transfer?

4.2.3.1. Evaluation of Passive SSC Reliability

The generalized characterization of inherent system safety in terms of passive system reliability provides a means to establish performance criteria. In addition, a structured decomposition of the factors influencing passive system operation are essential to establishing inspection or surveillance programs. As an example, assume that the peak temperature of the monolith structure, T_{peak}, can be correlated to the potential failure of one or more heat pipes based on the criterion $T_{peak} > T_{hp}^{fail}$. Upon failure, the potential for fission product release from the monolith and thus the inherent safety maintained due to passive fission product barriers would be lost.

A monolith structure temperature can be generally characterized using the traditional energy transport equation together with governing equations for the energy sources and sinks. For the solid heat pipe monolith temperature, the evolution of the structure temperature T_{hp} will be governed by the standard conduction equation,

$$\frac{\partial\left[\rho_{hp} h_{hp}(\boldsymbol{r},t)\right]}{\partial t} = \boldsymbol{\nabla} \cdot \left[k_{hp}\left(\boldsymbol{r}; T_{hp}\right)\boldsymbol{\nabla} T_{hp}(\boldsymbol{r},t)\right] \tag{4-2}$$

where:

h_{hp} is the specific enthalpy of the heat pipe monolith structure,

ρ_{hp} is the density of the heat pipe monolith structure, and

k_{hp} is the heat pipe monolith structure thermal conductivity that is assumed to vary with position as well as temperature.

The governing conduction equation satisfies a set of boundary conditions that reflect the heat added or removed from the structure due to

- radioactive decay and nuclear fission,
- evaporation of the heat pipe working fluid, and
- condensation of the heat pipe working fluid in the condenser section.

Equation (4-2) does not explicitly model the temperature of the fuel T_f within the heat pipe monolith. This can be represented by a similar energy transport equation that includes internal heat generation from radioactive decay and nuclear fission

$$\frac{\partial\left[\rho_f h_f(\boldsymbol{r},t)\right]}{\partial t} = \boldsymbol{\nabla}\cdot\left[k_f\left(\boldsymbol{r};T_f\right)\boldsymbol{\nabla}T_f(\boldsymbol{r},t)\right] + Q_n'''(\boldsymbol{r},t) \tag{4-3}$$

where:

h_f is the specific enthalpy of the fuel in the heat pipe monolith structure,

ρ_f is the density of the fuel in the heat pipe monolith structure

k_f is the thermal conductivity of the fuel in the heat pipe monolith structure, which is assumed to vary with position as well as fuel temperature, and

Q_n''' is the volumetric source of nuclear heat from radioactive decay and fission power.

At the interface between fuel and the heat pipe structure, there will be energy exchange due to a temperature gradient between the fuel material and the heat pipe structure. A heat transfer coefficient h_{f-hp} characterizes the energy transfer from the fuel to a heat pipe. The rate of energy transfer per unit area between the fuel and heat pipe can then be expressed as,

$$Q_{f-hp}''(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{f-hp}} = h_{f-hp}\left[T_f(\boldsymbol{r},t) - T_{hp}(\boldsymbol{r},t)\right]_{\boldsymbol{r}\in B_{f-hp}} \tag{4-4}$$

where:

B_{f-hp} specifies the boundary between the fuel and the heat pipes in the monolith.

The evaporation of heat pipe working fluid in the evaporator section results in an energy transport out of the heat pipe structure. This can be represented by an energy flow per unit area

$$Q_{hp-e}''(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{hp-e}} = h_{hp-e}\left[T_{hp}(\boldsymbol{r},t) - T_e(\boldsymbol{r},t)\right]_{\boldsymbol{r}\in B_{hp-e}} \tag{4-5}$$

where B_{hp-e} specifies the boundary between the heat pipes in the monolith and the evaporator-region of the heat pipes, and the effective heat transfer coefficient between heat pipe wall and evaporator-region fluid being given by h_{hp-e}.

The condensation of heat pipe working fluid in the condenser section can be expressed in terms of a heat removal rate per unit area

$$Q''_{c-hp}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{c-hp}} = h_{c-hp}\big[T_c(\boldsymbol{r},t) - T_{hp}(\boldsymbol{r},t)\big]_{\boldsymbol{r}\in B_{c-hp}} \tag{4-6}$$

with the effective heat transfer coefficient between condenser-region fluid and the heat pipe wall being given by h_{c-hp}, and B_{c-hp} specifying the boundary between the condenser-region of the heat pipes and the heat pipes in the monolith.

Finally, the heat pipe temperature in the condenser region will be governed by the heat removal rate on the secondary side. In a simplified systems model, this can be expressed as a boundary condition on the primary side through a fixed heat removal rate per unit area, Q''_{hp-ss}.

Not explicitly considered in the above interfaces is the energy transport that occurs from the heat pipe monolith to the medium in which it resides. For example, HPRs could be buried beneath grade. There will be some energy transport, though small, that occurs between the heat pipe monolith and the soil in which the HPR resides. While there is likely sufficient insulation to ensure that this heat loss is limited, it should be considered. For illustration purposes, this energy transport can be represented by a similar heat transport rate equation to those considered above

$$Q''_{hp-env}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{hp-env}} = h_{hp-env}\big[T_{hp}(\boldsymbol{r},t) - T_{env}\big]_{\boldsymbol{r}\in B_{hp-env}} \tag{4-7}$$

where:

T_{env} is a temperature for the interfacing environmental medium (e.g., soil) and is assumed to be a bulk constant for simplicity

h_{hp-env} expresses the heat transfer conductance to the environment from the heat pipe monolith

B_{hp-env} specifies the boundary between the heat pipe monolith and the environmental medium

With these various rates of energy transport defined, it is possible to specify the boundary conditions for Eq. (4-2) and Eq. (4-3). The fuel temperature must satisfy the following boundary condition,

$$k_f(\boldsymbol{r};T_f)\nabla T_f(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{f-hp}} = Q''_{f-hp}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{f-hp}} \tag{4-8}$$

The temperature of the heat pipe structure itself must satisfy boundary conditions at the interfaces between the fuel and the heat pipe, the heat pipe and the fluid in the heat pipe evaporator-region, and the fluid in the heat pipe condenser-region and the heat pipe. The boundary condition at the interface between the heat pipe and fuel is given by Eq. (4-8). The additional interfaces are given by the additional boundary conditions,

$$k_{hp}(\boldsymbol{r};T_{hp})\nabla T_{hp}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{hp-e}} = Q''_{hp-e}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{hp-e}} \tag{4-9}$$

$$k_{hp}(\boldsymbol{r};T_{hp})\nabla T_{hp}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{c-hp}} = Q''_{c-hp}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{c-hp}} \tag{4-10}$$

$$k_{hp}(\boldsymbol{r};T_{hp})\nabla T_{hp}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{hp-ss}} = Q''_{hp-ss}(\boldsymbol{r},t)\big|_{\boldsymbol{r}\in B_{hp-ss}} \tag{4-11}$$

The above model formulation for establishes key sources of variability that must be considered when addressing the potential for monolith structure temperatures to reach levels that could induce failure.

External perturbations drive a range of potential transients of relevance to safety, as represented by two key parameters,

- The nuclear heat generation rate Q_n''' and
- The rate of heat removal by the secondary side Q''_{hp-ss}

While the secondary-side heat removal also serves the role of a boundary condition, in this situation it is useful to classify it as a potential external perturbation to the system. These parameters belong to the class of external perturbations to the system that are generally introduced as $\boldsymbol{f}_{ext}$. Because these parameters can be realized in different ways across a class of accidents (e.g., due to a reactor power excursion, secondary-side under-cooling, or secondary-side over-cooling), they should be treated as random variables through the process of assessing the robustness of the passive heat pipe monolith structure.

- Boundary conditions specifying the internal interactions between components of the system,
  - Heat transfer between fuel and heat pipe structure Q''_{f-hp}, and in this formulation the relevant source of variability enters through the heat transfer coefficient h_{f-hp},
  - Heat transfer between the heat pipe structure and fluid in the evaporator-region of the heat pipe Q''_{hp-e}, which in this model formulation acquires variability through the heat transfer coefficient h_{hp-e}, and
  - Heat transfer between the heat pipe structure and fluid in the condenser-region of the heat pipe Q''_{c-hp}, which in this model formulation acquires variability through the heat transfer coefficient h_{c-hp}
- Boundary conditions specifying the external interactions of system components with interfacing, but not directly modeled,
  - Heat transfer between the heat pipe monolith and the medium in which it resides is given by the heat transfer rate Q''_{hp-env}, which in this model formulation acquires variability through the heat transfer coefficient h_{hp-env}.
- Model parameters that influence the behavior of the equations governing dynamical system evolution, specifically the energy transport equations for the fuel and heat pipe monolith structure in Eq. (4-3) and Eq. (4-2), respectively,
  - The thermal conductivity of fuel k_f
  - The specific enthalpy of fuel h_f, which in a standard formulation would reduce to variability in the fuel density (ρ_f) and specific heat ($c_{p,f}$)
  - The thermal conductivity of the heat pipe monolith structure

- The specific enthalpy of the heat pipe monolith structure, which in a standard formulation would reduce to variability in the heat pipe structure density (ρ_{hp}) and specific heat ($c_{p,hp}$)

- Failure threshold for the heat pipe monolith structure,
  - Temperature is assumed in this example to be the key metric identifying a regime in which the passive structural integrity of the heat pipe has been challenged,
  - The heat pipe failure threshold temperature T_{hp}^{fail} could be considered to be a deterministic parameter; however, it most likely is a stochastic variable given the sources of uncertainty that govern how it is derived, or
  - Multiple methods could be used to establish this failure threshold, from testing to analytical modeling of structural response under imposed thermal loading conditions.

Assessing the reliability of the passive monolith structure, given a transient that induces a thermal loading of the structure, it is necessary to evaluate the likelihood that the peak structure temperature remains below a threshold temperature for heat pipe failure. The reliability of the passive structure for a thermal transient event equates to the probability that the peak structure temperature remains below the heat pipe failure threshold temperatures, across all realizations of how the monolith structure temperature could evolve, $\mathcal{P}(T_{peak} > T_{hpf})$. This probability can be computed based on the range of dynamical system evolution realizations giving the transient monolith structure temperature, $T_m(t_0) \to T_m(t)$.

$$\mathcal{P}(T_{peak} > T_{hpf}) = \int d\mathcal{T}_{hpf} \int d\boldsymbol{\mathcal{M}}\, d\boldsymbol{\mathcal{B}}_{int} d\boldsymbol{\mathcal{F}}_{ext} d\mathcal{T}^0 [\mathcal{P}(\boldsymbol{\mathcal{M}}, \boldsymbol{\mathcal{B}}_{int}, \boldsymbol{\mathcal{F}}_{ext}, \mathcal{T}^0) \times \times \theta(T_{peak} - T_{hpf})] \quad (4\text{-}12)$$

where $\theta(T_{peak} - T_{hpf})$ is the Heaviside Step function. The evaluation of the probability of the peak monolith structure temperature exceeding the failure temperature involves evaluating the overlap between the distribution of peak monolith structure temperatures and the distribution of failure temperatures. This process is visually represented in Figure 4-4.

NOTE: In this example, the failure threshold is considered to be a derived parameter most directly related to the temperature of the heat pipe monolith. Since an external perturbation is considered here, the heat pipe thermal profile is the most direct state variable affected by the accident transient.

The heat pipe failure threshold temperature T_{hp}^{fail} could have been established via separate structural analyses, or even through a multi-physics simulation coupling the transient driving the thermal response of the structure with the mechanical behavior of the structure under thermal stress.

Alternatively, the accident analyses could be developed in a manner where the heat pipe monolith structural temperature and structural response are calculated simultaneously as part of a multi-physics simulation. Identification of failure for any realization of a heat pipe thermal loading accident would then be based on, for example, stress distributions calculated as part of the structural response analysis. This approach may be valuable when the derivation of simple heat pipe failure temperature thresholds introduces excessive uncertainty or becomes too conservatively challenging.

The complexity of the analysis should not be pre-specified; however, it is necessary to ensure that the interaction of physical processes is not sufficiently strong to necessitate a coupled multi-physics analysis. In the context of this example, a multi-physics simulation would be required should the structural deformation due to imposed thermal stresses significantly alter the transport of energy within the monolith, and thus the calculated monolith temperature profile.

4.2.3.2. Illustration of Quantitative Evaluation

This sub-section provides an example of how an evaluation of passive safety reliability could be performed. This evaluation was implemented using a standalone set of computer calculations to illustrate the process of utilizing results of a physics-based uncertainty study to develop an estimate of event probability that could be incorporated in a traditional PRA approach.

The transient over-temperature can be realized in a number of different ways depending on the nature of the event scenario. For internal events, the presence of a DHRS will likely ensure that monolith temperature transients remain relatively well-controlled, remaining below a peak temperature that could induce failure. The radiative heat transfer away from the monolith through the DHRS should, by design, remain effective. Within an internal events PRA, this means that there are no plausible random failure modes of the DHRS that reduce heat transfer out of the monolith sufficiently to result in a challenging over-temperature.

To illustrate the characterization of how a temperature excursion could result following an upset in the primary heat removal function, the following example is provided. It is assumed that an initiating event results in loss of primary heat removal function and the reactor power is reduced. Transfer to the secondary heat removal function is achieved. Depending on the operation of the reactivity control drums, the reactor power may not be sufficiently reduced to ensure that secondary heat removal is sufficient. The heat removed by the secondary heat exchanger is characterized by an effective heat transfer coefficient. The extent of the heat up of the core node is then influenced by the long-term reactor power and the magnitude of heat transfer to the secondary heat exchanger.

Figure 4-6 presents a simulation of the fuel nodal temperature transient under conditions in which reactor power is reduced sufficiently to enable effective functioning of the secondary heat removal system. While there is an initial transient in the calculated fuel nodal temperature, the reduction in reactor power with time results in a gradual reduction of fuel temperature. The realizations with a temperature excursion exhibit a relatively modest increase in temperature. This excursion occurs due to an assumption that the effective heat removal is somewhat more degraded than nominal and thus is not sufficient to remove the initial decay heat levels. For the remainder of realizations, the fuel temperature exhibits a decrease to a steady-state representing the secondary-side temperature of the heat exchanger.

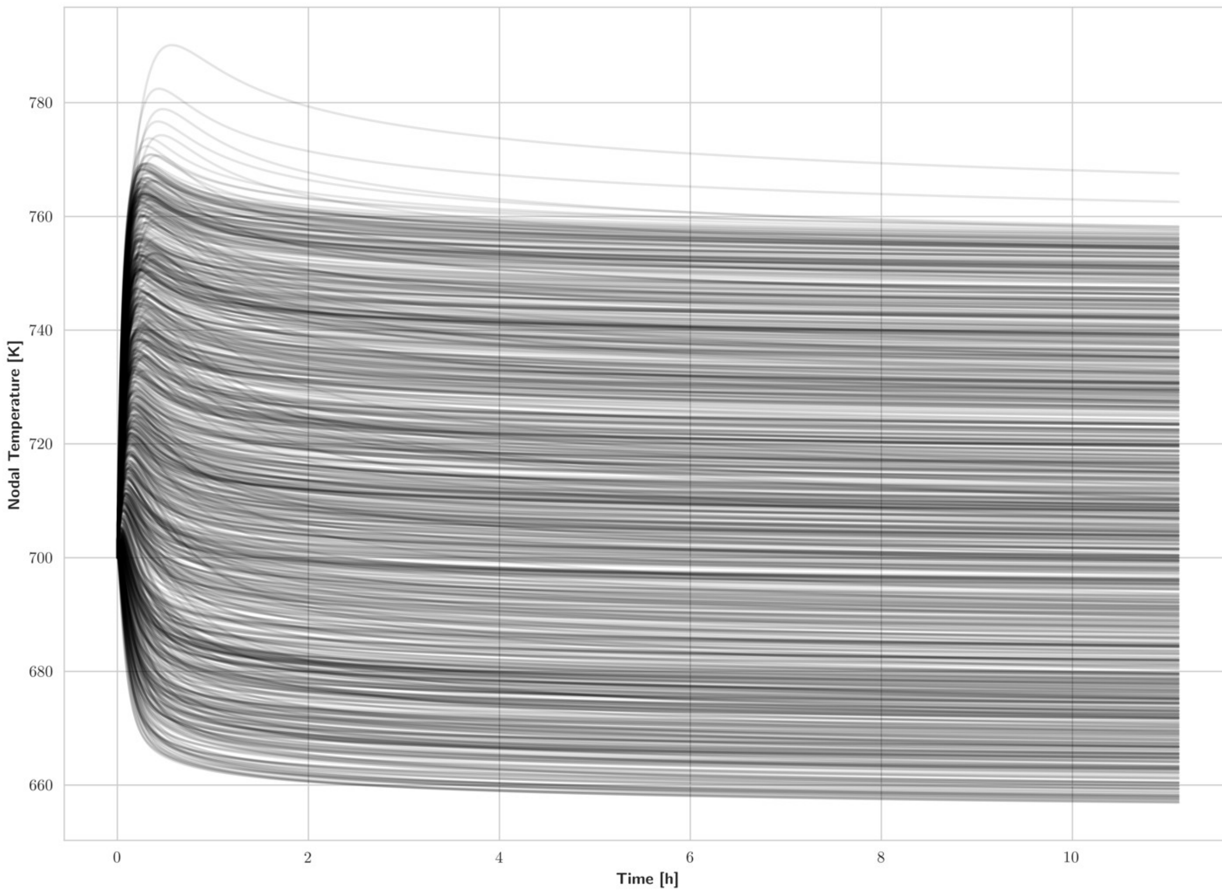


Figure 4-6. Simulation of Fuel Nodal Temperature Transient – Effective Heat Removal Function

More severe degradation of secondary heat removal function will result in a potential for unmitigated fuel temperature excursions. An *example illustration* of an outcome with relatively degraded secondary heat removal is provided in Figure 4-7. More severe temperature excursions are observed in this case, with the temperature transient generally not being controlled by means of heat removal. The long-term decrease in the fuel temperature occurs primarily due to the reduction in reactor power as a result of fission product decay. It is only in the long-term that the reduction in generated heat combined with the primary-to-secondary temperature gradient allows the fuel temperature to decrease.

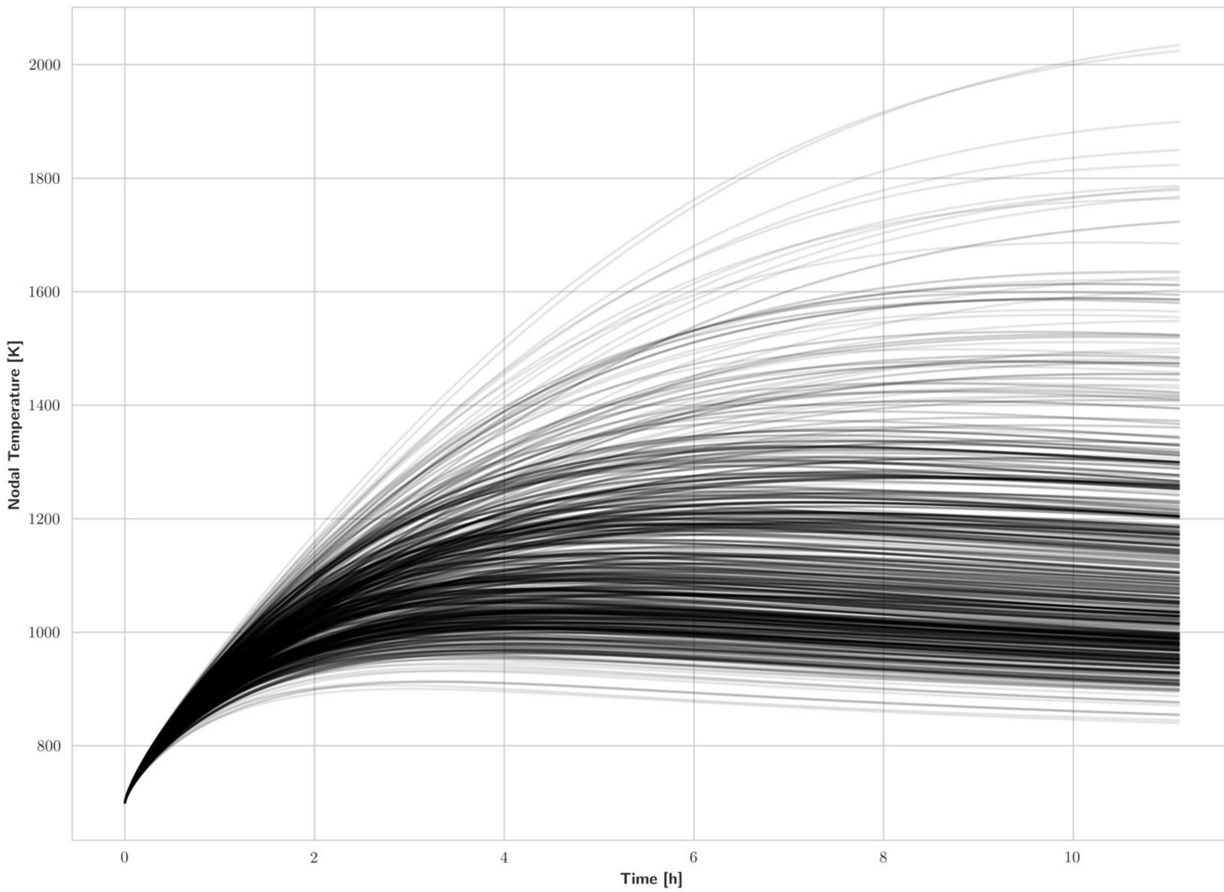


Figure 4-7. Simulation of Fuel Nodal Temperature Transient – Ineffective Heat Removal Function

A comparison of the peak fuel temperatures for these two illustrative scenarios is provided in Figure 4-8. The assumed distribution of structural failure temperatures is also shown overlaid on the two distributions of peak fuel node temperatures.

This is only intended to be an illustrative example of the concepts described above related to assessing inherent safety function reliability. The presence of distinctly different inherent safety performance is a critical theme to be expected for advanced reactor concepts like the micro-reactor. As noted above, the performance of advanced reactor concepts relying on inherent safety features is expected to be highly reliable. This ensures that the safe operation of the reactor can remain robust in face of traditional challenges such as random failure of SSCs typically evaluated by an internal events PRA. However, external events are expected to lead to conditions that could challenge the inherent safety functions. Any degradation of this inherent safety function will result in reactor conditions more severe than anticipated by simulation of plant response under a range of upset conditions considered by an internal events PRA. Thus, the range of plant response anticipated would be expected to exhibit the type of dramatic bifurcation observed in Figure 4-8 when considering initiating event perturbations traditionally considered as external events in LWR PRAs. This shift in the overall risk profile of a reactor incorporating significant levels of inherent safety should be expected; it reflects a key divergence from the traditional study of power reactor risk.

NOTE: Inherent or passive safety is typically assumed to be without failure mode. This is strictly not true under all circumstances. While random failures of a safety function are typically much less likely for passive systems, consideration of the potential for failure is necessary for many systems. Physical processes typically occur reliably, but the underlying conditions that support the physical processes may not always apply. For example, density gradients driving passive heat removal systems may not always be present if there is any perturbation in the extent of heat removal from the system.

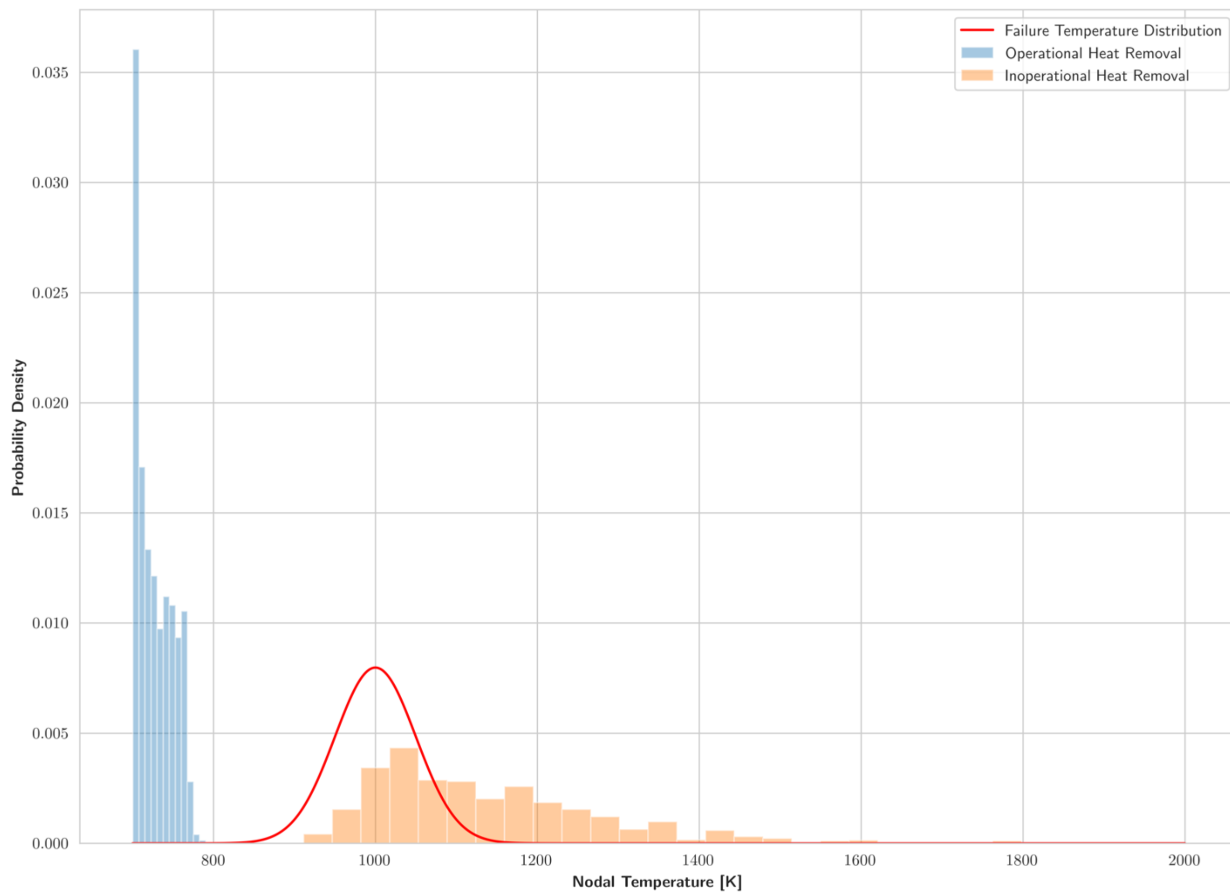


Figure 4-8. Distribution of Example Peak Core Over-Temperatures

4.2.3.3. Incorporation of Passive SSC Reliability into Risk-Informed Decision-Making Framework

As described above, the area giving the overlap between the distributions of realized peak and failure temperatures provides an estimate of the reliability of the inherent safety function. In Figure 4-8, peak temperatures realized for the case of an operational secondary heat removal function have no overlap with the failure temperature distribution. As such, a lower bound reliability for secondary heat removal function of about $1{\times}10^{-7}$ is a reasonable assumption. This represents a probability of failure conditional on the initiating event giving rise to the temperature excursion. A lower bound reliability of $1{\times}10^{-7}$ is used in this example, even though the numerical evaluation of the fraction of realizations with temperatures exceeding the failure temperature of the structure is zero. In performing such reliability evaluations, it is reasonable to use lower bound cutoff probabilities given that the operating experience is currently not extensive enough to justify assigning reliability values below a value of 1×10^{-7} to represent a very rare event. In the case of the challenged secondary heat removal function, the overlap between the two distributions gives a probability of failure of about 0.25 (i.e., the fraction of realizations with temperatures above the randomly realized failure temperatures is about 25%).

4.3. Incorporation of Operating Experience

A given reactor design concept may be informed by operating experience collected from experimental, commercial, or military programs. Operating experience can furnish information about:

- Start-up,
- Normal operation,
- Transients and accidents,
- Shut-down and decommissioning, and
- Reliability/failure and maintainability

An experimental program typically defines and follows a test matrix that can include but is not limited to:

- Basic physics experiments to gather data or demonstrate a concept,
- Separate effects tests focused on one (or at most a few) SSCs of a given reactor design, and
- Integral effects tests focused on the collective action of multiple SSCs of a given reactor design.

An experimental program will occasionally produce data with a full-scale test section. However, data is more often collected on a scaled test section that may include scaling distortions. Furthermore, experiments are carried out over a range of conditions and under certain limiting assumptions. Care should always be taken when extrapolating experimental results. Commercial and military programs are useful for gathering reliability, maintainability, and operating data from full-scale nuclear power systems across the range of operating regimes. Micro-reactors of either the monolith or assembly concept have design features to which past operating experience could be cautiously applied.

Operating experience interfaces with the LMP framework in several ways:

- Informs an initial LBE listing as a complement to engineering judgment,
- Furnishes hard quantitative data for use in PRA,
- Informs a listing of possible initiating events and transient/accident sequences,
- Provides certain quantitative information about safety functions performed by SSCs,
- Demonstrates the need for certain SSCs of reactor design, and
- Demonstrates certain concepts employed by SSCs of reactor design,
    - Operating principles
    - Passive/active intervention of SSCs
    - DID layer action and overall DID adequacy

Operating experience can therefore impact LBE assessment by quantitatively informing the PRA and SSC classification and by more completely informing the definition of LBE space (suggesting possible LBEs, initiating events, and transient/accident sequences).

An important technical issue related to operating experience as applied in the context of the LMP framework is the question of applicability. It may be useful to devise criteria of applicability for operating experience that could be incorporated into the LMP framework. The purpose of the applicability criteria would be to guard against misinforming the LMP process with data and/or observations drawn from inapplicable operating experience.

4.4. Testing Program

The NRC roadmap for licensing a new reactor design includes testing programs from the proof-of-concept through the combined operating licensing [59]. Testing occurs at the pre-acquisition, conceptual, design and construction, and acceptance phases of a new reactor life cycle project (see Figure 4-9). The culmination of the testing occurs with the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) phase to complete the licensing procedure[11] (i.e., see Figure 5-4 in Section 5). New micro-reactor designs are expected to have several design verification challenges for licensing. Testing and the associated component development are mechanisms to demonstrate satisfactory resolution of safety and design issues.

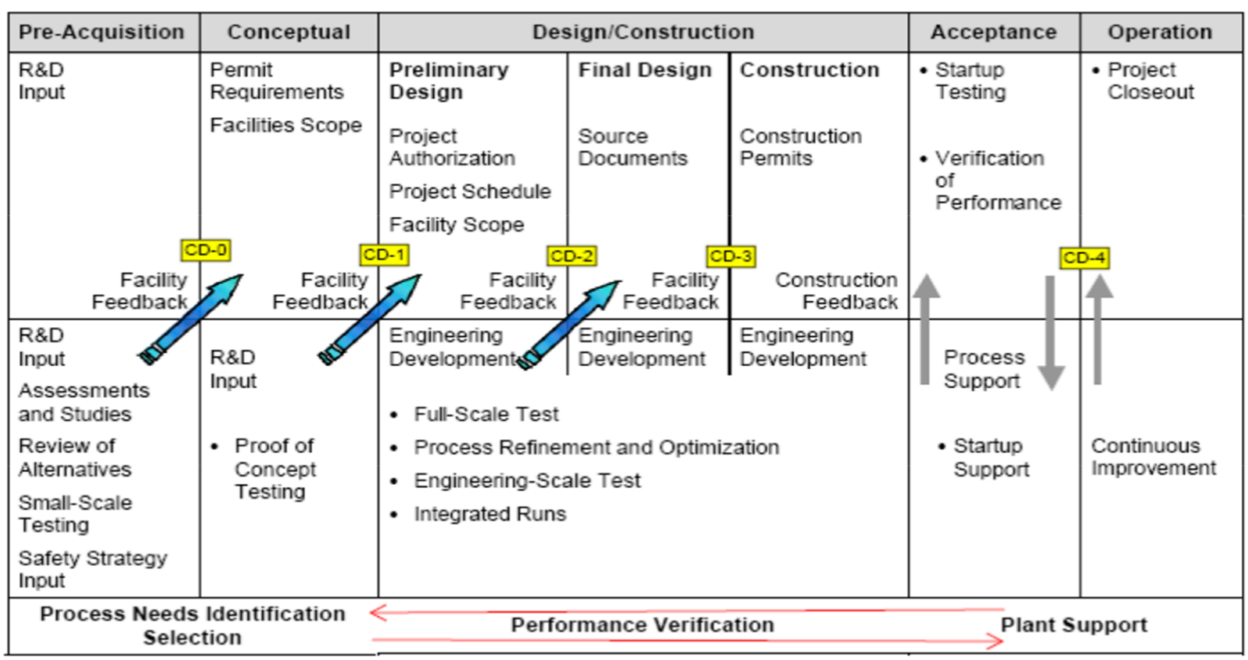


Figure 4-9. Role of testing in a new reactor lifecycle [59]

While not mandatory, the NRC roadmap to new reactor licensing has provisions for prototype reactor licensing leading to resolution of outstanding testing verifications before the final design certification (DC) submittal, which would be expected for any new micro-reactor design.

[11] The specific requirements are cited in 10 CFR 52.97 (b). ITAAC is used to finally resolve all testing acceptance criteria have been met.

Appendix C of the NRC roadmap to licensing explains the process to determine testing needs. The test facilities are a key component to justify performance and safety claims regarding a new micro-reactor design not previously licensed by the NRC. The types of test facilities may include individual components (e.g., the heat pipe), non-nuclear test facilities, and prototype facilities. NRC regulations (i.e., 10 CFR 52.2) includes special considerations for licensing prototype facilities. As an alternate to the prototype, the first-of-a-kind (FOAK) plant could include additional start-up tests to resolve final acceptance criteria.

The testing requirements largely arises from the NRC requirements for the safety evaluation of the new micro-reactor. The NRC advocates a systematic process where the performance bases for each SSC and safety function are defined. The design review includes rationale and justification whether analysis, existing data, or new data is needed to justify the performance requirements. The rigor of the performance evaluation is variable relative to the importance of the SSC or safety function and the associated uncertainties. Figure 4-10 shows a suggested flow chart for assessing testing requirements and the appropriate facilities for each requirement. When the operating history of a SSC or safety component in existing facilities cannot support cited claims of high reliability, then testing can be used to demonstrate the performance.

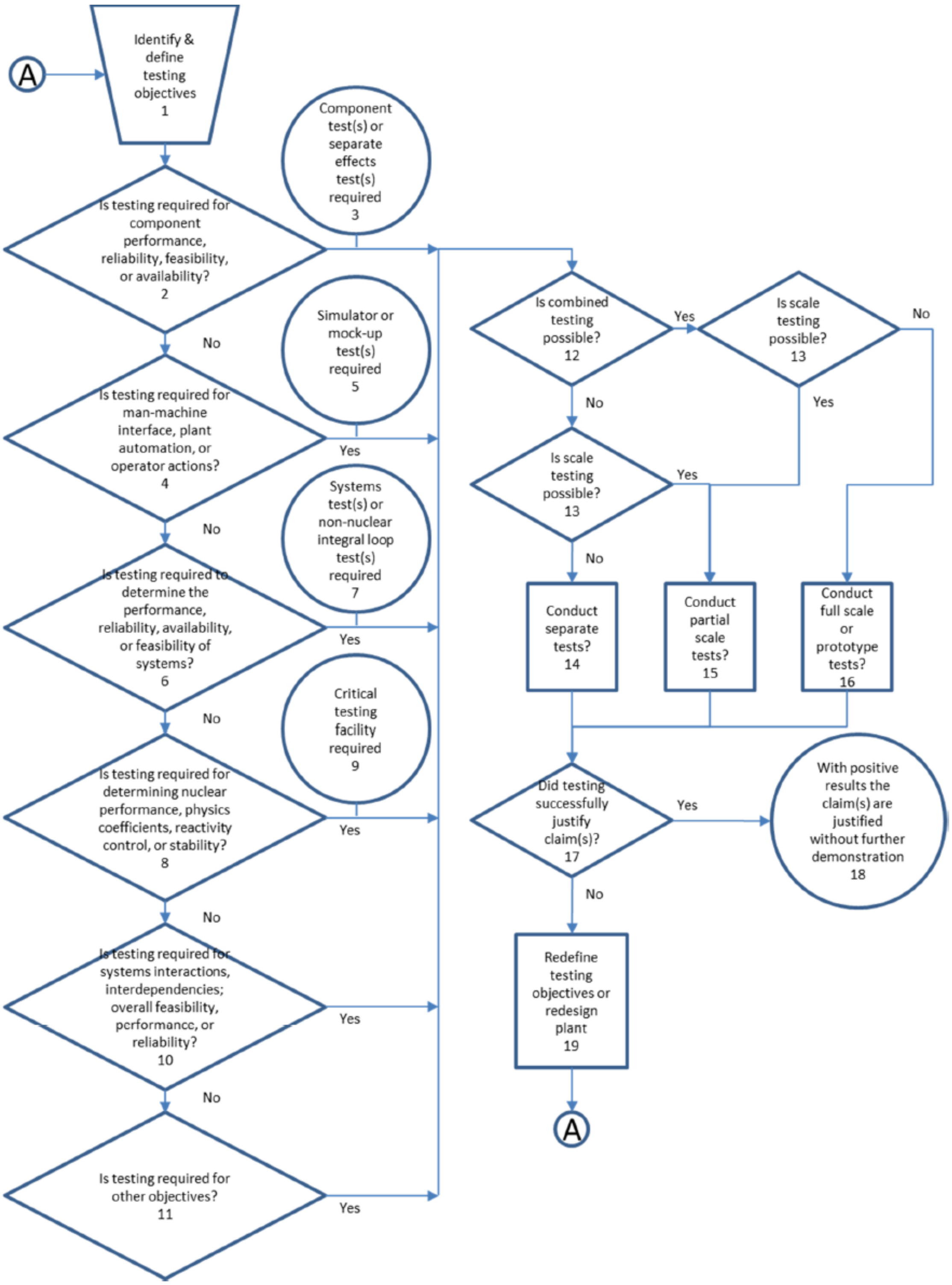


Figure 4-10. Process for evaluating testing news and outcomes [59]

A key outcome from testing is verification and validation of the design computer codes. Due to the lack of operating experience with micro-reactors, testing and the associated component development will be important to convey the proof of concept. As noted in the draft regulatory guidance for the MST for advanced reactors [43], "…sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic (source term) approach." Consequently, testing is expected to be important for micro-reactor design and licensing.

Assuming a heat pipe micro-reactor design, testing needs might include the following categories:

- Evaluation of heat pipe limits (HPLs) and safe operating regimes
- Demonstration of inherent safety
  - Reversible return to safe operation once an HPL is exceeded
  - Design DID following challenges to the HPL
  - Confirmation of no single failure or common mode failures in AOOs and DBAs
- Separate effect heat pipe performance versus integrated heat pipe performance
  - Heat removal characteristics across bundle of heat pipes
  - Incremental role of heat pipes or their failure on system response
- Neutronic characteristics
  - Role of sodium or other heat pipe working fluids on neutronic response[12]
  - Impact of the heat pipe failures
  - Anticipated transient without Scram and demonstration of (inherent) shutdown mechanisms
- Heat Pipe construction
  - Reliable construction of vacuum-filled heat pipe and quantification of impurities
  - Identification and control of working fluid impurities
- Corrosion and chemistry issues
  - Collection of working fluid impurities
  - Integral impact of heat pipe leaks
- Testing of functional containment defense in depth elements
  - Characterization of risk-significant challenges
  - LBE response
- Start-up and shutdown
  - Safe start-up
  - Long-term heat removal

[12] The working fluid in heat pipes is primarily voided with a small amount of liquid. The influence of the very small amount of the HP working fluid on the reactivity feedback in the extremes of most working fluid in the core versus nearly fully voided.

The LMP advocates risk-informed methods to guide testing priorities [9], which has draft acceptance by the NRC [43]. The critical testing can be an important element to directly reduce uncertainties in risk-significant responses or improve understanding where the testing reveals unexpected responses. Finally, the testing provides critical data for verification and validation of deterministic design codes.

5. COMPONENTS OF MICRO-REACTOR LICENSING SUBMITTAL

This section provides an evaluation of the components of a micro-reactor licensing submittal by comparison against the LWR SRP. It is important to note that the SRP is an LWR-specific document. However, because of the familiarity with the SRP, the elements of a micro-reactor licensing submittal are evaluated in the context of how each element of the SRP may need to be modified or may not apply to a micro-reactor. The actual organization of a licensing submittal may deviate considerably from the LWR SRP for non-LWRs. While the overarching safety objectives of the LWR SRP remain applicable to micro-reactors, the implementation of how these areas are addressed may differ based on the design.

The NRC new reactor licensing process is governed by Title 10 of the Code of Federal Regulations Part 50 or Part 52 (10 CFR Part 50 or 10 CFR Part 52). Some reactor vendors developing non-LWR concepts are evaluating licensing under 10 CFR Part 50. This section considers both 10 CFR Part 50 as well as 10 CFR Part 52.

10 CFR Part 50 (Domestic Licensing of Production and Utilization Facilities) consists of a two-step licensing process (NUREG/BR-0298, "Nuclear Power Plant Licensing Process" [60]):

- Construction Permit (CP)
- Operating License (OL)

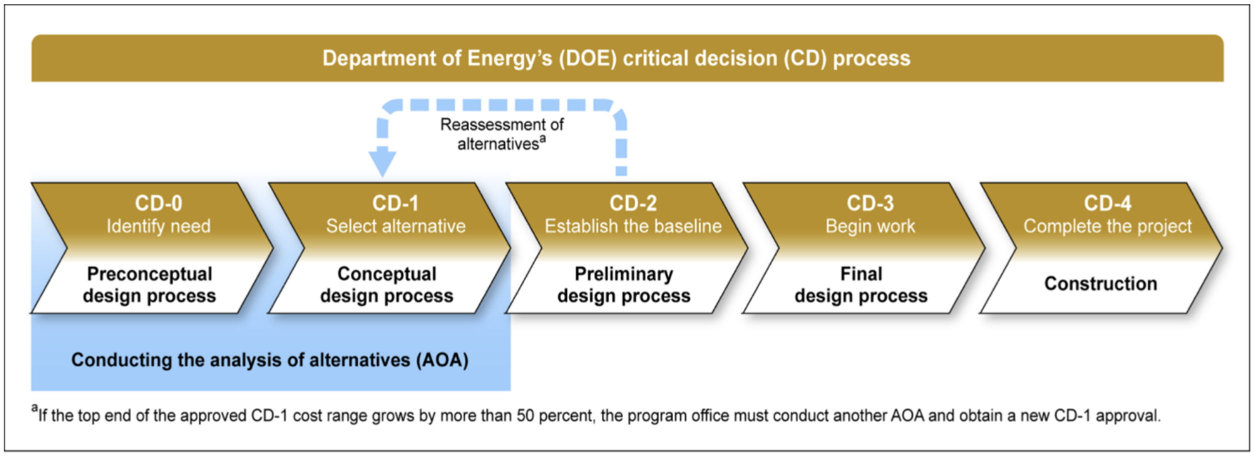
10 CFR Part 52 (Licenses, Certifications, and Approvals for Nuclear Power Plants) incorporates additional licensing processes (NUREG/BR-0298, "Nuclear Power Plant Licensing Process" [60]):

- Early Site Permit (ESP)
- Standard DC
- Combined License (COL)
- Additional Processes
    - Manufacturing License
    - Duplicate Plant License
    - Standard Design Approval (SDA)
    - Site Suitability Reviews

5.1. Roadmap to Licensing Advanced Reactors

The NRC has described possible roadmaps for licensing non-LWR designs [59]. The existing processes are flexible and support interactions for advanced design development and licensing. The structure of the engineering design process, and as a result the stages at which a vendor will engage the NRC, could be expected to follow the DOE critical design process shown in Figure 5-1. Figure 5-2 and Figure 5-3 show how technology development intersects with the design process and eventual entrance into operation. This is of particular relevance to a novel reactor concept.

Interactions between reactor designers and the NRC staff provide a mechanism to support regulatory decisions or exceptions related to a new design. The process includes exchanges with the NRC leading to conclusive NRC staff findings and a final NRC position. As an example, NuScale requested an exemption for Class 1E electrical systems, which was ultimately approved [61]. The pre-application process can include preliminary design submittals for feedback. The pre-application interactions are intended to ensure that the design control document (DCD) is prepared in accordance with NRC's regulations (i.e., an encouraged activity to reduce licensing uncertainty).



Source: GAO analysis of DOE's Order 413.3B. | GAO-15-37

Figure 5-1. Department of Energy Critical Decision Process [62]

Similar to new LWR designs (e.g., the AP1000), the micro-reactor DCD "must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design" [59]. As an alternative, the micro-reactor designer may submit a proposed final standard design. The NRC will document their conclusive findings in the SDA. However, unlike the DCD approval, the SDA does not prevent issues resolved in the SDA from being reconsidered in the DCD review. However, the SDA can be used to provide feedback on a portion of the final design to reduce regulatory uncertainties.

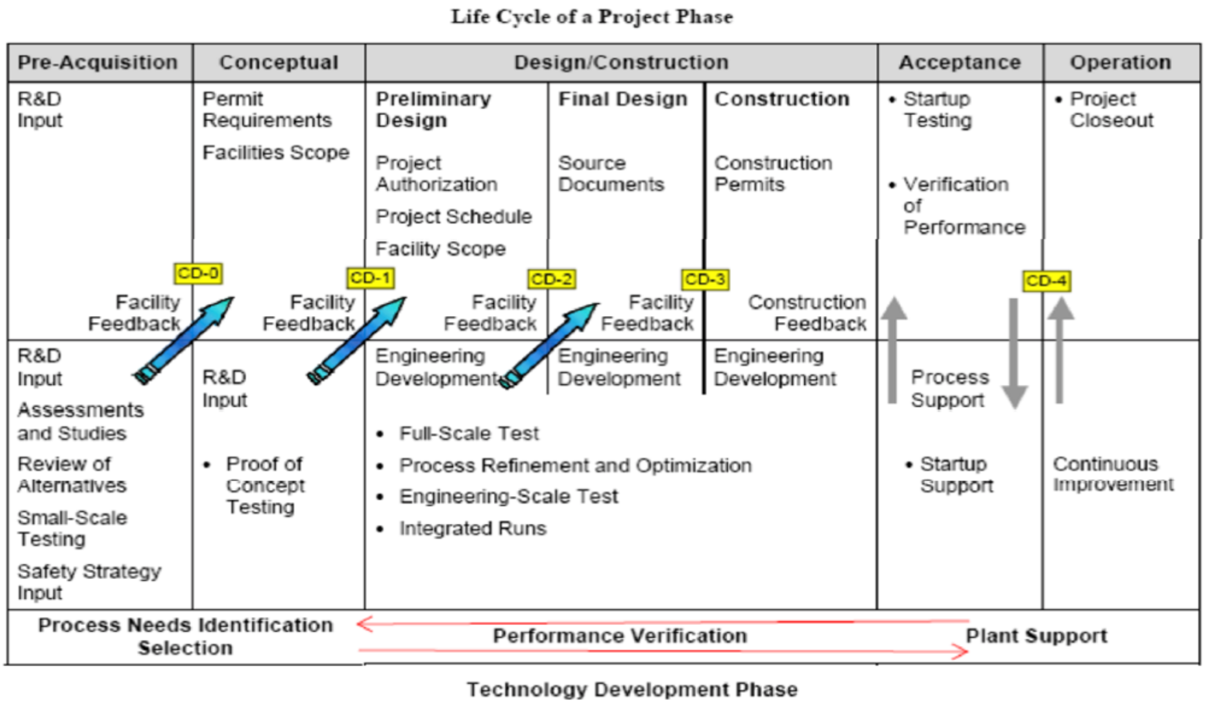


Figure 5-2. Department of Energy Illustration of Integration of Technology Development with Design Project Evolution [63]

A potential requirement for advanced reactors will be a prototype plant. 10 CFR 50.43(e) states data obtained from the operation of a research or test reactor could be used to fulfill the testing requirement if insufficient data are available from integral and separate effects testing. The FOAK application for an initial prototype allows issuance of a COL with restrictions to obtain testing data. The test program is used to satisfy any outstanding testing needs. Upon successful completion of the prototype testing, the COL restrictions are removed, and the final DCD can be issued. The licensing requirements for a prototype reactor are also flexible with options to submit for an SDA either before or after prototype testing. For example, the preparation of a 10 CFR Part 52 SDA could occur in parallel with a prototype 10 CFR Part 50 CP and OL. The two-step process for the prototype accelerates the development while the SDA or DC is prepared. Other options for integrating a prototype in series with an SDA or DC are also described in the NRC advanced reactor roadmap [61].

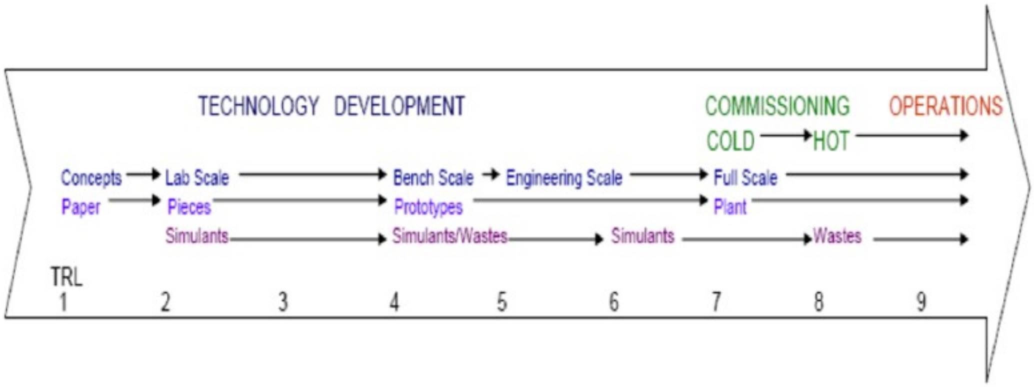


Figure 5-3. Department of Energy Office of Environmental Management Technology Readiness Levels [63]

5.2. Two-Step Licensing Process—10 CFR Part 50

5.2.1. Construction Permit

Under 10 CFR Part 50, an application for a CP consists of three types of information [60]:

- Preliminary safety analysis
- Environmental review
- Financial and antitrust statements
- Statement of need for power plant

The receipt of an application initiates a review by the NRC to ensure that the submitted information is complete. A complete submission (i.e., with all the information required for review) is then reviewed, with the NRC finding documented in a safety evaluation report. This report evaluates the application with respect to site safety characteristics and emergency planning.

As part of this process, the NRC encourages public involvement by holding public meetings in the locality of the proposed site, with the goal of familiarizing the public with the safety and environmental impact of the proposed plant, As part of this process, public meetings between the NRC and applicant are held that enable the public to gain more information regarding the design and construction of the proposed plant.

In addition to this safety review, the NRC also undertakes an environmental impact assessment under the National Environmental Protection Act (NEPA). This review assesses potential negative environmental impacts, contrasting with the benefits expected to be attained by deployment of the plant. A draft environmental impact statement is issued by the NRC following its review. This is subject to public comment, as well as input from other Federal, State, and local government agencies. A final environmental impact statement is issued that addresses all comments received by the NRC.

A public meeting is also held by the Advisory Committee on Reactor Safeguards (ACRS)[13], which reviews each CP application and safety evaluation report. The ACRS reports its findings to the five-member Commission. In addition to the ACRS public meeting, the Atomic Safety and Licensing Board holds a public hearing.

An applicant that adopts the 10 CFR Part 50 licensing approach has the potential to begin the licensing and construction process at an earlier stage in the design than allowed for under 10 CFR Part 52. A risk associated with the 10 CFR Part 50 licensing approach is that a CP does not imply that the NRC will not identify additional requirements, leading to design changes, prior to issuing an OL (see Section 5.2.2). This can introduce project risk leading to significant financial exposure to either the vendor or the eventual operator of the nuclear plant.

However, for designs such as the micro-reactor, which are FOAK, the opportunity to develop a prototype under 10 CFR Part 50 may be appealing to vendors. In this manner, a conceptual design could evolve considerably based on construction insights. This does not preclude subsequent deployment of the reactor design from following 10 CFR Part 52, once it has progressed beyond a FOAK prototyping phase. In this sense, subsequent applications can realize the regulatory certainty expected from a mature design.

For a micro-reactor, the construction phase of a plant is likely to be significantly different from that for a large-scale nuclear power plant. Even large nuclear power plants that are constructed in a modular fashion require significant on-site resources to be dedicated to construct the nuclear island and BOP. In contrast, micro-reactors may be able to be largely constructed off-site at a remote manufacturing facility. The activities related to "construction" on-site may largely be related to site preparation activities and installation of equipment not associated with the nuclear island (i.e., BOP components). The installation of the actual nuclear reactor could conceivably involve a relatively minor effort, fixing the unit in a below-grade structure and connecting the unit to equipment required to convert thermal energy into a desired output from the facility (e.g., electrical energy to process heat).

In the situation of a micro-reactor, a CP may likely involve activities primarily associated with:

- Manufacture of the micro-reactor (off-site)
- Transportation of a micro-reactor to an installation site with fresh fuel
- Construction activities to prepare the site to accept the micro-reactor unit
- Installation of the micro-reactor unit at the site

The benefits to a micro-reactor vendor from the 10 CFR Part 50 process may be more limited than with a large-scale nuclear reactor. Many of the benefits of the licensing process commencing earlier in the design, with the design being informed by the construction process, may not apply to a micro-reactor. It is conceivable that more benefits would be attained by micro-reactor vendors through a subsequent stage in the CP that would allow the installed facility to be extensively tested.

Many of the challenges faced by a FOAK nuclear plant, even a simplified one such as a micro-reactor, stem from the integral interaction of numerous physical processes and engineered systems

[13] The ACRS is an independent advisory board consisting of technical experts.

that can never be fully evaluated at smaller scales or through separate-effect testing. To achieve maturation of a design sufficient to provide reasonable assurance that public health and safety are protected, integral testing of a prototype reactor is necessary.

This may require novel methods to perform integral testing of the micro-reactor unit at the manufacturing site (prior to shipment to final installation). This may require development of an extended set of regulations by the NRC. This is a practice without significant precedent in the nuclear energy sector. Since the thermal energy produced by micro-reactors is relatively low, the associated risk may not be significant. However, evaluation of this risk and a regulatory framework by which this risk can be kept low during any integral testing at a manufacturing facility may need to be developed.

There are also efforts that have been underway for a number of years to deploy high-fidelity modeling and simulation technology to the nuclear energy sector. The advantage of this computational technology is the potential to eliminate the need for extensive integral testing. High-fidelity computational modeling technology has been deployed and utilized in the nuclear energy sector, with some examples of evaluating at high-fidelity specific phenomena observed in fuel testing facilities.

NOTE: The successful deployment of high-fidelity computational technologies to the nuclear weapons complex at DOE government labs occurred following the 1992 moratorium on nuclear weapons testing in the United States. The adoption of high-fidelity modeling and simulation, as part of the DOE Stockpile Stewardship Program (SSP), was necessary in light of this moratorium to ensure that the nuclear stockpile could be maintained. The Advanced Simulation and Computing (ASC) program was created in 1995 as part of the SSP to realize the goal of maintaining the nuclear stockpile without live nuclear testing.

It is important to note that significant effort was required under the ASC program to demonstrate the predictive capabilities of high-fidelity, multi-physics computer codes. A particular challenge in any computational simulation of integral system behavior (multi-physics modeling and simulation) is demonstrating that integral behavior is captured correctly. The integral response is governed by interaction of a multiple complex phenomena; and these interactions cannot be elucidated through separate-effect tests utilized to validate the individual physical models. This requires validation that shows the coupling of computational models for specific phenomena represents nature correctly. This is a non-trivial problem, and involves carefully designed integral testing.

An extensive effort was undertaken in the ASC program in the DOE nuclear weapons complex to develop and deploy methods for Verification, Validation and Uncertainty Quantification (VVUQ) [64].

There remain limited examples of the *validation* of high-fidelity modeling and simulation in the nuclear energy sector capturing integral response of a nuclear plant under operational and accident conditions. Some examples exist under the DOE Consortium for Advanced Simulation of Light Water Reactors (CASL) program, particularly in operational space. The CASL Virtual Environment for Reactor Applications (VERA) has demonstrated initial capabilities to predict operational response of PWRs. However, the capabilities are still in a developmental stage for non-LWRs designs, though work has begun on extending VERA to represent MSRs. More extensive deployment of predictive, high-fidelity computation would provide significant value to the nuclear energy sector. However, it is not at a technology readiness level yet to be utilized in the near-term to replace the need for integral testing of prototype micro-reactor designs. Significant investment in VVUQ would be required to demonstrate a technology-readiness level to accept high-fidelity modeling and simulation as part of a micro-reactor licensing basis.

NOTE: An example of the challenges with integrating high-fidelity computational models into the reactor design process is evidenced by the Multipurpose Applied Physics Lattice Experiment (MAPLE) test reactors. These facilities were designed by Atomic Energy of Canada Limited (AECL). They were designed to replace the Nuclear Reactor Universal (NRU) at Canadian National Laboratory (CNL), formerly Chalk River National Laboratory (CRNL). The NRU facility produced a significant amount of the world's supply of Molybdenum-99 (Mo-99) used in a range of nuclear medicine applications. The MAPLE reactors were designed and licensed based on a requirement that they would have a negative power coefficient. Unfortunately, the computational methods utilized to demonstrate this negative power coefficient were in error, which was introduced by the multi-physics coupling (integral effects) between neutronic and structural responses.

The above illustrate challenges to realizing the maturation of a micro-reactor technology where much of the effort to construct the nuclear system is achieved at a manufacturing facility. Reactor vendors may seek to achieve technology maturation outside of the 10 CFR Part 50 process by constructing prototype micro-reactors on U.S. Department of Defense (DOD) or DOE facilities. In this manner, development of the manufacturing processes could be performed on-site at an installation not regulated by the NRC. This process would have to develop the necessary confidence in the manufacturing technology to support commercial deployment, and a NRC licensing process with greater regulatory certainty. In this situation, maturation of a micro-reactor technology on a DOD or DOE facility would then support adoption of a 10 CFR Part 52 licensing approach.

5.2.2. Operating License

For applicants following the 10 CFR 50 approach, the construction period allows an applicant time to finalize a design and formalize operational plans. The final design information and operation plans are developed by the applicant prior to submitting an application for an OL. The application for an OL consists of the following documentation:

- FSAR—consisting of a description of the plant's final design, operational limits, anticipated plant response to accidents, and emergency response plans

- Updated environmental report (following requirements of the NEPA)

The NRC reviews this documentation and issues a final safety evaluation report. The ACRS reviews both the OL application and the NRC final safety evaluation report, providing findings and recommendations to the Commission. Members of the public that could be affected by operation of the nuclear plant may petition the NRC for a hearing. Any public hearing is convened by the Atomic Safety Licensing Board.

Further discussion of information requirements for a FSAR are presented in Section 5.3.3, where the contents of an LWR SRP are presented. This report does not propose that a micro-reactor, or any other non-LWR, applicant should follow the LWR SRP. Rather, the purpose of this presentation is to provide an illustration of the information needs for a micro-reactor relative to what is expected for an LWR. This approach is adopted given the extensive use of the LWR SRP.

For a micro-reactor, it is conceivable that a more extensive set of stakeholders may become involved. The nature of such a review will be determined by the applicant's plans to transport the fueled reactor from a manufacturing facility to its final installation point (although some of these issues will likely be encountered during approval of a CP). In addition, the applicant's plans to transport the reactor following its end-of-life may also result in a broader set of stakeholders engaging prior to issuing an OL.

It is at the phase of an OL that significant state-of-knowledge and practice challenges are likely to emerge. Examples of key areas requiring demonstration, that will need to be supported by efforts during a construction phase, are as follows:

- Operational limits and procedures will need to be established to either conservatively bound the range of operational states that could emerge, or be based on an extensive prototyping program

- Demonstration of autonomous control strategies and their domain of operability

- Demonstration of the capability of risk assessment methodologies to evaluate the risk profile for micro-reactors relying heavily on inherent safety functions[14]

- Correlation of risk with operational state (based on the internal *and* external conditions of the micro-reactor) to support a risk-informed framework for operational decision-making

- Consequence evaluation methodologies considering unique fission product speciation (a function of the working fluid), potentially short distances to site boundaries, and the impact on fission product release for reactors installed below-grade

- Material behavior under prolonged exposure to radiation and high temperature (and high-pressure, depending on the design)

[14] As noted above, inherent safety systems generally provide a robust means of achieving a particular safety function. This ensures that the facility is robust against random SSC failures that represent relatively significant contributors to risk for the current generation of operating LWRs. This risk is evaluated as part of an internal events PRA, which seeks to identify how consequences to public health and safety can emerge due to random failures of SSCs, many of which are active and require energy input and a change of state to achieve the desired safety function. It is possible that the risk profile for micro-reactors (as well as other non-LWRs), which rely on inherent safety systems, could be significantly dominated by external events (such as seismic or flooding events), or malevolent acts. This presents more significant challenges to the characterization of risk, as well as assessing how the operational state of a micro-reactor (internally and externally) is correlated with a level of risk. In the absence of this correlation, it is not likely that operational decision-making could be risk-informed.

- Monitoring strategies to ensure the reactor remains within a safe-operating envelope, including assessment of material conditions[15]

- Staffing requirements for an operating micro-reactor relying on significant levels of autonomous control and sited in a remote location

- The role of a control room (if any) and the extent to which operational procedures should require manual intervention given reliance on inherent safety functions

- The training requirements for on-site personnel give the significantly different demands relative to current large-scale nuclear power generation facilities

- Emergency response strategies given potentially more limited staffing requirements under normal operating conditions[16]

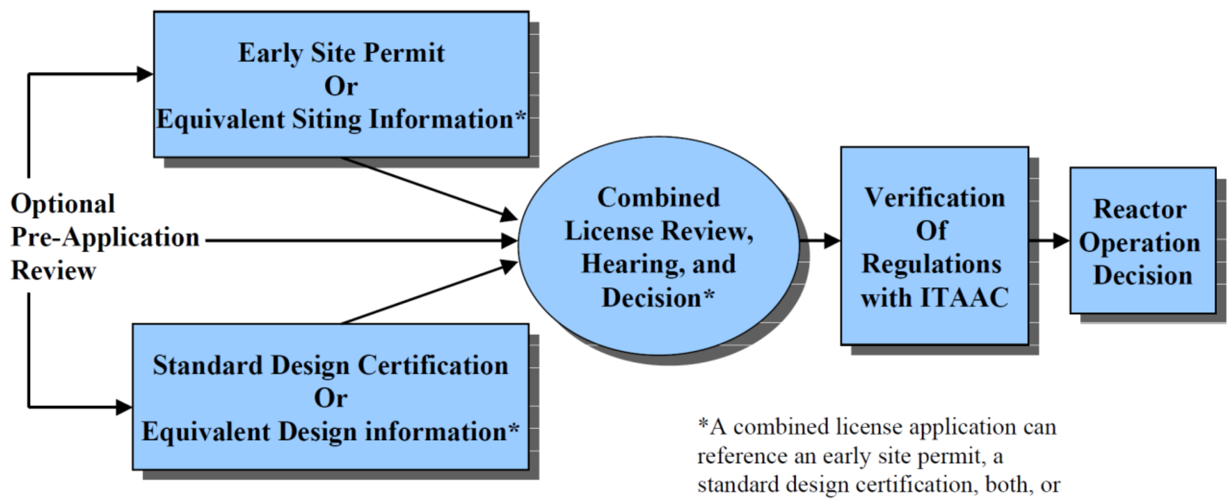
- Accident management actions given different accident behavior, plant design (e.g., working fluid and potentially energetic chemical interaction with water), and on-site staffing

5.3. Additional Licensing Processes—10 CFR Part 52

While designers may follow a licensing approach following the processes specified under 10 CFR Part 50, they are not excluded from instead adopting an approach following 10 CFR Part 52.

10 CFR Part 52 covers requirements for the ESP, DC, and COL. Figure 5-4 shows the relationship between the three processes that constitute the overall licensing approach under 10 CFR Part 52.

[15] This is particularly important in the event that sufficient operational experience does not already exist.
[16] This may need to include strategies to move equipment and personnel to a remote location to address an emergency for which the on-site staff is not sufficiently equipped.



*A combined license application can reference an early site permit, a standard design certification, both, or neither. If an early site permit and/or a standard design certification is not referenced, the applicant must provide an equivalent level of information in the combined license application.

Figure 5-4. NRC Licensing Procedure [65]

5.3.1. Early Site Permit

The ESP resolves site safety, environmental protection, and emergency preparedness issues independent of a specific nuclear plant design [66]. The application contains the following information:

- site boundaries;
- seismic, meteorological, hydraulic, and geologic data;
- existing and projected future population of the surrounding area;
- evaluation of alternative sites;
- proposed general location of each plant planned to be on the site;
- number, type and power level of the plants planned for the site;
- maximum discharges from the plant;
- type of plant cooling system to be used;
- radiation dose consequences of hypothetical accidents; and
- plans for coping with emergencies.

The specific challenges in the ESP process for deployment of a micro-reactor include limiting the site and exclusion boundary size for economic viability. The recent NRC public meeting on regulatory process improvements for advanced reactor designs [67] highlight some of the challenges in siting advanced reactors closer to population areas. Small micro-reactors would not challenge the Regulation Standard 4.7 siting criteria and would greatly benefit from evolving regulations for siting small modular reactors (SMRs) and non-light water reactors [68]. While new proposed siting criteria offers advantages for small reactors with low radioactive inventories, the new siting options are a significant change from existing LWR siting requirements that identify 10 miles for plume exposure and 50 miles for the ingestion pathway (i.e., 10 CFR 50.47(c)(2) or NRC Regulation Guide 4.7). The proposed siting methods include the challenge to develop an acceptable MST and the evaluation of the near-field consequences.

Similarly, the staffing of the micro-reactor, the emergency preparedness, the aircraft impact requirements (i.e., micro-reactor concepts are proposing installation below-grade in such a manner to alleviate or eliminate the hazard of an aircraft impact), the security protections and staffing, and the potential for remote operation are also cited as challenges for micro-reactor siting and operation. Unlike large current generation or even proposed SMRs reactors, the simplicity and scale of a micro-reactor challenges the fundamental concepts of emergency preparedness, the cross-section for malevolent aircraft impact, and the security or operations staffing requirements.

5.3.2. Combined License

A COL under 10 CFR Part 52 authorizes the facility construction and subsequent operation following verification of the regulations with ITAAC. The COL is typically supported by an ESP and a standard DC. Sections 5.3.1 and 5.3.3 discuss how these elements of the 10 CFR Part 52 licensing approach apply or require special considerations for the micro-reactor design.

5.3.3. Standard Design Certification

The LMP provides a framework by which a non-LWR designer can establish levels of safety following a risk-informed, performance-based approach [9]. The LMP framework does not provide guidance, however, for applicants regarding how to structure the content of an application. The NRC is currently collaborating with industry on a second phase of the LMP. This collaborative effort is focused on developing a methodology for using the results of an LMP-guided risk-informed application to develop a regulatory application. This effort is the Technology-Inclusive Content of an Application Project. Since this project is still ongoing, this section focuses on discussing how the contents of an SRP may or may not apply to preparing a regulatory application for a micro-reactor. The SRP outlines the following structure for the preparation of a DC [48].

- Chapter 1, Introduction and Interfaces
- Chapter 2, Sites Characteristics and Site Parameters
- Chapter 3, Design of Structures, Components, Equipment, and Systems
- Chapter 4, Reactor
- Chapter 5, Reactor Coolant System and Connected Systems
- Chapter 6, Engineered Safety Features
- Chapter 7, Instrumentation and Controls
- Chapter 8, Electric Power
- Chapter 9, Auxiliary Systems
- Chapter 10, Steam and Power Conversion System
- Chapter 11, Radioactive Waste Management
- Chapter 12, Radiation Protection
- Chapter 13, Conduct of Operations

- Chapter 14, Initial Test Program and ITAAC-Design Certification
- Chapter 15, Transient and Accident Analysis
- Chapter 16, Technical Specifications
- Chapter 17, Quality Assurance
- Chapter 18, Human Factors Engineering
- Chapter 19, Severe Accidents

In general, the overall safety objectives of the LWR SRP will be applicable to micro-reactors. The DC document includes relatively generic requirements to describe the plant (e.g., the steam and power conversion system and the SSC). The discussion of areas where SRP requirements require additional clarification for the micro-reactor DC are discussed in the following subsections. Not all of the above areas are discussed below. The following discussion identifies areas judged to require additional clarification. Note that Section 4.4 provides additional discussion in the area of testing programs.

5.3.4. Chapter 2, Sites Characteristics and Site Parameters

Chapter 2 contains the description and requirements for the plant siting. The site boundary, the exclusion area, and the evacuation requirements and acceptance criteria are described. The micro-reactor designers desire to have small site boundary with commensurate or no additional requirements for an exclusion zone. The technical approach for the smaller micro-reactor site boundary is an example where additional regulatory guidance is needed (e.g., through a staff-generated example and/or a micro-reactor siting regulatory guide). The unique needs for micro-reactor siting include guidance on (a) the MST, (b) the near-field atmospheric dispersion from an underground reactor, and (c) the small radionuclide inventory in a micro-reactor but from fuels with higher enrichment than current LWRs.

Much of the remaining requirements of Chapter 2 are directly applicable including evaluations and requirements for climatology, meteorology, hydrology, floods, dams, tsunami, ice, groundwater, seismic, subsurface stability, etc.

5.3.5. Chapter 3, Design of Structures, Components, Equipment, and Systems

The LMP advocates a risk-informed approach for classifying SSCs [9]. The SSCs for a micro-reactor will be different than conventional LWRs, but conceptually addressed through the same risk-informed process. When the SSC classification is combined with appropriate regulatory treatment, the process allows identification of safety and risk-significant SSCs.

Perhaps the most significant challenge in the Chapter 3 requirements is the specification of a functional containment versus standard requirements for concrete or steel containments (i.e., Section 3.8). The micro-reactor may use a functional containment, which may increase designer and regulatory burden to demonstrate functional containment compliance versus a traditional containment [41].

The fail-safe design of a reactor protection system for reactivity control will necessarily be significantly different for a micro-reactor (SRP Chapter 3.9.4). However, the safety requirements will be similar (i.e., "a positive means for inserting the rods shall always be maintained to ensure appropriate margin for malfunction, such as stuck rods"). For the micro-reactor, the control system may be a control drums rather than control rods.

Chapter 3 also includes many sections that may be non-applicable to micro-reactors including piping failures and leak before break (Chapter 3.6).

Many of the other functional requirements in Chapter 3 of the SRP will be applicable. These requirements include seismic classification (Chapter 3.2), high wind loading (Chapter 3.3), flooding protection (Chapter 3.4), and seismic evaluations (Chapter 3.7).

5.3.6. Chapters 4 through 6 – Reactor and Containment Design

The NUREG-0800 SRP for the reactor design is generally not applicable. The possible functional translations to the micro-reactor include,

- Nuclear fuel design attributes (Chapter 4.2 and 4.3)
- Reactor and control system structural materials (Chapter 4.5 and 4.6)
- Reactor vessel boundary materials (Chapter 5.3)

Chapter 6 would be replaced with a functional containment evaluation [41].

5.3.7. Chapter 7, Instrumentation and Controls

The instrumentation and control are expected to be relatively simple for the micro-reactor. A challenge for micro-reactors would be consideration of autonomous or remote operation. The other functional capabilities cited in the SRP are generally applicable. For example,

> The necessary functions are those needed to monitor variables and systems over their anticipated operating ranges, to maintain these variables and systems within their prescribed operating ranges, to automatically initiate the operation of systems and components to assure that fuel design limits are not exceeded as a result of anticipated operational occurrences, and to sense accident conditions and initiate the operation of systems and components important to safety. Chapter 7.1, [48]

5.3.8. Chapter 9, Auxiliary Systems

The majority of Chapter 9 is related to LWRs. However, Chapter 9 includes refueling and spent fuel pool requirements. Although the design details may change, the current micro-reactor designs will not be refueled. Consequently, the plant would immediately transition to decommissioning after the life of the factory fuel loading.

5.3.9. Chapter 11 and 12, Radioactive Waste Management and Protections

Radioactive waste management and radiation protection would be relatively non-applicable for micro-reactors. The components of the micro-reactor are intended to operate for the lifetime of the plant without refueling. The fuel has several barriers to the heat pipe (i.e., the common core heat transfer system) and will not generate any gaseous, liquid, or solid effluents that must be managed outside the reactor system. The "as low as reasonably achievable" concept is applicable but largely irrelevant due to the passive plant operation, no refueling, no spent fuel, and minimal maintenance. Only identified plant inspections require radiation management.

Micro-reactor concepts have also introduced the idea that upon completion of the operating lifecycle of the reactor, it will be transported from its site to another facility that will process the spent fuel. The consideration of radioactive waste management must as a result expand to consider transportation risk.

While it is reasonable to assume that transportation of a micro-reactor could be regulated following an approach similar to transportation of equivalent new fuel material, the transportation of a used micro-reactor poses fundamental challenges not considered in the regulation of spent fuel transportation. In particular, there has been limited consideration of the transportation package that would be required to safeguard the spent fuel inside an entire micro-reactor while in transport. It is not clear if a transportation package can be engineered that address current requirements for an entire micro-reactor.

5.3.10. Chapter 13, Conduct of Operations

Some areas of the conduct of operations are applicable to micro-reactors (e.g., design and construction qualifications, license holder and staff qualifications). However, the operational staff requirements will be significantly simpler. For example, some micro-reactor vendors have advocated for autonomous or remote operation, which certainly requires further regulatory guidance. There may be no requirements for traditional positions of chemistry, mechanical, electrical, fire protection, and training staff.

5.3.11. Chapter 15, Transient and Accident Analysis

As discussed in Section 3.1, the micro-reactor LBEs can be categorized using the SRP (i.e., the generic challenges or initiating events are applicable). However, the evaluation of non-light water micro-reactor LBEs will be significantly different (e.g., see Section 4). Furthermore, there may be some additional challenges unique to specific micro-reactor designs. One purpose of a micro-reactor PRA is the identification of LBEs, which comprise AOOs, DBEs, BDBEs, and DBAs. The regulatory challenges will be an assessment of completeness, accurate accident classification (e.g., AOOs, DBEs, BDBEs, and DBAs under the LMP), and the appropriate acceptance criteria (e.g., see the proposed LMP recommendation in Figure 5-5). The content in a micro-reactor application, aligned with the LWR Chapter 15 analyses, need to demonstrate the safety conservatism and the defense in depth historically present in current-generation LWRs.

Chapter 15 of the SRP also includes conservative radionuclide release and dose evaluations following loss-of-coolant accidents (Section 15.6.5). The risk-informed application defined by the LMP allows an applicant the flexibility to assess how a design limits off-site consequences based on a PRA. In this evaluation, the spectrum of radiological consequences are determined through structured identification of event scenarios that result in radiological releases to the environment. The application of PRA to assess various measures in the design to limit the extent of radiological release to the environment is currently within the scope of state-of-art PRA methods. The NRC Containment Protection and Release Reduction study illustrated (for a Boiling Water Reactor (BWR) Mark I) how engineered and accident management measures could ameliorate the off-site consequences from a spectrum of radiological release events [69] [70].

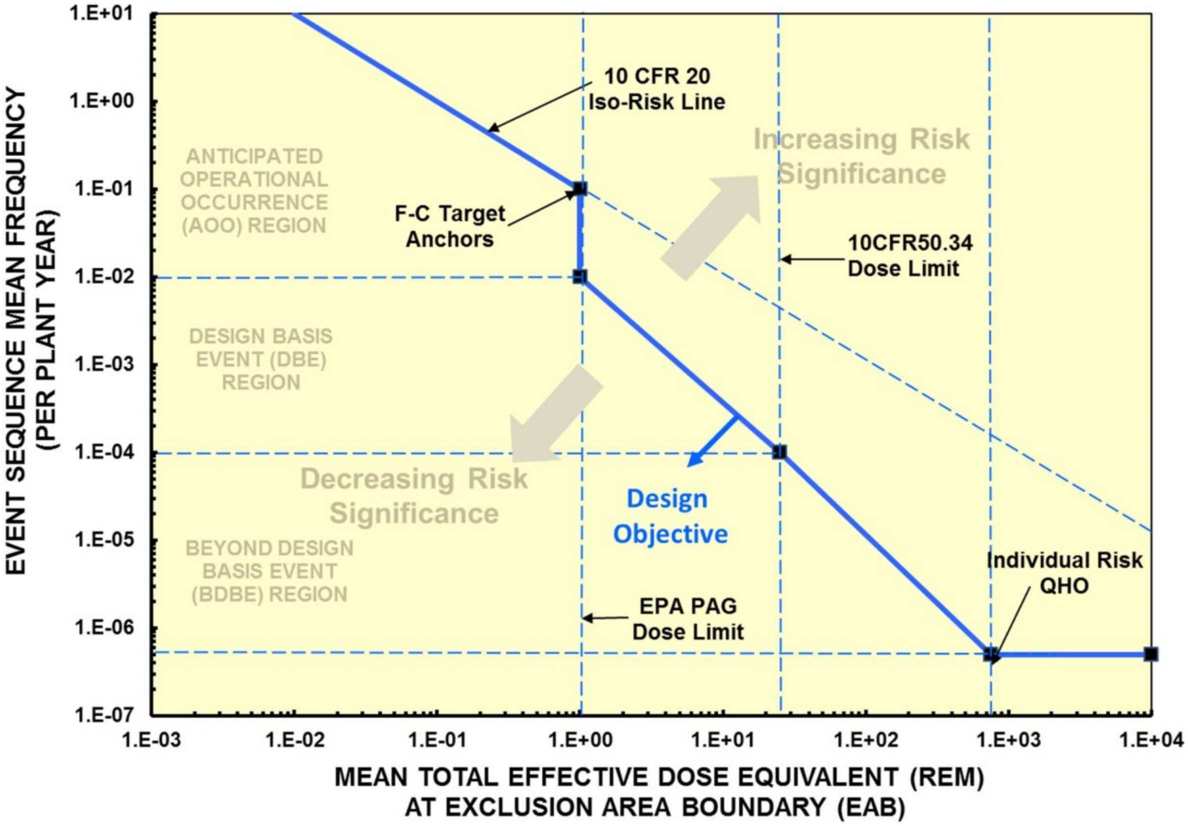


Figure 5-5. NEI 18-04 F-C Target Curve [9]

5.3.12. Chapter 19, Severe Accidents

Chapter 19 includes development and quantification of a PRA for LWRs. As discussed above, completeness of the evaluation and accuracy of the quantification is more important in this process than LWRs where Chapter 15 deterministic analysis requirements and historical precedents have governed their safety and design evaluations.

The LMP process is intended to identify the design's deterministic, probabilistic, and performance requirements. The content of this chapter for micro-reactors following the LMP will likely be combined with content traditionally placed in Chapter 15 of the LWR SRP. There are a number of ways in which the traditional content of the LWR SRP Chapter 15 and Chapter 19 information can be organized. For example, a micro-reactor application could be organized into sections or chapters as follows.

- Plant Safety Design: Classification of SSCs and DID Approach
- Internal Event Probabilistic Models for SSCs
- External Event Probabilistic Models for SSCs
- Internal Event Integrated Probabilistic Model
- External Event Integrated Probabilistic Models
- Accident Progression Analysis

- MST Analysis
- Radiological Release Categorization and Consequence Evaluation
- Assessment of Safety Design Robustness to Classification of SSCs
- Assessment of Safety Design Robustness to DID Approach

Chapter 19 also includes requirements for loss-of-large areas of the plant due to explosions and fires and aircraft impacts. The simplicity and small (i.e., likely underground) footprint of the micro-reactor may introduce special considerations for these events. It may be logically impractical for an aircraft to hit a micro-reactor and a large fire or explosion may not have a significant impact to a below-grade reactor. Nevertheless, the designers must address long-term heat removal requirements without offsite power and limited access from such significant events. The regulatory challenge will be determining the specific requirements for micro-reactors or allow reactor designers to apply for exceptions. These defined security events are beyond the scope of the PRA and therefore explicitly cited in the SRP Chapter 19 requirements.

6. SUMMARY OF DEVELOPMENTAL GAPS

6.1. Licensing

Commercial micro-reactors pose a challenge to the existing NRC process for licensing the construction and operation of power reactors. All current-generation licensed commercial power reactors are stationary, largely assembled on-site, and use oxide fuel cooled by water. The proposal of micro-reactors comes at a time when the NRC is considering the larger question of how to license non-LWRs in general. A new framework, the LMP, has been proposed by the nuclear industry to accommodate non-LWR reactor types by making the license application process more technology-inclusive [71]. The LMP was produced with a focus on liquid metal fast reactors, high temperature gas reactors, and MSRs and has not yet been rigorously evaluated against micro-reactor designs.

6.2. Engineering

Micro-reactors at the same time solve some engineering questions while creating others. Micro-reactors are much smaller than existing power reactors and will be built in better-controlled factory conditions. However, many of the proposed designs use novel fuels and have safety-significant phenomena that have not historically been studied to as much depth as those currently credited in power reactors. The potential micro-reactor vendors propose transportation of fueled reactors to operating sites as well as transportation of the intact reactor with its burned-up fuel load to a storage or processing site. These challenges combine to generate significant uncertainty around the safety basis for such reactors which will have to be managed through a robust engineering research and development program.

6.3. Fuel Cycle

Micro-reactors propose using HALEU fuel in order to allow for an extended core operating lifetime. HALEU has U-235 enrichment greater than 5% but less than 20%. Existing commercial reactors in the United States use fuel enriched to less than 5%. Assuming that commercial enrichment and fuel production will be pursued, this presents a number of technical and regulatory challenges for the front end of the fuel cycle, including [42]:

- Criticality safety studies must be performed in support of licensing and operating HALEU facilities.

- Special nuclear material, in reactor quantities, crosses from safeguards Category III into Category II at 10% enrichment, and into Category I at 20% enrichment. Guidance will be required for material control and accountability and physical security at Category II facilities. The NRC currently has no licensees that are considered Category II special nuclear material facilities.

- Transportation package design and qualification will be required for HALEU materials from raw forms (i.e., UF_6) to manufactured fuel elements and assembled reactors.

In the meantime, the DOE has proposed down-blending existing HEU stocks to provide approximately 10 metric tons of HALEU for research and development purposes for commercial industry and government entities [72].

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