BNL-207816-2018-INRE

Phenomena Important in Liquid Metal Reactor Simulations

Manuscript Completed:

August 31, 2018

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is preparing for the future licensing of advanced reactors that will be very different from current light water reactors. Part of the NRC preparation strategy is to identify the simulation tools that will be used for confirmatory safety analysis which includes normal operation and abnormal conditions. This report advances that strategy for reactors with a fast neutron spectrum that use liquid metal coolants. This includes reactors using sodium, lead, or a lead-bismuth eutectic. Although all types are discussed in this report, the emphasis is on simulating sodium-cooled reactors as more information is available on those designs. The specific designs discussed in the report are a subset of many deigns being considered in the U.S. and elsewhere. This subset of reactors are designs that are considered the most likely to begin the application process with the NRC in the near future.

The objective herein is to identify the dominant physical phenomena and modeling requirements expected for the simulation of liquid metal fast spectrum reactors. Phenomena are discussed in terms of their impact on normal and transient/accident conditions, and their modeling in neutronics, heat transfer, and fluid dynamics simulation tools. The study makes use of past experience with such concepts, for example, from the Experimental Breeder Reactor (EBR-II) that ran for almost 30 years at Idaho National Laboratory; from recent conceptual design studies such as for the PRISM concept; and from similar technology-gap studies carried out for the Department of Energy.

Lists of important phenomena that will be modeled to simulate normal operation and transients/accidents were generated. These lists can be used by a panel of experts as the starting point for generating a phenomena identification and ranking table (PIRT). A thorough PIRT exercise in the future could determine what research and development may be required before the necessary simulation tools have sufficient capability.

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ACKNOWLEDGEMENTS

The authors appreciate the thoughtful technical comments from Stephen Bajorek used to guide this document and the assistance of Lynda Fitz in handling the administrative needs of this project.

ACRONYMS

ALFRED	Advanced Lead-Cooled Fast Reactor European Demonstrator
ALMR	Advanced Liquid Metal Reactor
AOO	Anticipated Operational Occurrence
ARC	Advanced Reactor Concepts
ATWS	Anticipated Transient Without SCRAM
BDBA/E	Beyond-Design-Basis Accident/Event
DBA/E	Design Basis Accident/Event
DLFR	Demonstration LFR
EMP	Electromagnetic Pump
FP	Fission Products
GNEP	Global Nuclear Energy Partnership
HCDA	Hypothetical Core Disruptive Accident
IHTS	Intermediate Heat Transport System
IHX	Intermediate Heat Exchanger
IRACS	Intermediate Reactor Auxiliary Cooling System
LBE	Lead-Bismuth Eutectic
LFR	Lead-Cooled Fast Reactor
LMR	Liquid Metal Reactor
LWR	Light Water Reactor
NRC	U.S. Nuclear Regulatory Commission
NRO	Office of New Reactors
NSSS	Nuclear Steam Supply System
PHTS	Primary Heat Transport System
PIRT	Phenomena Identification and Ranking Table
PSID	Preliminary Safety Information Document
RVACS	Reactor Vessel Auxiliary Cooling System
SFR	Sodium-Cooled Fast Reactor
SG	Steam Generator
TRU	Transuranics
TWR	Traveling Wave Reactor
ULOF	Unprotected Loss of Flow
ULOHS	Unprotected Loss of Heat Sink
UNF	Used Nuclear Fuel
UTOP	Unprotected Transient Overpower
U-TRU-Zr	Uranium-Transuranic-Zirconium

1 INTRODUCTION

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) is preparing for the future licensing of advanced reactors that will be very different from the light water reactors (LWRs) that are currently used to generate electricity throughout the U.S. In particular, these advanced reactors will use gas, liquid metal or molten salt rather than water as a coolant. The Office of New Reactors (NRO) has developed a vision and strategy document [1-1][a] which outlines the tasks that must be undertaken to achieve technical and regulatory readiness for these non-LWRs. This document is supported by an implementation action plans (IAPs) [1-2] that cover the actions to be taken in the next five years based on six basic strategies.

IAP Strategy 2 is to “acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews.” Ultimately, the consequences of reactor accidents are assessed with regard to the theoretical magnitude of radiological (or hazardous material) exposure to members of the public. Although it will be necessary for the NRC to develop tools for assessing the release and transport of radioactive material in accidents, the immediate priority is to identify a set of computational tools that can be developed to model neutronics, heat transfer and fluid dynamics. This will allow for time-dependent simulations in the fuel and coolant of neutron flux, power density, temperature, flow rate, and pressure for these advanced non-LWRs. It is vital to understand and be able to predict behavior during normal operation, anticipated operational occurrences, transient events, and accidents and the physical phenomena that dominate those events.

As part of the NRC evaluation of a new design, confirmatory safety analyses are performed in order to understand the validity and accuracy of computational methods being used by licensees, the sensitivity of results to uncertainties, and the safety margin under varying conditions from normal operation to design-basis and beyond design-basis accidents. This may be particularly important with non-LWR designs where there is much less regulatory and computational experience relative to LWR designs. The calculations performed by the NRC as part of a confirmatory analysis do not represent licensing basis for a design. This is the responsibility of the applicant, who may choose to perform conservative calculations with bounding assumptions. The NRC calculations are generally performed with realistic assumptions and models that are intended to represent the actual behavior of the plant.

Before one can select the simulation codes that are needed, it is necessary to first understand the scenarios that will need to be analyzed. This requires understanding the design of advanced reactors and then understanding normal operation and the potential upsets that may occur. The next step is to understand what physical processes must be modeled by the computer codes. It is at that point that NRC can

[a] References are provided separately in each chapter.

survey the existing codes to see which might be optimal, with one of the criteria being that it requires the least amount of resources. “The emphasis in the [NRC] staff’s approach is to leverage, to the maximum extent practical, collaboration and cooperation with the domestic and international community interested in non-LWRs with the goal of establishing a set of tools and data that are commonly understood and accepted.” [1-1, 1-2]

The intent is to consider all designs under consideration in the U.S. that might result in a license or design submittal to the NRC in the next decade. Of the three types of non-LWR designs mentioned above, NRC has had the most experience with analysis of gas-cooled reactors. Hence, the most pressing needs with respect to Strategy 2 are with liquid metal and molten salt cooled reactors. Liquid metal reactors are fast spectrum reactors and use either sodium or lead or a lead-bismuth eutectic as coolant. They are the subject of this report while molten salt reactors with either a thermal or fast spectrum are considered in a companion report. [1-3]

1.2 Objectives

The objective of this study is to start the process outlined in Strategy 2 for liquid metal fast spectrum reactors, that is, to understand the modeling needs of simulation codes. Since variants of these reactors have been designed and built in the U.S. and other countries there is considerable information available on the subject. Hence, the first step is a literature survey to gather information on the proposed designs, the scenarios that need to be simulated and the physical processes (aka phenomena) that need to be modeled. The focus (boundary conditions) for the project (to be relaxed at a later time) is the primary system but some consideration is given to other parts of the plant. The scenarios to be considered include accidents up to the point where clad may be breached and fission products are no longer contained as designed, or where liquid metal can leak from the system. The recommendation of specific computer tools for use by the NRC is not a part of this project.

This objective is the first step toward developing a phenomena identification and ranking table (PIRT) for liquid-metal cooled reactors. For a PIRT, subject matter experts are brought together to define phenomena and rank their importance in modeling specific events for a particular reactor design of interest. The level of understanding (i.e., knowledge) is also obtained from the panel of experts so that any future research can focus on the most important phenomena with the least amount of knowledge. This report will provide input to the panel that may be convened when more details are available on a design that NRC will be asked to review.

1.3 Outline of Report

Chapter 2 describes the liquid metal fast spectrum power reactors of interest to the NRC currently. Chapter 3 discusses licensing-basis events that are expected to be simulated with computer tools and Chapter 4 discusses the physical processes that must be modeled in order to have a viable simulation capability. Tables of phenomena are found in Chapter 4 based on systems expected to be present in a typical liquid metal reactor. Tables of phenomena for a similar generic sodium fast reactor are also found in the Appendix where phenomena are grouped in correspondence with postulated anticipated operational occurrences and accidents. References are found at the end of each chapter.

1.4 References

1-1 “NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness,” 2016.

1-2 “NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy – Staff Report: Near Term Implementation Action Plans,” Volume 1, Executive Information and Volume 2, Detailed Information, (ADAMS Accession No. ML16334A495), 2016.

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2 DESCRIPTION OF FAST SPECTRUM REACTORS

2.1 General Design Features

The liquid metal fast spectrum reactors (LMRs) that are to be considered herein are those designs for generating electricity by vendors who expect to bring these designs to the NRC for review in the near future. The reactors use solid fuel and a variety of coolants – sodium, lead, and lead-bismuth eutectic (LBE). The designs range in size from a few 10s of MWt, up to the range of current commercial power reactors (e.g., one design proposes a prototype of 600 MWe).[b] The key to maintaining a fast spectrum is to avoid material in the fuel, coolant and structures that can effectively thermalize the fission neutrons. Liquid metals are “heavy” so that a neutron collision does not significantly degrade its energy. Fuel forms are primarily oxide or metallic with a preference for the latter because it results in a harder neutron spectrum with attendant performance and safety benefits. Nitride fuel has also received some attention due to its thermo-physical properties.

The reactors with liquid metal coolant typically operate at or near atmospheric pressure, and are of two general types – loop, and pool. They usually include intermediate loops with intermediate heat exchangers (IHXs) which separate the primary system coolant from the power conversion system. In the loop system the primary pumps and IHXs are located in compartments separated from the reactor vessel with interconnecting piping. In the pool system the entire primary system (i.e., reactor, primary pumps, and IHXs) is located in a large liquid-metal pool in the reactor vessel. Compatibility of primary and secondary system must be considered, for example, sodium in the primary system would react with water in a typical power conversion system. The liquid metal coolants are all opaque, and this introduces challenges for fuel loading/unloading and shuffling, as well as in-service inspection.

In contrast with commercial light-water cooled reactors, LMRs tend to have high neutron leakage (on the order of 15%) which is one of the characteristics affecting design and behavior. A significant component of negative reactivity feedback is thermal expansion/bowing of the core. Most uranium fueled, sodium-cooled LMRs have a positive void coefficient which can be partially mitigated by design features or the use of thorium-based fuels/blankets. LMRs can be configured as burners, break-even, or breeders thereby serving several roles in the fuel cycle. The higher fission-to-capture reaction rates in a fast spectrum reduce the production of higher actinides, which is generally desirable for repository performance and is more effective as a “burner” of transuranics (TRU). Blankets with fertile material are usually required for breeding, which in a fast spectrum can result in breeding/conversion ratios significantly higher than 1.0.

In the following sections, the design features that are known for some proposed LMRs are explained. The emphasis is on those designs with specific vendors, although it is

[b] Small nuclear “batteries” were not considered in this project.

recognized that some of these designs are also being supported by work at universities and national laboratories in this country. It is also recognized that there are related designs being worked on in research centers in the U.S. and around the globe. The work being done on those other designs, not emphasized herein, is very important and is discussed in this report when applicable.

Five proposed LMR designs are listed in Table 2-1 with key parameters to help understand basic design and how accident scenarios might progress. The features given in the table come from a variety of sources as discussed below. Each potential vendor has a website but the extent of useful information in those websites and corresponding references varies in level of detail from one design to the next. These designs range from very preliminary, to quite mature (i.e., having had some interactions with the NRC). However, more important than the specifics of the design are the common features that must be modeled in any future accident simulations.

Note that the concepts employ both metallic and ceramic fuels, once-through and recycle fuel cycles, and low-enriched uranium and plutonium/TRU for the fissile components of the fuel. In some cases, a particular concept may envision the use of different fuel forms and/or fissile materials at various stages in their evolution/deployment. The characteristics of the fuel will have an impact on the performance of the reactor and its response to transients/accidents, and must be considered in the discussion of phenomena that need to be modeled. For example, plutonium has a significantly lower delayed neutron fraction than uranium, which impacts transient response, and introduction of minor actinides deteriorates the Doppler and coolant/void negative reactivity coefficients/feedbacks. In addition, LMRs can be configured as breeders (conversion ratio, CR > 1), breakeven (CR ~ 1), or burners (CR <1) depending on their role in the nuclear fuel cycle which also impacts reactor performance and safety characteristics.

Sections 2.2-2.6 provide brief descriptions of these five LMRs taken from a number of sources, including references [2-1] and [2-2]. For each reactor there are several concept-specific references listed in Section 2.7 that provide additional information; however, it must be stressed that some of these concepts are evolving and hence the descriptions are snapshots of their current status.

Table 2-1 Fast Spectrum Reactor Concepts

Organization	Advanced Reactor Concepts	GE-Hitachi	Gen4 Energy Hyperion Power	TerraPower	Westinghouse
Reactor Name	ARC-100	PRISM/S-PRISM	G4M	TWR	DLFR*
Power	260 MWt	840/1000 MWt	70 MWt	600 MWe (prototype)	-
Fuel	LEU-10Zr U-TRU-Zr U-Pu-Zr	U-TRU-Zr U-Pu-Zr	UN	LEU-Zr (start) DU/NU-Zr (feed)	Annular pellets UO_2 (start) / UN (later?)
Coolant	Sodium	Sodium	Lead-bismuth	Sodium	Lead
Reprocessing	Recycle	Recycle	Once-through	Once-through	Once-through/ Recycle
Reactivity Control	Control Rods Feedback Leakage/ expansion	Control Rods Feedback Leakage/ expansion	Control Rods Feedback Leakage/ expansion	Control Rods Feedback Leakage/ expansion	Control Rods Feedback Leakage/expansion
Website	arcnuclear.com	gehitachiprism.com	gen4energy.com	terrapower.com	westinghousenuclear.com/new-plants/lead-cooled-fast-reactor

*Demonstration LFR
LEU – low-enriched uranium
DU – depleted uranium
NU – natural uranium
UN – uranium nitride

2.2 Advanced Reactor Concepts (ARC-100)

The ARC-100 [2-3, 2-4, 2-5, 2-6] is being commercialized by Advanced Reactor Concepts (ARC) LLC, a startup company incorporated in the fall of 2006. A schematic of the reactor is shown in Figure 2-1.

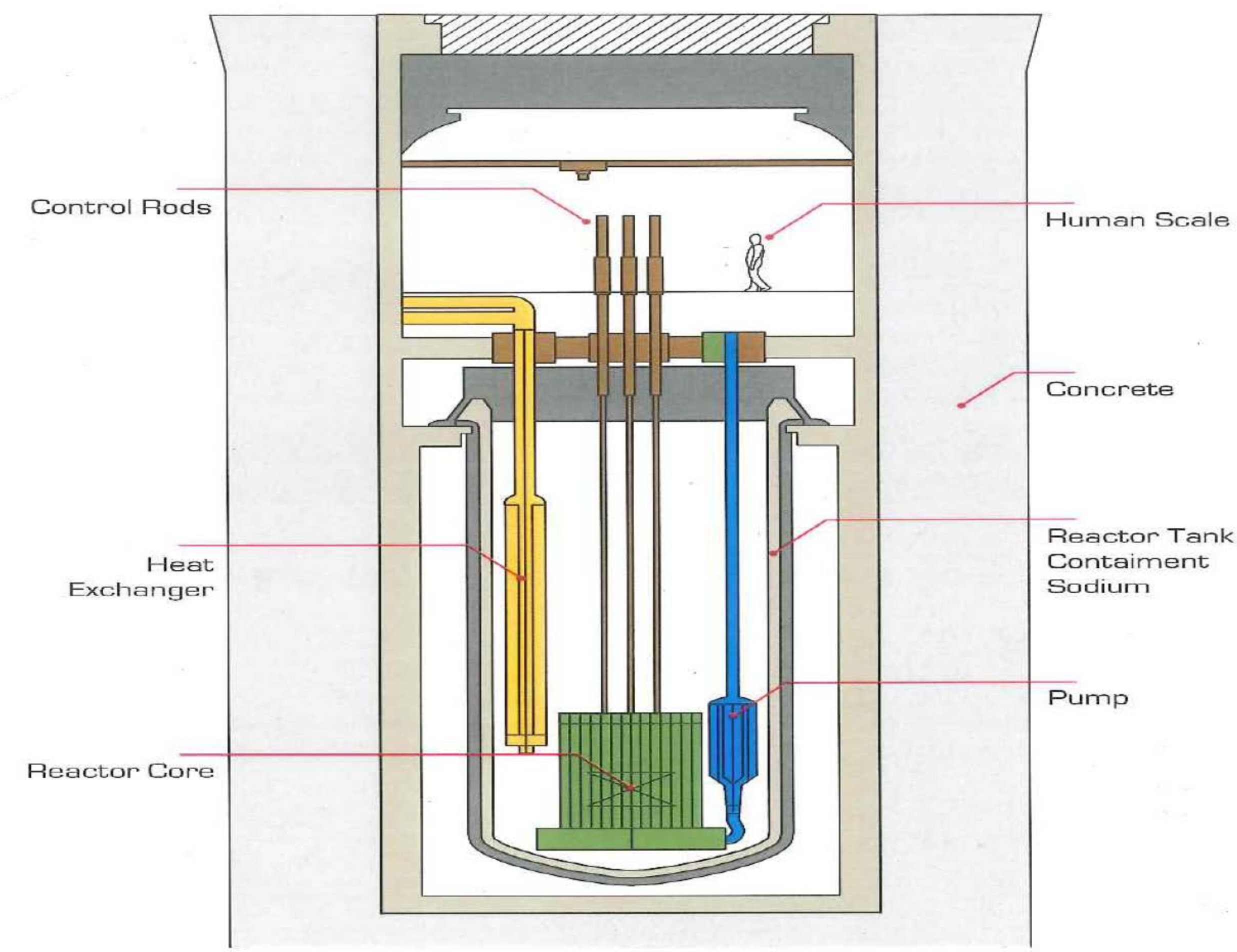

Figure 2-1 ARC-100 Reactor

ARC-100 is a 100 MWe (260 MWt), sodium-cooled fast reactor (SFR) targeting a long whole-core refueling interval (20+ years). Its initial fuel load is a low-enriched uranium metallic alloy (U-Zr) fuel slug in sodium bonded to ferritic-martensitic cladding. The reactor exhibits an internal breeding ratio near unity such that its reactivity burnup swing is small and its core is fissile self-sufficient. It attains 80 MWd/kg fuel average burnup with 13.5% enriched uranium, and upon pyro-metallurgical recycle at completion of its more than 20-year burn cycle, depleted uranium makeup feedstock is all that is required for the reload core. Upon multiple recycles, the core composition gradually shifts to an equilibrium transuranic fuel composition, which is also fissile self-sufficient–requiring only natural or depleted uranium makeup.

The forced circulation coolant delivers heat at ~500°C through a sodium intermediate loop that drives a supercritical CO_2 Brayton Cycle power converter attaining ~40% conversion efficiency and is capable of incorporating bottoming cycles for desalination, district heat, etc. The plant is sized to permit factory fabrication of rail and barge shippable modules for rapid assembly at the site. Its features are targeted to meet the infrastructure and institutional needs of rapidly growing cities in the developing world as well as nonelectric industrial and municipal niche applications in all nations.

2.3 GE-Hitachi (PRISM)

PRISM [2-7 to 2-12], Power Reactor Innovative Small Module, is a small modular SFR operating at 840 MWt (311 MWe) per each unit. An artist's rendering of the plant is shown in Figure 2-1. The reactor development was initiated in the 1980s as part of the U.S. Advanced Liquid Metal Reactor (ALMR) program directed by the U.S. Department of Energy (DOE). Under the Global Nuclear Energy Partnership (GNEP) program, GE-Hitachi (GEH) proposed the Advanced Recycling Center as a commercial solution for nuclear recycling. The Center consists of six PRISMs to generate a total of 1866 MWe and a single Nuclear Fuel Recycling Center NFRC, which supplies the fuel for the six PRISMs by processing light water reactor used nuclear fuel (UNF), PRISM UNF, and weapons-grade materials, depending on demands. For a break-even core, the discharge burnup is ~106 GWd/t with ~21% TRU content in the fresh fuel.

Based on the processing technologies developed by the Integral Fast Reactor program (another DOE program), metallic fuel is used and the uranium and transuranic material is continuously recycled via electrometallurgical processing (pyroprocessing). Two PRISMs are paired together in a power block to supply a 622 MW steam turbine.

PRISM has been developed with two different thermal power ratings targeting different shipping methods: 425 MWt (Mod A) for rail transportation and 840 MWt (Mod B) for barge transportation. In addition, GE has developed a 1,000 MWt power reactor to assess the technical viability and economic potential of a follow-on fast reactor called Super PRISM (S-PRISM). Currently, the PRISM considered by GEH is the Mod-B with a power rating of 840 MWt.

A Preliminary Safety Information Document (PSID) was submitted by DOE in 1986 and the NRC issued a Preapplication Safety Evaluation Report to document its review in 1994 [2-12]. Currently, PRISM is also being considered for use in the United Kingdom [2-11].

More details on the PRISM concept are found in Chapters 3 and 4 where the reactor is used as the basis for identifying phenomena that need to be modeled in the simulation tools of interest.

1. Steam Generator
2. Reactor Vessel Auxiliary Cooling System (RVACS) Stacks (8)
3. Refueling Enclosure Building
4. Vessel Liner
5. Reactor Protection System Modules
6. Electrical Equipment Modules
7. Seismic Isolation Bearing
8. Reactor Module (2), 311 MWe Each
9. Primary Electromagnetic Pump (4 per module)
10. Reactor Core
11. Intermediate Heat Exchangers (2)
12. Lower Containment Vessel
13. Upper Containment Building
14. Sodium Dump Tank
15. Intermediate Heat Transfer System
16. Steam Outlet Piping to Turbine
17. Feedwater Return Piping

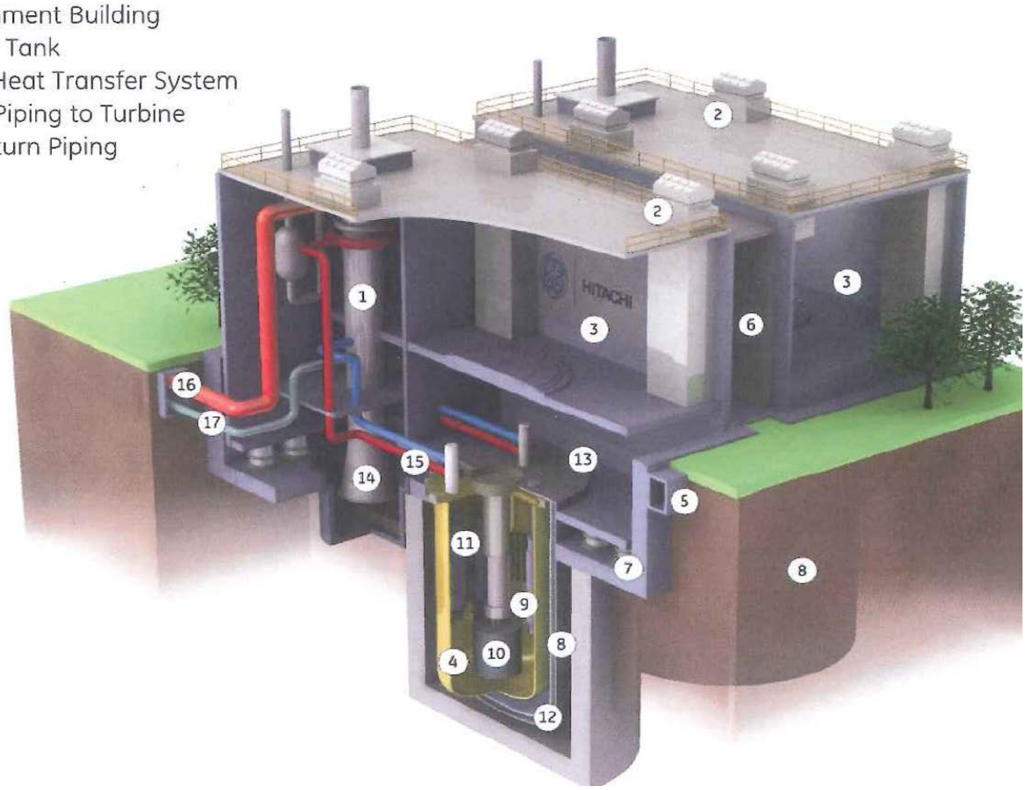

Figure 2-2 PRISM Power Block

2.4 Gen4 Energy (G4M)

Gen4Module (G4M) [2-13, 2-14, 2-15] is a lead-bismuth eutectic cooled fast reactor with 19.5% enriched uranium-nitride fuel. In 2007, G4M was introduced as the Hyperion Power Module (HPM), but the reactor was renamed G4M when the company was renamed Gen4Energy from Hyperion Power Generation Inc. A conceptual drawing is shown in Figure 2-3.

The G4M was designed to deliver 70 MW of heat or 25 MW of electricity for 10 years without refueling. After 10 years, the entire reactor module is replaced. The reactor is to be sited in an underground containment vault, and factory assembly allows for standardized design, potentially faster constructions and on-site deployment.

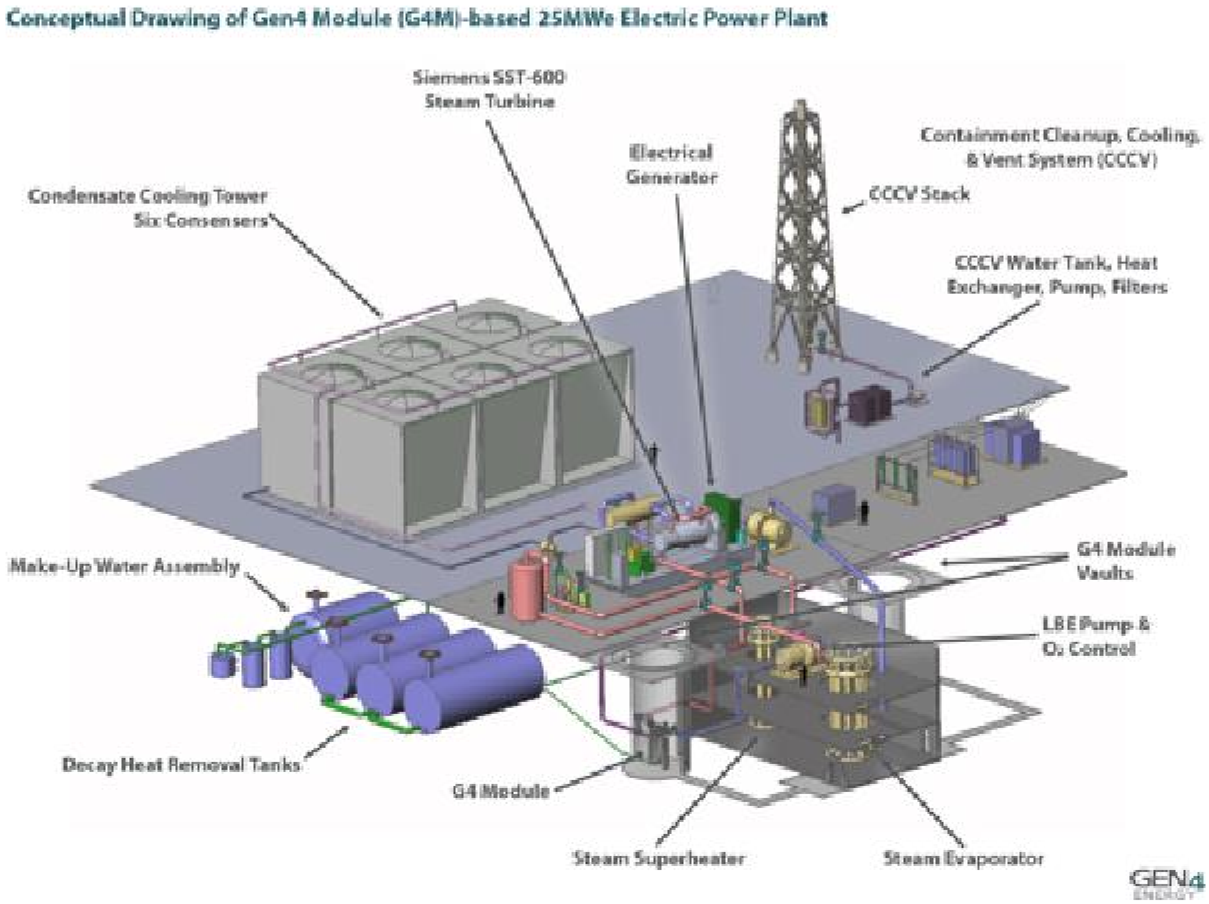

Figure 2-3 Conceptual Drawing of G4M Plant

2.5 TerraPower (TWR)

The Traveling Wave Reactor (TWR) [2-16 to 2-21], designed by TerraPower, is a sodium-cooled fast reactor designed to enable high burnup (20–50%) based on a breed-and-burn or nuclear-burning-wave mode of operation. It is designed to require no fuel reprocessing, use depleted or natural uranium as its primary fuel, require only a small amount of enriched uranium at start-up to ignite the reactor and never need *external* refueling. A diagram of the reactor is shown in Figure 2-4.

The TWR utilizes existing fast reactor technologies, but will adopt some innovative concepts in order to achieve very high burnup, for example, low smear density metallic fuel in advanced ferritic-martensitic steel cladding, fission gas vented fuel, and passive absorber insertion module.

It has been stated that TerraPower is committed to the near-term deployment of TWR technologies, starting a 600 MWe prototype traveling-wave reactor (TWR-P) deployment in the mid-2020s, followed by global commercial plant deployment. The TWR-P is planned to be utilized for irradiation tests of their innovative metallic fuels and demonstration of advanced safety concepts. Based on the current version of design concepts, TWR-P uses ~16% enriched uranium-zirconium fuel to achieve about 10% burnup with cycle length of 495 effective full power days.

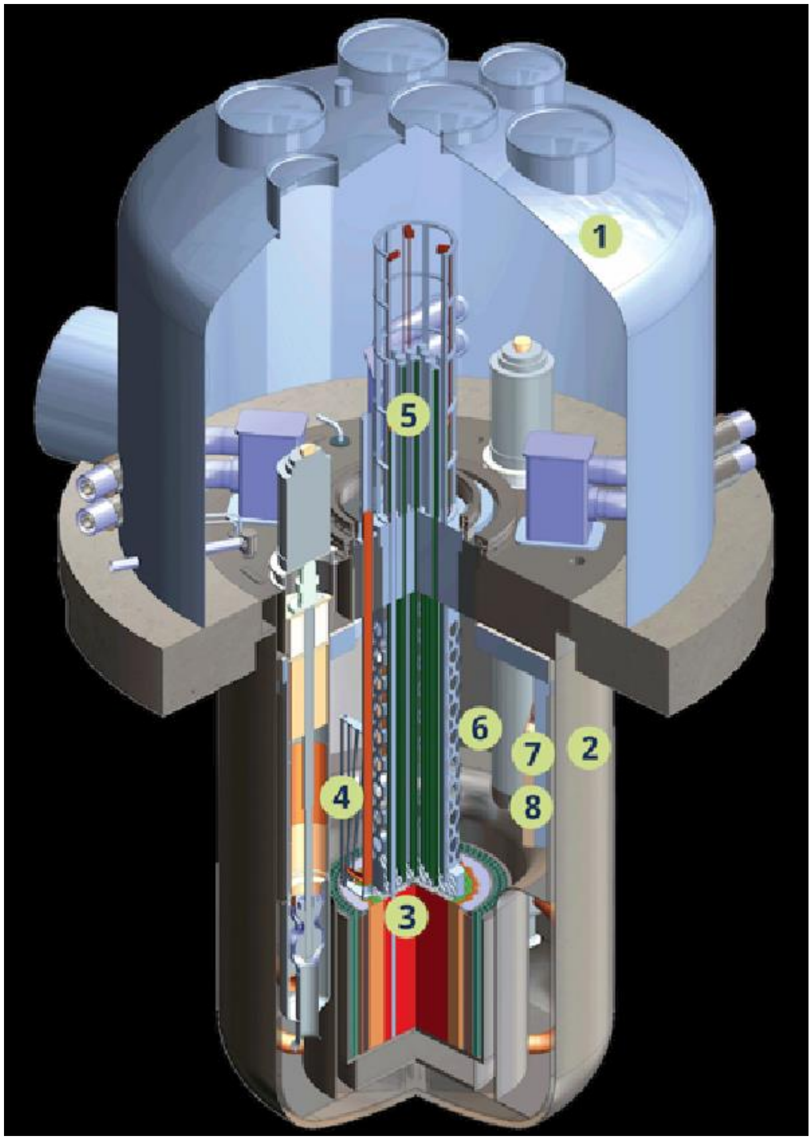

Figure 2-4 TWR Reactor

2.6 Westinghouse (W-LFR)

Westinghouse is developing a lead-cooled fast reactor (LFR) [2-22 to 2-26]. The demonstration LFR (DLFR) is shown in Figure 2-5. The benefits of lead cooling over sodium cooling are its high boiling point and lack of chemical activity; specifically, it does not react with water. The latter allows additional flexibility in the technology for steam generators and balance of plant. It also means that water can be used as a coolant in an emergency. The coolant high boiling point means that the formation of large voids is less likely than with sodium and less likely to have a significant reactivity insertion from void reactivity which may still be positive as in sodium-cooled fast reactors. Lead also absorbs and immobilizes fission products resulting in a reduced source term in the event of fuel failure.

There is no civilian operating experience with LFRs (there were Soviet naval LBE-cooled reactors [2-27]). However, there is a planned Russian BREST-300 reactor that is an LFR with nitride fuel (similar to the W-LFR). In the description of the DLFR, Westinghouse cites design and technology developments for the Advanced Lead-Cooled Fast Reactor European Demonstrator (ALFRED) as support for the design and safety characteristics of their concept as well as for the relative technological maturity.

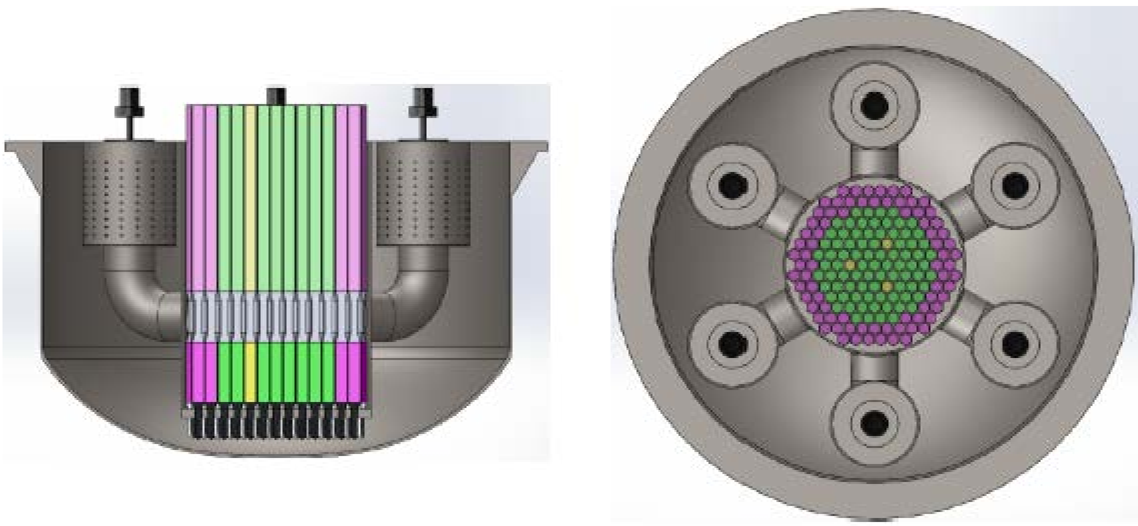

Figure 2-5 DLFR Primary System Layout

2.7 References

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3 SIMULATION SCENARIOS

3.1 Introduction

For any reactor concept the events that need to be simulated include normal operation, anticipated operational occurrences (AOO), design-basis events (DBE), and potentially some beyond-design-basis events (BDBE). Collectively, these can be considered the licensing-basis events and they are the categories considered by the NRC for current reactors, and will need to be addressed for all advanced reactors. Several of the LMR concepts described in the previous chapter are at an early stage of development and the proponents have not devoted a great deal of effort to safety related issues, focusing instead on normal operation and desirable economic and fuel cycle performance.

The sodium-cooled systems tend to be significantly more mature than the lead or lead-bismuth designs, and share many similar characteristics. The most mature concept is the sodium fast reactor PRISM, which was developed during the advanced liquid metal reactor (ALMR) program in the 1980s and produced a comprehensive Preliminary Safety Information Document (PSID) [3-1]. This document was reviewed by the NRC [3-2]. Subsequently, in 2011-2012 under the Global Nuclear Energy Program (GNEP), the SFR design evolved further. In connection with the GNEP program, several “expert groups” were formed to assess the status of SFR technology to support the planned GNEP mission. The results of these assessments were collected in a series of Department of Energy (DOE) reports [3-3, 3-4, 3-5].

The TWR and ARC-100 are both SFRs although with differences from PRISM. The lead (W-LFR) and lead-bismuth (G4M) cooled concepts share some of the characteristics of the sodium-cooled systems, but with some notable differences as far as safety is concerned, e.g., these coolants do not react with air or water. Although all designs need to be considered, the emphasis in the following sections relies heavily on the PRISM PSID [3-1], and the SFR expert assessments [3-3, 3-4, 3-5] that were done for a generic SFR.

3.2 Sodium Fast Reactors

The Table of Contents for the “Chapter 15 Accident Analysis” portion of the PRISM PSID is shown in Table 3-1. The basic approach to safety consists of three “levels.”

- inherent and basic design characteristics
- protection against anticipated and unlikely events
- protection against extremely unlikely events

The decision on which specific accident scenarios/events are evaluated in detail is based on extensive Probabilistic Risk Analysis (PRA), which considers the spectrum from low probability to high probability events. More detail is then analyzed for those events that can be classified as design-basis events or beyond-design-basis events except for those with extremely low probability of occurrence. Table 3-2 defines the characteristics of the “levels” considered above.

Table 3-1 Table of Contents for Chapter 15 - Accident Analysis [3-1])

Table 3-2 Event Categories and Definitions

Event Category	Definition
Normal operation	Any condition of system startup, design range operations, hot standby or shutdown.
Anticipated event	An off-normal condition which individually may be expected to occur once or more during the plant's lifetime.
Unlikely event	An off-normal condition which individually is not expected to occur during the plant's lifetime; however, when integrated over all plant components, events in this category may be expected to occur a number of times.
Extremely unlikely event	An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant's lifetime, but which nevertheless represents extreme or limiting cases of failure which are identified as design bases.
Beyond design-basis event	Off-normal conditions of such extremely low probability that no events in this category are credible during the plant's lifetime, but which have such extreme consequences that the risk (probability times consequence) from these events merits their consideration in establishing the design.

As shown in Table 3-1, two "classes" of Design Basis Events (DBEs) are considered: (1) Reactivity Insertion; and (2) Undercooling. In each category, an explicit "bounding event" is considered: "Uncontrolled Rod Withdrawal at 100% Power," and "Loss of Normal Shutdown Cooling," respectively. Other than these explicitly identified DBEs that were analyzed, and brief discussions of the radiological consequences of Sodium Spills, Fuel Handling and Storage Accidents, and Release of Cover Gas, the remainder of the safety analysis that was considered was "Local Fault Tolerance," where "local faults" are defined as: "…fuel failures that result from heat removal imbalance within a single assembly."

The discussion in the PSID indicates that the PRISM design relies heavily on the ability to detect fuel failures via cover gas and delayed neutron monitoring, and is based on the extensive experience with metallic fuel at EBR-II [3-6]. In addition, the consequences of increased heat generation due to enrichment errors or oversized fuel and reduced heat removal due to blockages are claimed to be minimal. The operating experience of EBR-II has also shown acceptable performance of metallic fuels in case of a cladding breach. Note that all three SFRs considered in this report assume the use of metallic fuel and rely on the experience data base from EBR-II and FFTF [3-7] to support licensing.

The approach followed in the PRISM PSID as described above and in Tables 3.1 and 3.2, does not necessarily reflect the approach that would be followed (or required) for licensing an SFR/LFR in the current regulatory environment. For example, if a similar approach is followed to that for licensing light water reactors (LWRs), a more detailed specification of accident scenarios would need to be developed and considered (e.g., Chapter 15 in [3-8]).

In order to do the gap analysis for SFRs that is documented in references [3-3, 3-4, 3-5], there is a categorization of accidents with similarities to the LWR approach. They consider three general categories:

- protected accidents
- unprotected accidents
- severe accidents with core melting

Within each category, three general types of upset conditions were considered:

- reduction or loss of core cooling,
- addition (or insertion) of reactivity to the reactor core, and
- reduction or loss of heat removal capability from the reactor

A significant aspect of the safety case for SFRs is based on analyses and demonstrations at EBR-II that a system can be designed that will accommodate Unprotected Loss of Flow (ULOF), Unprotected Transient Overpower (UTOP), and Unprotected Loss of Heat Sink (ULOHS) with no or only minor fuel damage.

3.3 Lead-Bismuth and Lead Fast Reactors

The previous section demonstrates the experience in the U.S. with SFRs. The two LMRs considered in this work (G4M and W-LFR) that employ lead or lead-bismuth eutectic (LBE) have benefited from work done primarily in Europe. Experience with reactors cooled by LBE is from Russian submarines. Lead-cooled reactors fueled with oxide or nitride fuel are being explored in the BREST and ALFRED projects discussed in Section 2.6. Based on the similar physical properties of lead-based coolants, it is expected that they will have similar impact on licensing basis events. [3-9]

The spectrum of accidents that will be considered for lead and LBE cooled LMRs will likely also be similar to those described for SFRs. However, the experience base is significantly smaller, especially when considering operation, and response to some of the unprotected transients in EBR-II that provide the confidence in the expected performance of SFRs with metallic fuel. The fuel forms for G4M and W-LFR also have limited experience.

However, when the unprotected transients considered for the SFR above–ULOF, UTOP and ULOHS–were analyzed for the ALFRED reactor, either there was no significant fuel damage, or the cladding failure time exceeded ten days.

3.4 Representative Sodium Fast Reactor Plant Systems

Among the systems described in Chapter 2, PRISM [3-1, 3-10] has the most comprehensive design information available in the open literature. This study thus adopts PRISM as the notional or representative system to facilitate the identification of phenomena important for the safety of sodium fast reactors. In addition, we consider the Toshiba Corporation small sodium fast reactor design, the 4S (Super-Safe, Small and Simple) [3-11]. As part of Toshiba's effort to obtain design approval from the NRC in the past, a Phenomena Identification and Ranking Table (PIRT) exercise was completed [3-12]. This PIRT is a useful reference for the current effort as the 4S has a plant layout that is similar to the PRISM design.

The brief discussion of the PRISM plant system is based on the information presented in [3-10]. A schematic of the PRISM nuclear steam supply system (NSSS) is shown in Figure 3-1. The primary heat transport system (PHTS) removes heat from the sodium coolant in the reactor vessel via intermediate heat exchangers (IHXs). An intermediate sodium loop removes heat from the IHX to the steam generator (part of the intermediate heat transport system (IHTS)). There are passive decay heat removal systems that rely on natural circulation air cooling to remove heat from the reactor vessel and the steam generator.

PRISM uses metallic fuel. Its core consists of a heterogeneous layout of multiple assembly types. These types include fuel, blanket, control, reflector, and shield assemblies. Fuel composition and assembly configuration vary based on the core mission. For the used nuclear fuel (UNF) recycling core mission the fuel is composed of a uranium-transuranic-zirconium alloy (U-TRU-Zr) with two fuel zones (see Figure 3-2).

Fuel rods are assembled by loading fuel slugs into cylindrical rods, backfilled with sodium (to improve heat transfer between the fuel and the cladding prior to fuel-cladding contact) and a small amount of argon-neon tag gas (unique blends of stable gas isotopes for tracing fuel failure to a specific fuel assembly). A large gas plenum volume ~1.5 times the length of the active core is included in each pin to accommodate fission gas release during operation. A typical fuel rod is shown in Figure 3-3. HT9 has been chosen for the cladding and much of the internal structures for its resistance to radiation damage from fast neutrons and strength.

Each PRISM fuel assembly consists of a hexagonal duct that surrounds the bundled fuel pins arranged in a hexagonal lattice. A spiral wire wrap around each pin maintains the pin spacing. Flow orificing is used to accommodate different heat generation rates among the fuel assembly types. A fuel assembly schematic is shown in Figure 3-4.

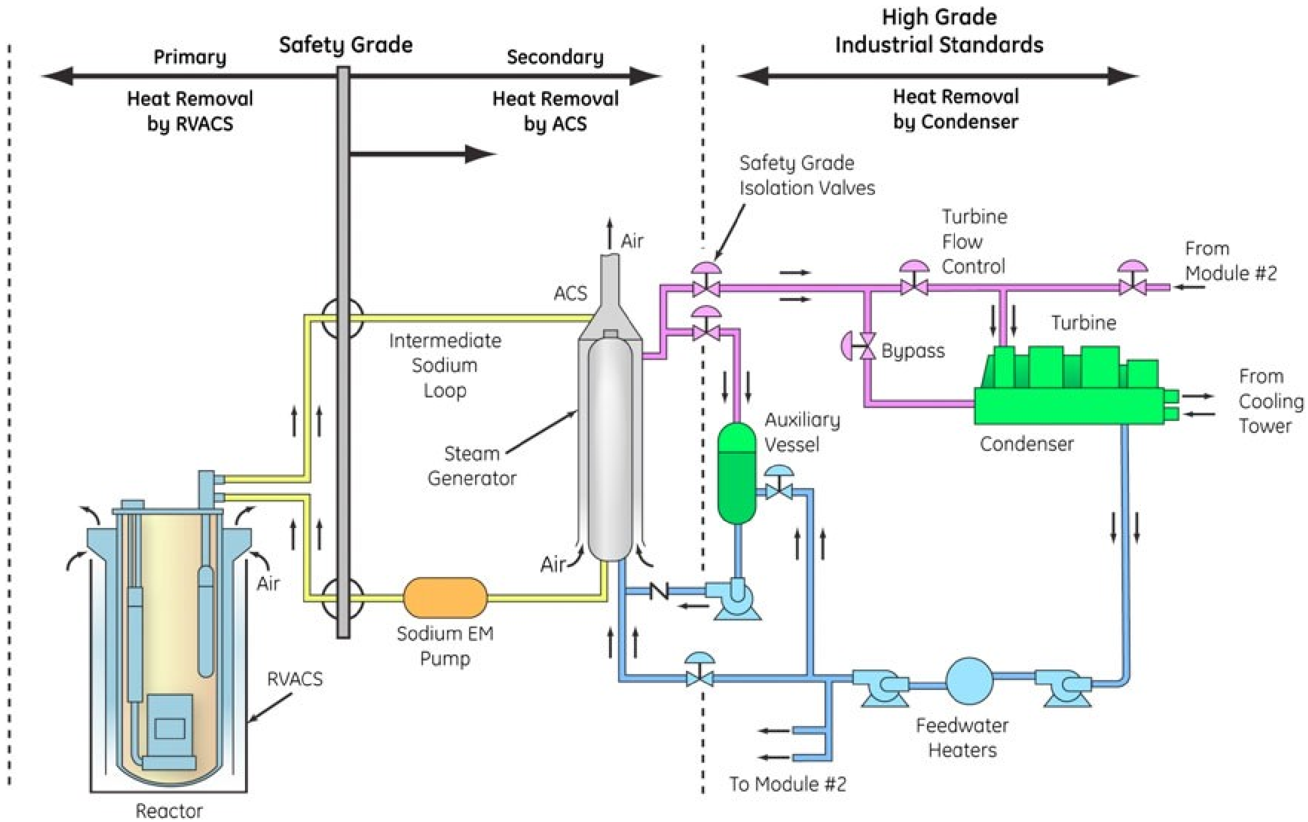

Figure 3-1 PRISM Nuclear Steam Supply System [3-10]

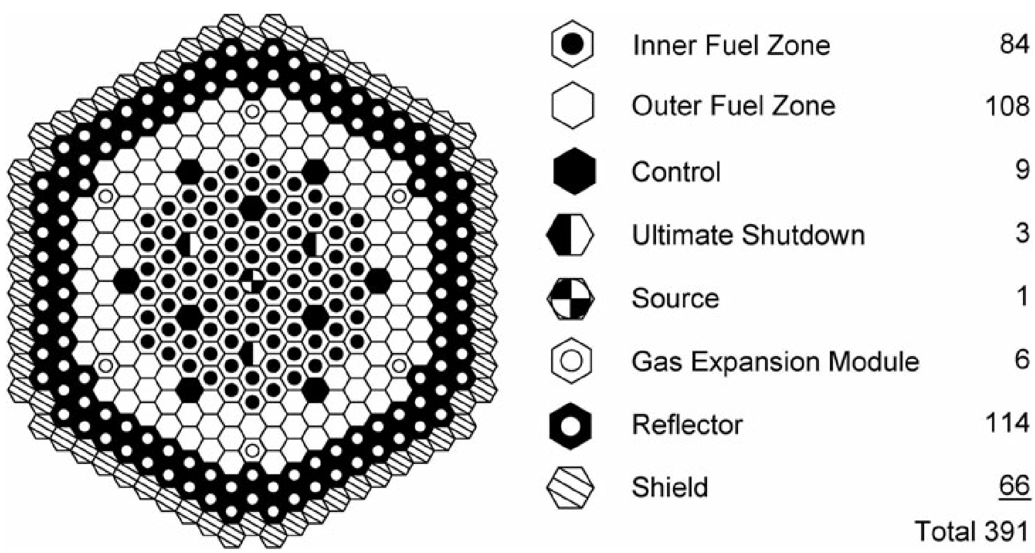

Figure 3-2 Fuel Assembly Layout for UNF Recycle Core [3-10]

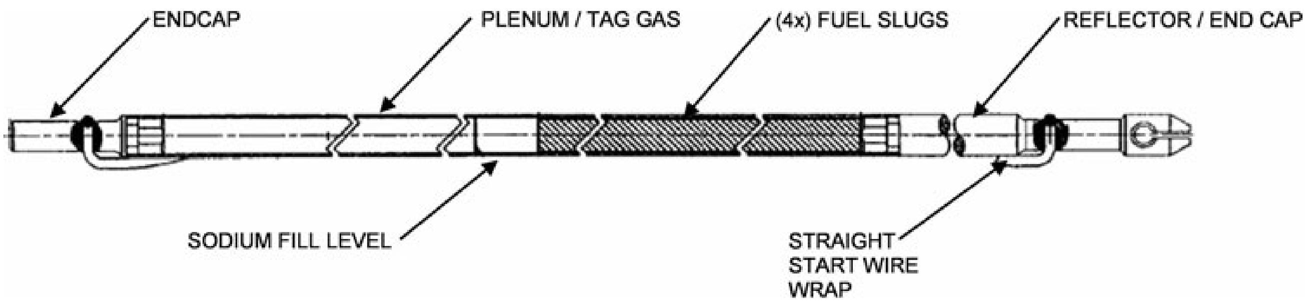

Figure 3-3 PRISM Fuel Rod [3-10]

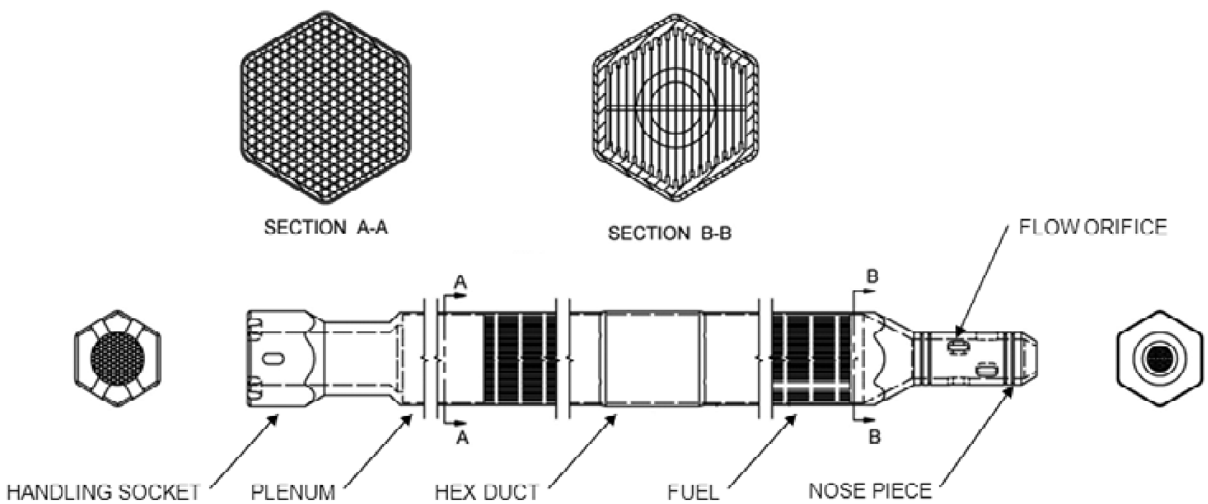

Figure 3-4 PRISM Fuel Assembly [3-10]

Reactivity control for normal startup operation, load following, and shutdown is accomplished with control assemblies consisting of B_4C absorber rods. The primary shutdown system (nine control assemblies) is backed up by an ultimate shutdown system (three ultimate shutdown assemblies). These control rods use magnetic latches, which can be actuated by either the reactor protection system or automatically when the latch temperature exceeds the magnetic Curie point temperature of the latch.

An overview of the structural components of the PRISM reactor module and containment building is shown in Figure 3-5. The major components making up the reactor module (see Figure 3-6) are the reactor vessel, reactor closure, containment vessel, reactor core and internal components. The reactor vessel is filled with liquid sodium and there is a helium cover gas at approximately atmospheric pressure at normal power conditions. A 20-cm gap filled with argon at a pressure slightly above the reactor cover gas sits between the reactor vessel and the containment vessel. Some of the major internal components are suspended from the reactor closure and they include two intermediate heat exchangers (IHXs), four electromagnetic pumps (EMPs), primary control rod drives, ultimate shutdown rod drives and in-vessel instrumentation.

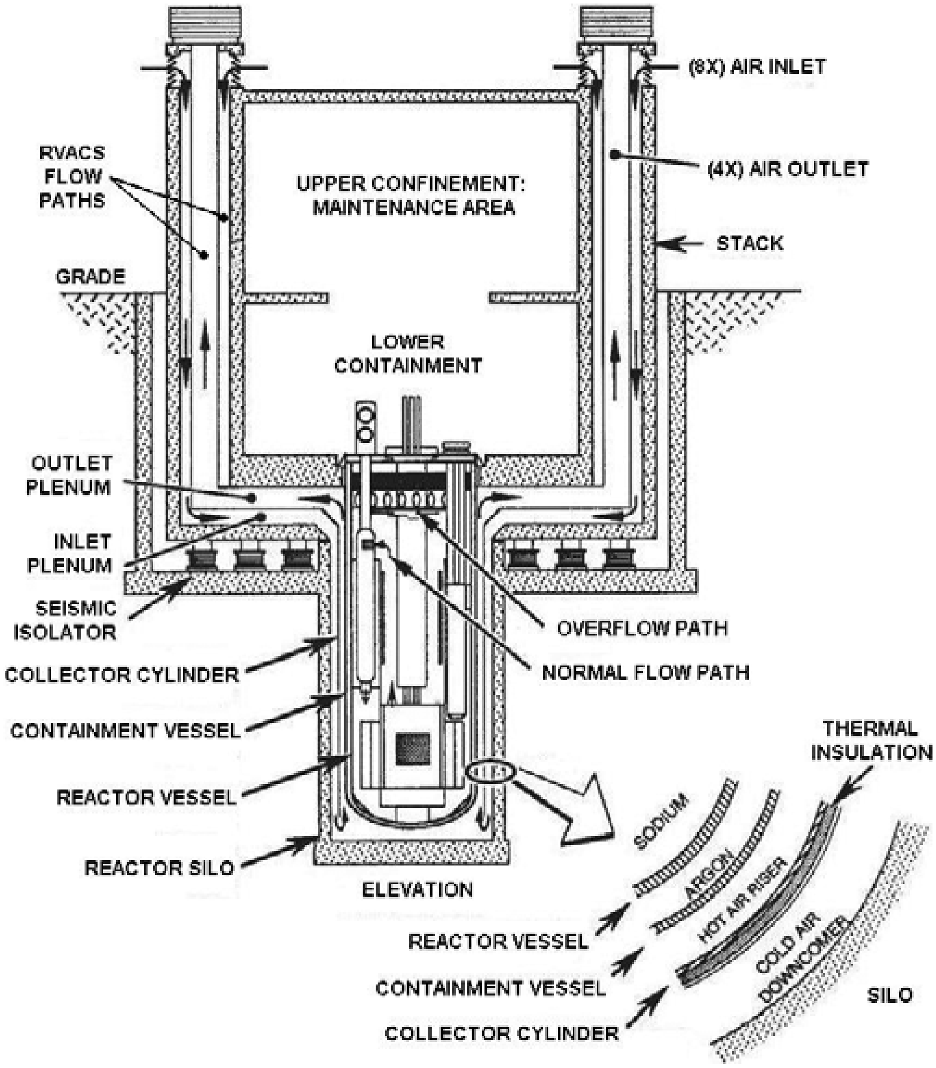

Figure 3-5 Structural Components of Reactor Module and Containment [3-10]

The PRISM containment consists of the containment vessel surrounding the reactor vessel and a lower containment over the reactor closure. The containment vessel is a leak tight stainless steel vessel with no penetrations. It is sized to retain all of the primary sodium leaked from the vessel in an accident while keeping the core, stored spent fuel, and IHX inlets covered with sodium. The lower containment is designed to provide a barrier in the event of closure breach during a hypothetical core disruptive accident (HCDA) and its volume is defined by the vessel closure head access area.

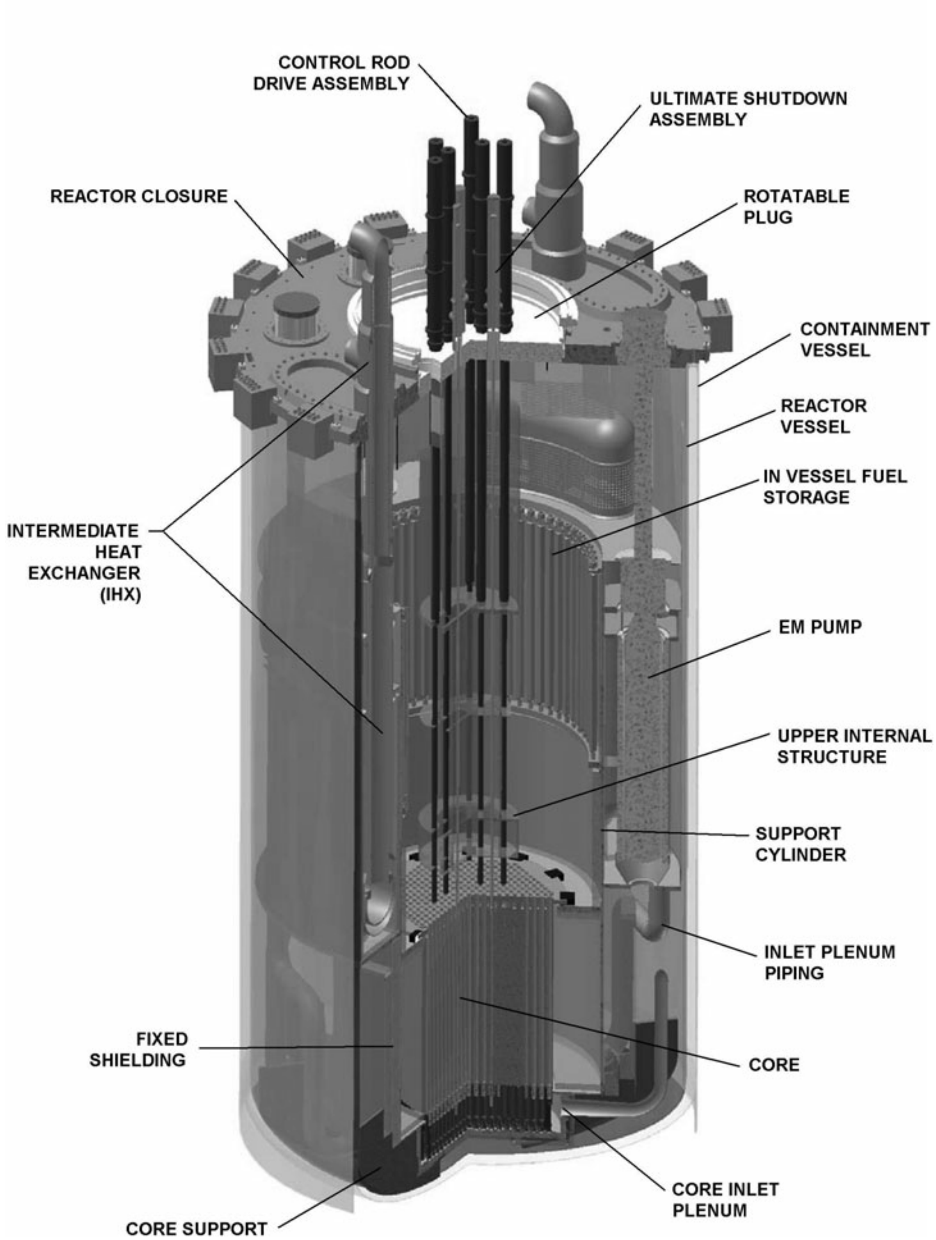

Figure 3-6 PRISM Reactor Module [3-10]

The primary heat transport system (PHTS) removes nuclear heat from the reactor core. The PHTS is entirely contained within the reactor vessel. The system carries the sodium coolant to flow through the reactor core, the hot pool (above the core), the tube side of the IHX, the cold pool, the EMPs, the pump discharge piping, and the core inlet plenum. Within each reactor module, four EMPs circulate the primary sodium through two IHXs. The primary coolant flow path is shown in Figure 3-7.

The intermediate heat transport system (IHTS) is a closed-loop system that carries the reactor heat to the steam generator (SG) system. The intermediate sodium is circulated through the shell side of the IHX and the shell side of the SG by two EMPs.

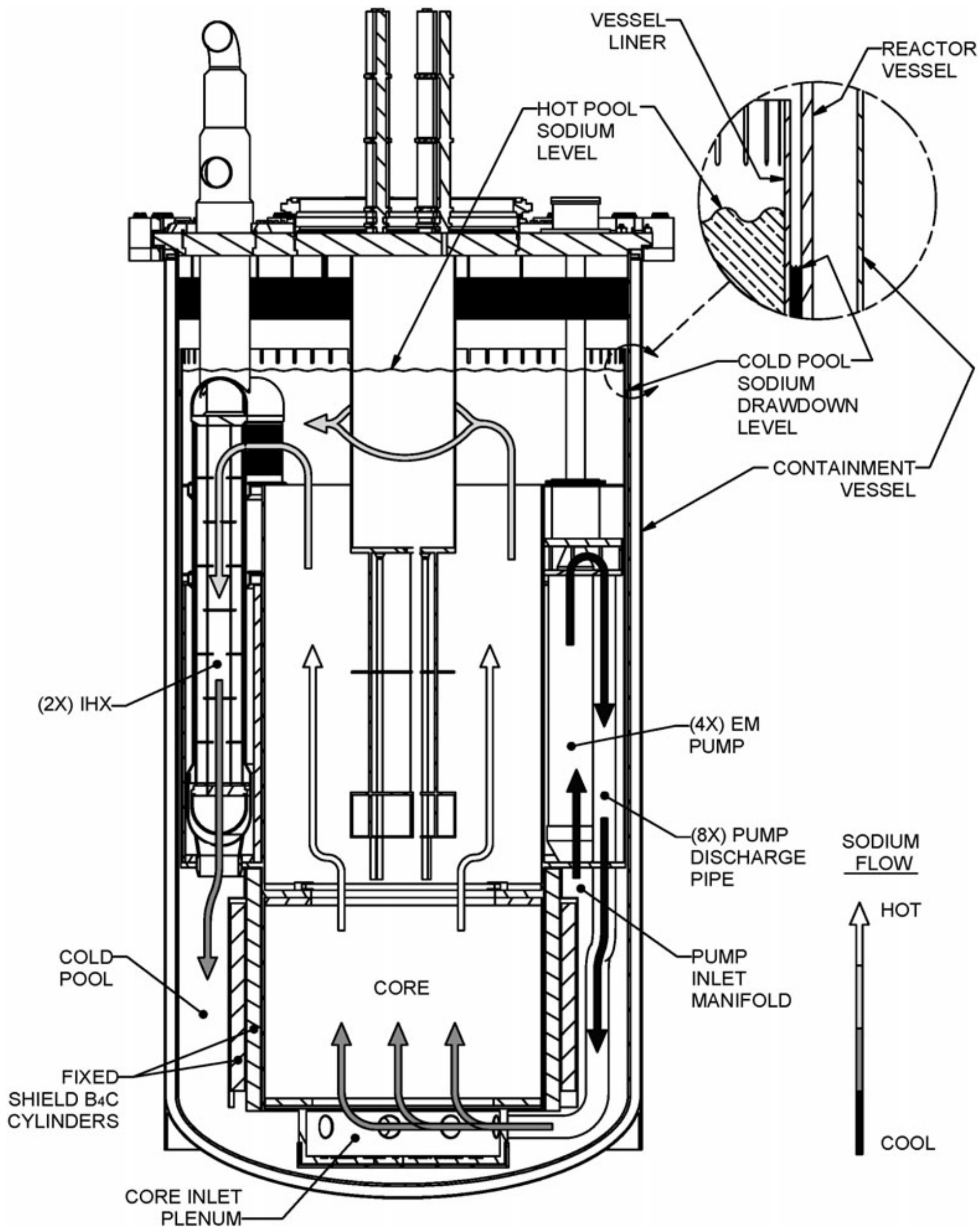

Figure 3-7 PRISM Primary Heat Transport System [3-10]

The steam generator is a vertically oriented, helical coil, sodium-to-water counter flow heat exchanger. Intermediate sodium flows down the shell side of the SG. Feedwater enters from the bottom and converts into superheated steam. The steam is used in a Rankine cycle to drive turbines to generate electricity.

Both active and passive systems are available to remove decay heat after shutdown of the reactor. During normal operation, reactor shutdown heat is removed by the turbine condenser using the turbine bypass. An intermediate reactor auxiliary cooling system (IRACS) provides an alternative method to remove shutdown decay heat during maintenance or repair operations. The IRACS uses natural or forced circulation of atmospheric air to remove heat from the outside shell of the SG. The IRACS (or ACS for short) is shown in Figure 3-8. The reactor vessel auxiliary cooling system (RVACS) is a passive system that relies on natural circulation air-cooling to remove heat from the reactor module (see Figure 3-5). Atmospheric air is drawn into the reactor building and flow over the outside of the containment vessel. The warm air then returns to the stack and is exhausted. The RVACS is always operating to remove heat from the containment vessel.

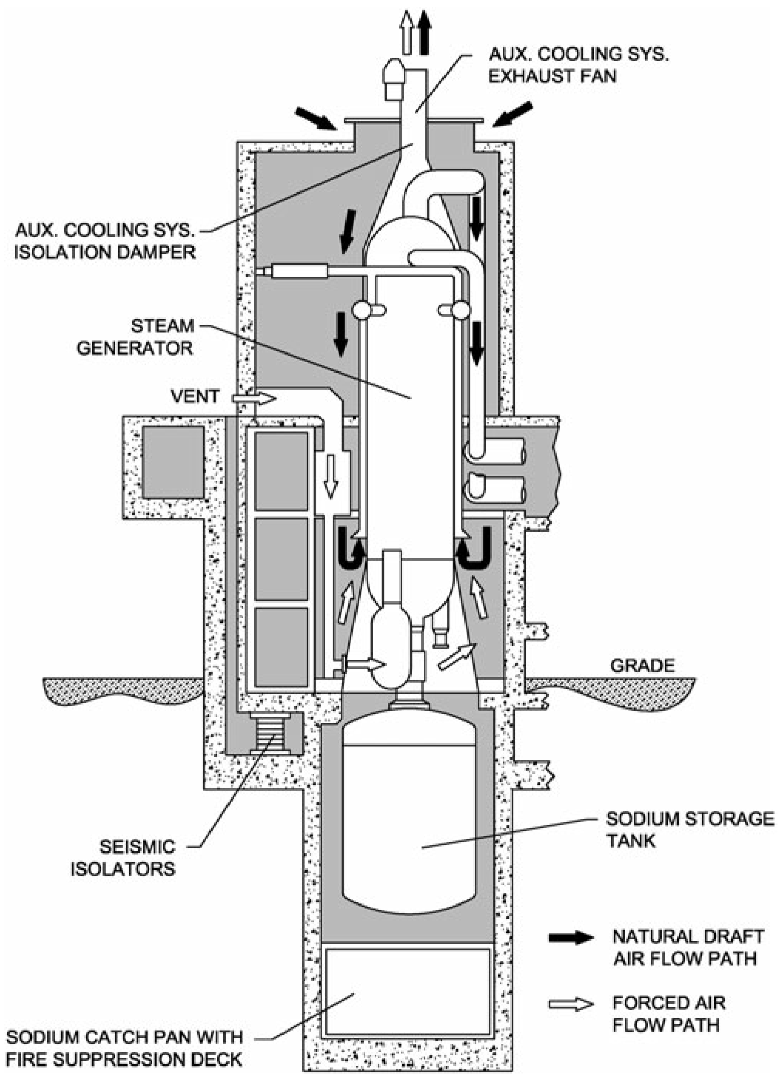

Figure 3-8 PRISM Auxiliary Cooling System [3-10]

3.4.1 Partitioning of Plant System

Following the 4S partitioning of the plant system [3-11], the pool-type, metal-fueled PRISM is divided into five subsystems for the purpose of identifying phenomena important to the safety of the reactor (in Chapter 4):

- core and fuel assemblies
- reactor system
- primary heat transport system (PHTS)
- intermediate heat transport system (IHTS)
- decay heat removal system

The five subsystems and their corresponding components have been described above. Table 3-3 provides a summary of the major components associated with each subsystem.

Table 3-3 Subsystems, Components, and Subcomponents

Subsystem	Component	Subcomponent
Core/fuel assemblies	Fuel Blanket Control elements Reflector Shield	Fuel rod, assembly hex duct Fuel slug+gas plenum+sodium-filled gap+cladding
Reactor	Reactor vessel	
	Reactor closure	
	Reactor internal structures	Core support Core barrel and support cylinder Inlet (lower) plenum Upper plenum Upper shroud
Primary heat transport system	Intermediate heat exchanger	Tube side Shell side
	Primary EMPs	
Intermediate heat transport system	Steam generator system	Tube side Shell side
	Intermediate EMPs	
Decay heat removal system	Intermediate reactor auxiliary cooling system (IRACS)	
	Reactor vessel auxiliary cooling system (RVACS)	
	Main condenser	

It is assumed in this study that the PRISM containment vessel and the lower containment are designed to accommodate primary coolant leakage and primary cover gas boundary leakage. Thus, for the purpose of identifying physical phenomena pertinent to the design-basis accidents for PRISM, events to confirm the adequacy of the containment system will not be considered. This approach is supported by noting that the integrity of the PRISM containment will not be challenged as long as the shutdown systems and the decay heat removal systems perform their normal functions.

Furthermore, this is consistent with one of the boundary conditions for this study which is to limit the scope of the simulations to be considered.

3.5 Event Definition

Generic SFR events and the relevant key systems/components are given in two tables below based on Reference [3-3]. Table 3-4 provides information for protected events and Table 3-5 the same information for unprotected anticipated transients without SCRAM. These tables were generated with the following philosophy [3-3]:

"Although it is possible that the potential accident phenomena could be identified and evaluated without consideration of accident sequences, the use of a general set of accident sequences is useful in understanding the relative importance of specific phenomena and how any given phenomenon relates to the safety performance of the reactor. For this purpose, three general categories of accidents have been defined:

- protected accidents - an accident initiator occurs, such as a component failure, failure of a safety grade system (other than the reactor protection systems), or an external event, followed by successful activation of the plant protection systems to shut down the reactor

- unprotected accidents – an accident initiator occurs as in the case for protected accidents, but the reactor protection systems fail to function. Such accidents may result in fuel damage, fuel melting, and fuel pin failures. For the purposes of these evaluations, accidents where fuel melting and fuel pin failures are widespread throughout the reactor core are treated in the severe accident category

- severe accidents with core melting – typically an unprotected accident where the failure of the reactor protection system results in conditions within the reactor such that widespread melting and failure of the reactor fuel occurs [not considered in this report]

Given these general categories, the three general types of upset conditions were considered, (a) reduction or loss of core cooling, (b) addition (or insertion) of reactivity to the reactor core, and (c) reduction or loss of heat removal capability from the reactor."

In support of the NRC review of the PRISM PSID [3-1, 3-2], several protected and unprotected accidents were analyzed [3-13], including three major unscrammed events, loss-of-heat sink (LOHS), loss of flow (LOF), and transient over power (TOP). Results of the analyses were similar to those submitted by the designer/vendor.

Table 3-4 Descriptions of Protected Events [3-3]

Event Description	Key Systems/Components
Loss of Core Cooling	
Equipment Failure: electrical faults loss of site power controller failures internal flow blockage mechanical faults pump mechanical failure loss of piping integrity *Operator Error:* turning off pump power opening breakers to power supplies *External Events:* earthquakes, fire, flood, tornado, terrorist	primary pump power supplies shaft/ bearing/ impeller off-site power connection primary piping and vessel system core and assembly coolant flow channels fuel cladding reactor control and protection systems shutdown heat removal systems reactor containment electrical-magnetic pump power leads
Reactivity Addition	
Equipment Failure: uncontrolled control-rod motion overcooling from pump speed increase balance of plant (BOP) system pressure loss gas bubble entrainment *Operator Error:* control-rod movement error coolant pump control error actuation of BOP pressure relief valve *External Events:* earthquakes	reactor control system and control rod drives primary pumps BOP heat removal systems shutdown heat removal primary and intermediate cooling systems reactor protection systems BOP control systems reactor containment
Loss of Normal Heat Rejection	
Equipment Failure: steam generator failure intermediate heat transport system failure supercritical CO_2 system failure loss of electric grid load flow blockage in heat transfer loop *Operator Error:* stopping intermediate loop flow steam generator blow down isolating plant from the grid *External Events:* earthquake, fire, flood, tornado, terrorist	secondary sodium pumps secondary system piping steam generators sodium-CO_2 heat exchanger turbine-generators shutdown heat removal systems intermediate heat exchanger reactor protection systems reactor containment

Table 3-5 Descriptions of Unprotected Events [3-3]

Event Description	Key Systems/Components
Loss of Core Cooling (ATWS)	
Reactor shutdown system failure following: electrical faults mechanical faults loss of site power loss of piping integrity internal flow blockage	primary pump power supplies pump mechanicals off-site power primary piping system core and assembly coolant flow channels core structure fuel and subassemblies primary coolant system Inherent and passive safety systems flow coast down extenders
Reactivity Addition (ATWS)	
Reactor shutdown system failure with: uncontrolled withdrawal of a single control rod overcooling from pump speed increase	reactor shutdown systems control rod drive system fuel and subassemblies primary pumps BOP heat rejection system
Loss of Normal Heat Rejection (ATWS)	
Reactor shutdown system failure with: steam generator failure intermediate heat transport failure supercritical CO_2 system failure decay heat removal system failure	secondary sodium pumps secondary system piping and intermediate heat exchangers steam generators decay heat removal systems Na-CO_2 heat exchanger

3.6 References

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4 PHYSICAL PROCESSES FOR LIQUID-METAL FAST REACTORS

4.1 Safety Characteristics of Sodium Fast Reactors

4.1.1 Thermal-Hydraulic Characteristics

From a thermal-hydraulics perspective, among the unique features of SFRs that have safety implications are their compact core size of relatively high power density and the use of low-pressure sodium as primary coolant. The following characteristics are relevant:

- Liquid metals such as sodium (and potassium) have relatively low melting temperature (to avoid having to preheat the system to obtain a liquid coolant) and high boiling temperature (or low vapor pressure, to avoid boiling). They remain in liquid form over a wide range of temperatures.
- A pool-type reactor coupled with a low-pressure primary system makes the occurrence of a large loss-of-coolant accident (LOCA) unlikely, however leaks through instrumentation penetrations need to be considered.
- Operating at low pressures, sodium will not completely flash on depressurization. (Light water reactors are subject to system rupture followed by coolant depressurization on loss of coolant.)
- For liquid metals, the Prandtl number, Pr, is less than 1 and the convective heat transfer coefficient (given in the Nusselt number, Nu) is a function of the Peclet number, Pe = RePr where Re is the Reynolds number.
- Generally, for Pe<100 heat transfer is dominated by heat conduction (Nu does not vary much with Pe) and is not affected by the coolant flow rate.
- The modest vertical elevation of the intermediate heat exchanger (IHX) relative to the core in a potential pool-type reactor design would make it less than ideal for establishing natural circulation flow in the primary system. Natural circulation depends on these elevations as well as hot and “cold” temperatures, and resistances in the loop.
- The relatively large mass of sodium in a pool-type reactor (versus a loop-type) provides large heat capacity to dampen temperature rise in off-normal transients but also influences the control and load following characteristics of the overall heat transport systems.
- No moderation in a fast spectrum core leads to a more compact core, that is, higher power density and higher specific power. This translates to more restrictive coolant flow passages and more severe heat removal requirements/constraints.

- High specific powers and power densities require large heat transfer areas and high heat transfer coefficients to be used to reduce fuel centerline and cladding temperatures and avoid melting.

- SFR fuel is typically in the form of small diameter tubes or fuel pins.

- Enhanced heat transfer from the fuel pins is facilitated by roughened cladding surfaces and turbulence promoters (e.g. wire-wrap spacers).

- A compact core makes core orificing more challenging but essential to counter the effects of power peaking.

- Testing performed at the EBR II [4-1] has demonstrated,
    - metal fueled fast reactors can be self-protecting against anticipated transients without scram (ATWS)
    - load-following control is manageable
    - passive transition to natural convective core cooling
    - passive rejection of decay heat

4.1.2 Neutronic Characteristics:

From a neutronics perspective, among the unique features of SFRs that have safety implications are their compact core size, operation with a fast neutron spectrum, and utilization of Pu and higher enriched uranium (relative to thermal reactors) in the fuel. The following characteristics are relevant:

- Fast fission cross sections are a few hundred times lower than for thermal fissions requiring a higher concentration of fissionable fuel in a fast spectrum core.

- Smaller loss by parasitic capture in fuel and lesser poisonous effects from fission products lead to the possibility of higher fuel burnup and lower excess reactivity requirements for SFRs.

- With high burnups, good fission gas retention or venting becomes a major consideration in SFR fuel system design.

- Fuel burnup in SFRs is usually limited not by reactivity but by radiation damage to the fuel pins (e.g., swelling).

- In general, the possibility of leakage of dense hydrogenous material into an SFR core must be avoided because of concern over prompt criticality brought on by positive reactivity associated with the softening of the neutron spectrum.

- SFRs generally have short prompt neutron lifetimes which may cause a large reactivity insertion rate in the event of a core disruption (core collapse) accident.

- In SFRs the effective delayed neutron fraction (β_{eff}) is impacted negatively by Pu-239 (β for Pu-239 is only 0.00215 compared to 0.0068 for U-235) and positively by fast fission of the fertile U-238 (β= 0.0158).

- SFRs are subject to positive void reactivity effects. The presence of void in the core reduces moderation (a positive reactivity effect) but also induces more neutron leakage (a negative reactivity effect). In small-sized SFRs, the effect of neutron leakage predominates over the effect of flux hardening. The situation is just the opposite for large-sized SFRs; significant flux hardening near core center and less predominant leakage near the core boundary. The general design philosophy for large-sized cores is to use special core design features, such as a pancake core with axial and radial blankets, to achieve a high-leakage core resulting in an overall negative sodium void reactivity.

- The presence of a harder neutron spectrum in metal-fueled SFRs leads to significantly smaller Doppler feedback than in ceramic-fueled reactors.

- Bowing of fuel assemblies due to radial temperature gradients across the core can lead to reactivity changes.

4.1.3 Additional Concerns

Two concerns dominated the early safety analyses for fast spectrum reactors in the U.S., core compaction—due to slumping or melting—and prompt criticality. The core compaction issue originates from the fact that fuel densification would increase the system reactivity in contrast to the effect in thermal reactors. The prompt criticality concern is related to the small effective delayed neutron fraction and the short prompt neutron lifetime of an LMR. It is postulated in a hypothetical core disruptive accident (HCDA) that a core meltdown may rapidly lead to a prompt criticality condition with an extremely rapid power increase creating a potential energy release of magnitude larger than practically containable. Subsequent improvements have been made to the analytical techniques, most of which demonstrate reduced consequences from HCDA. Furthermore, HCDA generally is not a concern for metallic fuels; one of the drivers for their use rather than oxide fuels.

4.2 Safety Characteristics of Lead-Bismuth and Lead Fast Reactors

4.2.1 Thermal-Hydraulic Characteristics

The basic physical properties of sodium, lead, and lead-bismuth summarized in Table 4-1 lead to different thermal-hydraulic characteristics. As liquid metals all three have similar benefits as reactor coolants relative to water or gas (e.g., low pressure and good heat removal). However, there are significant differences between the characteristics of lead and LBE coolants relative to the characteristics for sodium. These differences and other characteristics impact performance and safety:

- Significantly higher boiling temperature for Pb/LBE – less likely to boil/void in transients/accidents

- High melting temperature for Pb – may require a heater to prevent freezing
- Low chemical activity with water, steam, air, water vapor – may avoid need for intermediate loop; facilitates having additional sources for cooling in accidents
- Lower neutron moderation allows larger pitch which enhances natural circulation
- Radiation resistant, low activation; however, polonium buildup with LBE may plate out throughout the primary coolant circuit -- presents a handling concern.
- Higher volumetric heat capacity
- Higher retention of fission products in Pb
- Higher density has impact on seismic events
- Erosion limits flow velocity relative to Na (*v* in table)
- Compatibility with structural components requires control of coolant oxygen and preservation of oxide coatings to minimize damage
- As noted in Chapter 2, the principal experience with LBE has been in Russian nuclear powered submarines. The BREST-OD-300 reactor in Russia is intended as a demonstration of an LFR; there are also active studies of LFRs, for example the ALFRED project.

Table 4-1 Basic Physical Properties of Liquid Metal Coolants

Property	Na	Pb	Pb-Bi
ρ g/cm^3]	0.847	10.48	10.15
T_m [K]	371	601	398
T_b [K]	1156	2023	1943
c_p [kJ/(kg K]	1.3	0.15	0.15
ρc_p [J/m^3 K]	1.1 x10^6	1.6 x 10^6	1.5 x 10^6
k [W/(m K]	70	16	13
v [m/s]	10	2.5	2.5
Density (ρ), melting temperature (T_m), specific heat (c_p), thermal conductivity (k), and maximum velocity (v) are given at 700 K.			

4.2.2 Neutronic Characteristics

Lead and LBE result in similar neutron spectra, close to those for SFRs. The largest difference is in the void coefficient of reactivity which is generally less of an issue/concern than for a sodium fast reactor. As noted above, the higher mass of Pb/LBE relative to Na means lower neutron moderation and this couples to the thermal-hydraulics as it allows for a larger pitch which can enhance natural circulation. Lead/LBE requires higher initial fissile inventory but results in a lower burnup reactivity swing which has an impact on control requirements.

4.2.3 General Observations

As noted earlier, there is significant experience in the U.S. with SFRs. This includes development of designs, extensive methods development, safety analyses, and operational experience obtained over several years with several reactors, oxide and metallic fuel, and pool and loop configurations. In addition, there have been interactions with the NRC based on the development and preliminary review of a PSID for the PRISM reactor. The two LMRs considered in this work that employ lead or lead-bismuth eutectic have been explored primarily in Europe. Experience with reactors cooled by LBE is primarily from Russian submarines. Lead-cooled reactors fueled with oxide or nitride fuel are being explored in the BREST and ALFRED projects discussed in Section 2.6. Based on the similar physical properties of both of these coolants, it is expected that they will have similar impact on licensing basis events.

The spectrum of accidents that will be considered for lead and LBE cooled LMRs will likely be similar to those described for SFRs. However, the experience base is significantly smaller, especially when considering operation, and the response to some of the unprotected transients in EBR-II that provide the confidence in the expected performance of SFRs with metallic fuel. There is also limited experience with the fuel forms for G4M and W-LFR. Nevertheless, when the unprotected transients considered for the SFR above–ULOF, UTOP and ULOHS–were analyzed for the ALFRED reactor, either there was no significant fuel damage or the cladding failure time exceeded ten days.

4.3 Phenomena Identification

SFR phenomena that must be modeled in simulation tools for steady state and time dependent scenarios have been identified in previous studies [4-2, 4-3, 4-4]. The tables below identify the plausible phenomena, with definitions and/or related comments, for each of the subsystems defined in Section 3.4.1 for the PRISM reactor (see Table 3-3). Several generic phenomena (i.e. not directly linked to a subsystem) associated with the presence of liquid sodium in the fast reactor are also identified.

The phenomena identified in the tables below are generic without consideration of specific events that would need to be simulated. Similar tables, based on specific transient/accident events are found in the Appendix. The importance of each phenomenon depends on the specific event being considered. The importance is typically evaluated against a set of figures-of-merit (FoM) for a particular event, for example a regulatory acceptance criterion. An FoM might be a property in any of the systems or components of the reactor. For example, in the core an FoM might be the cladding temperature (to prevent cladding breach by stress rupture or to remain below eutectic temperature [4-5]), k_{eff} to stay below prompt criticality, or coolant void production.

Table 4-2 provides the phenomena for the core and fuel assemblies. This includes thermal-hydraulic, neutronic, and fuel performance phenomena. Table 4-3 has phenomena for the reactor system components the vessel and internal structures where the latter includes the upper and lower plena, vertical shroud, reflector and shields. Table 4-4 provides phenomena related to the primary heat transport system, which includes the intermediate heat exchanger (IHX) and primary electromagnetic pump (EMP). Table 4-5 does the same for the intermediate heat transport system, which includes the intermediate EMP and steam generator (SG) system and Table 4-6 for the decay heat removal system, which includes the main condenser, intermediate reactor auxiliary cooling system (IRACS) and reactor vessel auxiliary cooling system (RVACS). Lastly, Table 4-7 gives phenomena for the sodium coolant and some of them are related to phenomena outside of the vessel. The current lists of phenomena may need to be supplemented or modified when considering an LMR that is different from the representative SFR described in Section 3.4. An example is the adoption of supercritical CO_2 in the power conversion cycle, with the potential of not requiring an intermediate sodium loop.

Table 4-2 Phenomena in Core/Fuel Assemblies

Phenomenon	Definition and Related Comments
Pressure loss in core region	Flow rate is dependent on pressure loss. Forced flow and natural circulation may involve different closed loop flow paths and the fractional contribution of core pressure loss to the total loop pressure loss will be different for the two modes of core flow.
Inter-assembly flow distribution	Due to different heat generation rates among fuel, blanket, reflector, and shield assemblies, flow orificing is used to distribute proper flow to different assemblies. It affects pressure drop and parallel channel flow stability.
Intra-assembly flow distribution	Flow distribution inside an assembly depends on the bundle geometry. Under natural convection, the flow also depends on temperature distribution inside the assembly. Manufacturing tolerances contribute to uncertainty in flow and temperature predictions.
Natural convection	Under low flow or loss of forced flow, local natural convection can establish natural circulation through fuel assemblies, e.g. upflow in hotter assemblies and downflow in cooler assemblies (or core flow redistribution in transition).
Gap conductance between fuel and cladding	Sodium bonding reduces the temperature drop across the gap (relatively high gap conductance).
Heat transfer between cladding and coolant	For liquid metals, the Prandtl number, Pr, is less than 1 and the thermal boundary layer is substantially larger than the thickness of the hydrodynamic boundary layer. The heat transfer coefficient (given in the Nusselt number) is a function of the Peclet number Pe=RePr. Generally, for Pe<100 the heat transfer is dominated by heat conduction (Nu does not vary much with Pe) and is not affected by the coolant flow rate. Beside conduction the heat transfer from a fuel pin is affected by the presence of grid spacers or wire wraps. These protrusions augment the flow field by increasing turbulence through flow scattering downstream of grids and inducing azimuthal flow through flow sweeping caused by helical wire wraps or fins.
Inter-assembly heat transfer	Radial heat transfer between fuel assemblies tends to couple the assemblies thermally during low-flow natural convection conditions, i.e., heat transfer, temperature distribution and natural convection flow among the fuel assemblies are interdependent.
EMP coastdown	Inadequate pump coastdown performance (e.g. loss of one of the pumps) may cause Na boiling if the reactor failed to scram.
Radiation heat transfer from reactor vessel to containment vessel	This is the heat transfer mechanism to remove decay heat from the reactor vessel in the event of a loss of heat sink.
Natural air convection cooling of containment vessel	The reactor vessel auxiliary cooling system (RVACS) relies on natural circulation to cool the containment vessel.

Phenomenon	Definition and Related Comments
Natural air convection cooling of steam generator	The auxiliary cooling system (ACS) relies on natural circulation and forced air (flow over the outside shell of the SG) to provide an alternative method for shutdown decay heat removal.
Natural circulation cooling of primary coolant	These are part of auxiliary systems engineered to remove decay heat after reactor shutdown by passive means.
Heat capacity of core assemblies	The mass and specific heat of materials in the core influence the rate of temperature changes of the core and fuel assemblies during transients and accidents. Stored energy in core assemblies is a source of energy after reactor shutdown.
Coolant boiling	Boiling temperature of Na is 881ºC at 0.1 MPa and rises with pressure.
Fission heat	Most of the nuclear source energy is deposited in the fuel and cladding (typically modeled as the thermal source), but there is a small fraction that is deposited directly in the coolant and other structural components (direct gamma deposition).
Decay heat	Energy from the decay of fission products is the dominant source of energy in the core after reactor shutdown. There is also some energy from delayed fission soon after a shutdown.
Fuel slug radial power distribution	The radial power distribution defines the radial heat source within the fuel pin.
Heat capacity of core support structures	Support structures exchange thermal energy with the coolant and act as sources of stored energy after reactor shutdown.
Reactivity feedback	Reactor power changes when reactivity is inserted. Several phenomena can cause reactivity feedback including thermal expansion and contraction of the fuel, coolant and support structures due to temperature changes, the Doppler effect from fuel temperature changes and void in the coolant. Temperature changes affect the neutron cross-sections of core materials.
Reactivity feedback from mechanical changes in core structure	Expansion of core grid structure Expansion of control rod drive Deformation of structures over time (e.g. swelling) Bowing of fuel assemblies and blanket Deformation of core restraint system Axial thermal expansion of fuel and cladding
Reactivity feedback from fuel temperature changes	Doppler coefficient
Reactivity effects from burnup	Changes in fuel composition over time Changes in critical control rod position Changes in control rod worth Changes in fuel structure (e.g. axial growth under irradiation)
Sodium density effects	Temperature coefficient of reactivity Void coefficient (void from fission gas release to coolant channel or boiling of sodium)
Minor actinide content in fuel	High minor actinide content affects fuel performance: source term is different; physics is different; chemistry is different.
Rate of scram reactivity insertion	The scram reactivity affects the promptness of reactor shutdown.

Phenomenon	Definition and Related Comments
Delay of reactivity insertion	This is the time delay between the generation of the scram signal and the actual movement of the shutdown rods.
Eutectic reaction between fuel and cladding	Eutectic reaction refers to the chemical phenomenon in which two solid phases melt upon contact and form a liquid phase. In metallic fuel, eutectic reaction can occur between the fuel alloy, consisting of uranium, plutonium, and fission products, and cladding materials such as iron. Eutectic reaction occurs at high temperature (higher than 650ºC) and becomes more severe as temperature increase.
Temperature dependence of physical properties (thermal and mechanical) of materials	Physical properties such as specific heat, density, thermal conductivity, and creep characteristics of core components are temperature dependent. Parameters in fuel thermal diffusivity affect transient fuel temperature and stored energy.
Burnup or fluence dependence of physical properties (thermal and mechanical).	Strength of materials and thermal conductivities of fuel and some cladding materials are dependent on the operating history of the core. Thermal conductivity degradation in oxide fuel is known to increase the initial stored energy in fuel rods. This effect also affects the fuel rod dimensions (fuel slug radius, gap size, and cladding thickness) and creep characteristics of fuel/cladding over time.
Fission product (FP) release from fuel slug into gas plenum of fuel pin	FP gas forms pores in the fuel slug. FP gas is released to the gas plenum when these pores link and reach the outside surface of the fuel slug. The rate of FP gas release is dependent on the fuel temperature. The fuel pin internal pressure is a function of the FP gas release and it can limit the life cycle of a fuel pin (rod failure due to over pressurization). In addition, fuel swelling due to fission gas release can have reactivity and heat transfer impacts. Effective thermal conductivity of fuel slug depends on the pore density and the extent of sodium penetration through the pores.
FP transport from fuel to sodium bond and sodium coolant	In case of a fuel cladding failure, FP released from the fuel into the sodium bond leaks into the sodium coolant.
FP transport from sodium coolant to cover gas	In case of fuel failure, FPs leaked from fuel are transported to the cover gas.
Core bypass flow	Coolant flow at the gap between fuel assemblies serves to couple the fuel assemblies thermally at low flow conditions, such as during natural circulation decay heat removal.
Flow-induced vibration	Fuel pin bundles vibrate at high coolant flows due to fluid and structure interaction.
Power distribution, axial, radial and local	The core power distribution drives the core temperature distribution under steady-state. The power distribution is a function of the core composition and geometry.
Fuel cladding failure mechanisms	Fuel-clad physical and chemical interaction.
Flow blockage	Can reduce coolant flow and lead to sodium boiling.
Fuel swelling	Changes in coolant and fuel geometry can impact thermal-hydraulics and reactivity.

Phenomenon	Definition and Related Comments
Fuel-pin behavior with breached cladding	This is more a concern for oxide fuel.

Table 4-3 Phenomena in Vessel and Internal Structures

Phenomenon	Definition and Related Comments
Temperature fluctuation of reactor vessel by change of liquid level	If there is a cold sodium gap between the vessel liner and the vessel wall then level changes would not lead to temperature fluctuation on the vessel wall.
Coolant mixing in upper and lower plena	The extent of flow mixing (momentum and energy) is influenced by turbulence and coolant flow. In the upper plenum, the coolant reaches the inlet to the intermediate heat exchanger (IHX) after passing through the upper shroud. Coolant exits the core to the upper shroud region. The lower plenum receives flow streams through multiple passages.
Thermal stratification in upper and lower plena	Thermal stratification phenomena are observed in an upper plenum of liquid metal fast breeder reactors under reactor scram conditions, which gives rise to thermal stress on structural components. Thermal stratification in the lower plenum is significant only if there are marked differences in the temperatures of the incoming sodium streams.
Thermal striping in upper and lower plena	Thermal striping is a phenomenon that leads to random temperature fluctuations in the interface between non-isothermal streams arising out of jet instability. Because of the high heat transfer coefficient associated with liquid metal coolants such as sodium, the temperature fluctuations are transmitted to the adjoining structures with minimal attenuation, which eventually leads to high cycle fatigue and crack initiation in the structures.
Heat transfer between coolant and structure	The resistance at the interface between coolant and structure dictates how tightly the two are coupled thermally.
Flow through and around internal structure	Coolant flow (forced flow and natural convection) removes heat generated in structure.
Flow-induced vibration	Structure vibrates when coolant flows around it at high velocity.
Temperature dependence of physical properties (thermal and mechanical) of structural materials	Physical properties such as specific heat, density, thermal conductivity, and mechanical characteristics of structural components are temperature dependent.
Deformation of structural components due to thermal effect and irradiation	Thermal deformations by core temperature fluctuation and gamma heating and deformation by neutron irradiation and transmutation.
Direct heating of internal structures	Gamma and neutron capture generates heat in metal components.
Heat conduction in structure	Conduction is the heat transfer mechanism modulating the thermal response of structures.
Radiation heat transfer from vessel	This is one of the modes of heat transfer from the reactor to the outside environment.
Convective heat transfer from vessel	This is a mode of heat transfer for passive heat removal under some accident conditions.

Phenomenon	Definition and Related Comments
Pressure loss	Wall friction and local losses due to expansion and contraction of flow.
Heat capacity of coolant, vessel and internal structures	Support structures exchange thermal energy with the coolant and act as sources of stored energy after reactor shutdown.
Radial heat transfer through internal structures	This is the heat transfer across internal structures (e.g. radiation shield) that separate hot and cold sodium.

Table 4-4 Phenomena in Primary Heat Transport System

Phenomenon	Definition and Related Comments
Heat capacity of coolant	Heat capacity (product of mass and specific heat) of coolant in the primary heat transport system.
Natural circulation	Natural circulation is driven by a balance between buoyancy (density difference in different parts of the flow path) and pressure losses along the closed flow path. There can also be flow recirculation inside a subsystem or component driven by local buoyancy effects.
Pressure loss on primary side of IHX	Inlet and outlet losses, wall friction from flow in heat transfer tubes.
Heat transfer from primary to intermediate coolant	Intermediate coolant flows on the shell side of the IHX flows countercurrent to the primary coolant flow while being heated.
Primary coolant flow rate	Forced flow driven by primary EMP and natural circulation flow driven by buoyancy.
Intermediate coolant flow rate	Forced flow driven by intermediate EMP and natural circulation flow driven by buoyancy.
Heat capacity of IHX structure material	Structure includes, heat transfer tube, shielding, baffle, etc.
Flow distribution on primary side of IHX	Self-explanatory.
Flow coastdown of primary EMP	Flow coastdown performance characterizes the rate of flow decay after a primary EMP trip.
Pressure loss in primary EMP	Self-explanatory.
Primary EMP head curve	The head curve defines the rated pump performance within the operating range.
Heat capacity of primary EMP and Joule heat at flow coastdown	EMP consists of iron core, coil and structure material. It generates heat by the Joule effect during flow coastdown and releases it into the coolant.

Table 4-5 Phenomena in Intermediate Heat Transport System (IHTS)

Phenomenon	Definition and Related Comments
Pressure loss	IHTS flow path (sodium side) consists of IHX (shell side), steam generator (shell side), intermediate EMP and piping connecting the IHX and SGS.
Steam generation	The SG is a vertically oriented, helical coil, sodium-to-water counter flow heat exchanger. Feedwater enters the bottom of helical tubes, picking up heat and turning into superheated steam. Intermediate sodium flows down the shell side of the SG.
Natural circulation of intermediate sodium	IHX acts as a heat source while the SG is a heat sink. The IRACS (see discussion on decay heat removal system below) also is a heat sink for the intermediate sodium.
Heat transfer between the hot pool (in the upper plenum) and intermediate coolant.	Heat transfers from the primary coolant in the hot pool through the outer shell of the IHX to the intermediate coolant on the shell side of the IHX.
Flow coastdown of intermediate EMP	Flow coastdown performance characterizes the rate of flow decay after an intermediate EMP trip.
Pressure loss in intermediate EMP	Self-explanatory.
Intermediate EMP head curve	The head curve defines the rated pump performance within the operating range.
Heat capacity of SG	Heat capacity for structural materials, helical tubes, intermediate sodium coolant, and water/steam in SG.
Steam/water-sodium reaction	A SG tube leak can lead to steam/water-sodium reaction.
Pressure pulse impact from chemical reaction	Potential for propagation from a local failure to a system failure.
Flow distribution on intermediate side of IHX	Self-explanatory.

Table 4-6 Phenomena in Decay Heat Removal System

Phenomenon	Definition and Related Comments
Heat transfer to main condenser	During reactor shutdown heat is removed by the turbine condenser using the turbine bypass.
Heat transfer between SG wall and air flow	An intermediate reactor auxiliary cooling system (IRACS) provides an alternative method to remove shutdown decay heat during maintenance or repair operations. The IRACS uses natural or forced circulation of atmospheric air to remove heat from the outside shell of the SG.
Heat transfer between intermediate sodium and SG wall	Intermediate sodium flows downward on the shell side of the SG. Heat transfers from the sodium to the outside wall of the SG.
Pressure loss in the IRACS	The IRACS consists of an insulated shroud around the SG shell with an air intake at the bottom through the annulus and an isolation damper and exhaust fan located above the SG building. Natural circulation is initiated by opening the damper.
Pressure loss in the RVACS	The RVACS is a passive system that relies on natural circulation air-cooling to remove heat from the reactor module. Atmospheric air is drawn into the reactor building and flows over the outside of the containment vessel. The warm air then returns to the stack and is exhausted. The RVACS is always operating to remove heat from the containment vessel.
Heat transfer between reactor vessel and containment vessel (or guard vessel)	Increased temperature in sodium and reactor vessel will increase radiant heat transfer across the argon gap to the containment vessel.
Heat transfer between the containment vessel and air flow	Heat is transferred mainly by convection from the containment vessel to the upwardly flowing atmospheric air around the vessel.
Asymmetric airflow	The airflow path in the RVACS is an annulus between the containment vessel and the collector cylinder located in the reactor silo. The tortuous flow path of the RVACS has the potential to create asymmetric flow resulting in asymmetric temperature distribution in the reactor.

Table 4-7 Phenomena for the Sodium Coolant

Phenomenon	Definition and Related Comments
Formation of sodium oxide	Problems have been encountered at the sodium/cover-gas interface, resulting from sodium oxide formation that can lead to binding of rotating machinery, control rod drives, and contamination of the sodium coolant.
Sodium vapor condensation and plate out	Sodium vapor tends to plate out in the interstitial spaces in the reactor closure (or the shield plug) hampering normal operations.
Structural material corrosion	The ability of Na to dissolve oxygen is the chief reason for its corrosive property. Na_2O is highly corrosive and is relatively insoluble in Na especially at low temperatures. Deposition of Na_2O in cooler passages can plug narrow passages. Self-welding and thermal-gradient transfer are two corrosion mechanisms that can cause damage to components of SFRs. Self-welding is caused by Na reducing the oxides of surfaces in contact and can result in the malfunction of pumps and valves. Thermal-gradient transfer operates by transferring materials from high temperature region to low temperature region. This is enabled by the different solubility of the materials in Na at different temperatures. Extended operation with coolant circulation can result in corrosion in hotter regions and plugging in cooler regions of the system.
Sodium purity control	Oxygen that enters the system will react with Na and purification of the Na coolant is done by bypassing a portion of the hot coolant (where oxide solubility is highest) and depositing the oxide in a cold trap.
Sodium spray dynamics	Sodium leakage from primary or intermediate heat transfer system at high pressure (~1 MPa) into a compartment of the reactor containment. Jet/spray breakup and spray combustion; heat transfer from spray to atmosphere and structure; and aerosol (smoke) formation from spray
Sodium jet dynamics	Sodium leakage from primary or intermediate heat transfer system at low pressure (~0.1 MPa) into a compartment of the reactor containment. Surface combustion; aerosol (smoke) formation; heat transfer to atmosphere and structure.
Sodium-pool fire on an inert substrate	Of interest in the event of a sodium leak.
Aerosol dynamics	Of interest in the event of a sodium leak.
Sodium-cavity liner interaction	Of interest in the event of a sodium leak.
Sodium-concrete interaction	Of interest in the event of a sodium leak.
Heat Transfer from sodium fire	Heat transfer from atmosphere to structure

4.4 References

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4. K. Kawada et al., “Development of PIRT (Phenomena Identification and Ranking Table) for SAS-SFR (SAS4A) Validation,” Proceedings of the 2014 22nd International Conference on Nuclear Engineering, ICONE-22, July 7-11, 2014.
5. NRC, “Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor,” NUREG-1368, U.S. Nuclear Regulatory Commission (NRC), February 1994.

APPENDIX

A list of events with corresponding phenomena that would need to be simulated for a generic sodium fast reactor are given in Tables A-1, A-2, and A-3[c] for anticipated operational occurrences (AOOs), design-basis accidents (DBAs) and beyond design-basis accidents (BDBAs), respectively. For the sake of completeness, they include events with sodium leaking from the primary or fuel disruptions although those events are beyond the scope of the review in this document.

[c] The tables are modified from R. Schmidt et al., "Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Safety," SAND2011-4145, Sandia National Laboratories, June 2011.

Table A-1 Anticipated Operational Occurrences and Relevant Phenomena

Event Description	Phenomena
AOO-1: Protected reactivity insertion event (e.g. control rod withdrawal or drop) and subsequent system response to SCRAM	**Reactivity effects prior to scram** * reactivity feedback at high power * end-of-life prediction of reactivity feedback * burnup control swing / control rod worth * integrity of fuel with breached cladding * integrity of fuel with load following
AOO-2: Protected reactivity insertion event due to seismic event and subsequent system response to SCRAM	**Relative motion of core and control rods** **Reactivity effects prior to scram** * reactivity feedback at high power * end-of-life prediction of reactivity feedback * burnup control swing / control rod worth * integrity of fuel with breached cladding * integrity of fuel with load following
AOO-3: Protected loss of core cooling due to equipment failure or operator error and subsequent system response to SCRAM	**Thermal-hydraulics** * single phase transient sodium flow * thermal inertia * pump coast-down profiles * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena **Reactivity effects prior to scram** * mechanical changes in core structure * intact fuel expansion * fuel/coolant/structure temperatures
AOO-4: Protected loss of normal heat sink due to equipment failure or operator error, and subsequent system response to SCRAM	**Thermal-hydraulics** * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena

Table A-2 Design-Basis Accidents and Relevant Phenomena

Event Description	Phenomena
DBA-1: Protected Reactivity Insertion event (e.g. accident due to rapid withdrawal of control rods) and subsequent system response to SCRAM	**Same as AOO-1 case (see Table A-1) plus** * reactivity effects of gas bubble entrainment
DBA-2: Protected reactivity insertion event due to seismic event and subsequent system response to SCRAM	**Same as AOO-2 case (see Table A-1) but** * larger relative motion of core and control rods
DBA-3: Protected loss of core cooling due to equipment failure or operator error and subsequent system response to SCRAM	**Same as AOO-3 case (see Table A-1)**
DBA-4: Protected loss of local core cooling due to a partial internal flow blockage and subsequent system response to SCRAM	**Thermal-hydraulics** * effect of subassembly flow redistribution * single phase transient sodium flow * thermal inertia * pump-coast down pump coast-down profiles * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation
DBA-5: Protected loss of normal heat sink due to power-conversion system tube rupture and subsequent system response to SCRAM	**Thermal-hydraulics** * sodium-steam chemical reaction * CO_2-sodium chemical reaction * pressure-pulse impacts from chemical reaction * sodium stratification * transition to natural convection core cooling core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena • reaction product formation and deposition
DBA-6: Protected loss of normal heat sink due to equipment failure other than steam-generator tube rupture, and subsequent system response to SCRAM	**Thermal-hydraulics** * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena

Event Description	Phenomena
DBA-7: Sodium leakage from the primary or intermediate cooling system at high pressure (~1 MPa) into a compartment of the reactor containment.	* sodium-pool fire on an inert substrate * aerosol dynamics * sodium-cavity-liner interactions * sodium-concrete-melt interactions
DBA-8: Sodium leakage from the primary or intermediate cooling system at low pressure (~0.1 MPa) into a compartment of the reactor containment;	* sodium jet dynamics * sodium-pool fire on an inert substrate * aerosol dynamics * sodium-cavity-liner interactions * sodium-concrete-melt interactions

Table A-3 Beyond Design-Basis Accidents and Relevant Phenomena

Event Description	Phenomena
BDBA-1: ATWS unprotected reactivity insertion event (e.g. accident due to rapid withdrawal of control rods), not leading to severe accident case.	**Same as for DBA-1 case (see Table A-2) plus** **Thermal-hydraulics** * heat removal path/capacity **Reactivity effects** * reactivity feedback at high power * coolant heating and margin to boiling * core reactivity feedback * core thermal and structural effects **Material behavior** * fuel cladding structural integrity at elevated temperatures * cooling systems structural integrity at elevated temperatures * containment structure integrity
BDBA-2: Unprotected reactivity insertion event due to seismic event, not leading to severe accident case.	**Same as DBA-2 case (see Table A-2) plus** * even larger relative motion of core and control rods
BDBA-3: ATWS unprotected loss of core cooling due to equipment failure or operator error, not leading to severe accident case.	**Same as for DBA-3 case (see Table A-2) plus** **Thermal-hydraulics** * margin to boiling at peak temperature * core thermal and structural effects * heat removal path and capacity **Reactivity effects** * core reactivity feedback • fuel motion in intact fuel pins • core restraint system performance * reactor shutdown mechanism **Material behavior** * long-term performance of structures at elevated temperatures * fuel cladding integrity at elevated temperature

Event Description	Phenomena
BDBA-4: Unprotected loss of local core cooling due to a partial internal flow blockage, not leading to severe accident case.	**Thermal-hydraulics** * effect of subassembly flow redistribution * single phase transient sodium flow * thermal inertia * pump coast down profiles * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation
BDBA-5: Unprotected loss of normal heat sink due to power-conversion system tube rupture, not leading to severe accident case.	**Same as for DBA-5 case (see Table A-2) plus** **Thermal-hydraulics** * thermal inertia * core thermal and structural effects **Reactivity Effects:** * core reactivity feedback * fuel motion in intact fuel pins (metal fuel) * core restraint system performance * reactor shutdown mechanism **Material behavior** * long-term performance of structures and piping at elevated temperatures * fuel cladding structural integrity at elevated temperatures * containment structure integrity
BDBA-6: ATWS unprotected loss of normal heat sink due to equipment failure other than steam-generator tube rupture, not leading to severe accident case.	**Same as for protected events plus** **Thermal-hydraulics** * thermal inertia, core thermal / structural effects **Reactivity Effects:** * core reactivity feedback fuel motion in intact fuel pins core restraint system performance * reactor shutdown mechanism **Material behavior** * long-term performance of structures at elevated temperatures * fuel cladding structural integrity at elevated temperatures * containment structure integrity

Event Description	Phenomena
BDBA-7: Sodium leakage from the primary or intermediate cooling system at high pressure (~1 MPa) into a compartment of the reactor containment.	* sodium spray dynamics * sodium-pool fire on an inert substrate * aerosol dynamics * sodium-cavity-liner interactions * sodium-concrete-melt interactions
BDBA-8: Sodium leakage from the primary or intermediate cooling system at low pressure (~0.1 MPa) into a compartment of the reactor containment.	* sodium jet dynamics * sodium-pool fire on an inert substrate * aerosol dynamics * sodium-cavity-liner interactions * sodium-concrete-melt interactions • plant dynamics
BDBA-9: Severe accidents – substantial core melting such as: * severe loss of core cooling event * severe reactivity addition event, * severe loss of heat rejection capability (but not including protected complete loss of heat rejection capability, i.e. BDBA-10)	**Essentially the same as other BDBAs plus** **Fuel and Core Behavior:** * sodium voiding effects * temporal and spatial incoherence * fuel pin failure * fuel dispersal and coolability * re-criticality * potential for energetic events (oxide fuel) * primary vessel thermal and structural integrity (oxide fuel) * radiation release and transport (oxide fuel)
BDBA-10: Protected complete loss of heat rejection capability leading to a severe accident (substantial core melting).	**Same as for BDBA-9 but accident time-scale is longer**