
NRC Regulatory History of Non-Light Water Reactors (1950-2019)

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ABSTRACT

This report describes the U.S. Nuclear Regulatory Commission's (NRC's) history with the licensing of non-light water reactors (non-LWRs). The focus is on regulatory policy and licensing issues that have arisen in the past so that NRC staff will be in a better position to deal with these matters in the future. It is not an objective to discuss in any detail the technology of non-LWR designs. Documenting the historical policy issues and licensing approaches is particularly important to NRC staff unfamiliar with non-LWRs and is an important incentive for the report. Hence, the report is written as a tutorial rather than as an historical archive. The subject is approached chronologically going from the early days of the Atomic Energy Commission to the formation of the NRC, then into the era when the regulatory structure newly developed for LWRs provided guidance for other types of reactors, followed by the era when policy was made specifically for advanced reactors including non-LWRs. This background provides the transition to current planning for future non-LWR licensing and policy—also described herein.

TABLE OF CONTENTS

ACRONYMS	vi
1 INTRODUCTION.....	1-1
1.1 Background.....	1-1
1.2 Objective	1-1
1.3 Outline of Report.....	1-2
1.4 References.....	1-2
2 EARLY NON-LWR HISTORY	2-1
2.1 AEC Period (1946-1974).....	2-1
2.2 Early Non-LWR Licensing at NRC	2-5
2.3 References.....	2-6
3 NON-LWR REGULATORY AND LICENSING ISSUES	3-1
3.1 Gas Cooled Reactors.....	3-1
3.1.1 Introduction	3-1
3.1.2 Licensing of High Temperature Gas Cooled Reactors.....	3-2
3.1.3 Operating Experience and Lessons Learned	3-7
3.1.4 Regulatory and Licensing Issues	11
3.2 Liquid Metal Cooled Reactors	14
3.2.1 Introduction	3-14
3.2.2 Licensing of Liquid Metal Reactors	3-14
3.2.3 Operating Experience and Lessons Learned	3-19
3.2.4 Regulatory and Licensing Issues	3-22
3.3 Heavy Water Reactors.....	3-23
3.3.1 Introduction	3-23
3.3.2 Regulatory and Licensing Issues	3-24
3.4 Liquid-Fuel Reactors	3-27
3.4.1 Introduction	3-27
3.4.2 Licensing of Liquid-Fuel Reactors.....	3-27
3.4.3 Regulatory and Licensing Issues	3-29
3.5 References.....	3-29
4 TRANSITION TO THE PRESENT AND BEYOND.....	4-1
4.1 Advanced Reactor Policy	4-1
4.2 Principal Design Criteria.....	4-5
4.2.1 Background.....	4-5
4.2.2 Design Criteria for Non-LWRs.....	4-5
4.2.3 Development of RG 1.232.....	4-6
4.2.4 Applicability, Intended Use, and Key Assumptions	4-7
4.3 Additional NRC Statements on Advanced Reactor Policy.....	4-8
4.4 Next Generation Nuclear Plant (NGNP) Interactions with NRC	4-10
4.5 Planning for the Future.....	4-12
4.5.1 NRC's Vision and Strategy for Regulating Non-Light Water Reactors	4-12
4.5.2 Regulatory Review Roadmap for Non-LWRs.....	4-14
4.5.3 Licensing Modernization Project	4-16
4.5.4 International Activities	4-22
4.6 References.....	4-23

LIST OF FIGURES

Figure 2-1 President Truman Signing the AEA	2-1
Figure 2-2 Artist Rendering of CRBR Plant	2-5
Figure 2-3 Timeline of Events in Nuclear Regulation and for Non-LWRs.....	2-7
Figure 3-1 FSV Refueling Floor.....	3-3
Figure 3-2 TRISO Fuel Particle	3-8
Figure 3-3 EBR-II Control Room	3-15
Figure 3-4 PRISM Plant	3-18
Figure 3-5 Darlington CANDU Plant.....	3-26
Figure 3-6 SHINE Isotope Generator	3-28
Figure 4-1 NGNP Reactor and Power Conversion Vessels	4-11
Figure 4-2 NRC Non-LWR Readiness Roadmap [4-39]	4-13
Figure 4-3 NRC Licensing-Related processes	4-15
Figure 4-4 NRC Approach to RIPB Decision-Making	4-17
Figure 4-5 Proposed Traveling Wave Reactor	4-18

LIST OF TABLES

Table 2.1 Non-LWR Reactors Supported by the AEC.....	2-3
Table 3.1 Gas Cooled Reactors Around the World	3-1
Table 3.2 Liquid Metal Cooled Reactors Around the World.....	3-14
Table 4.1 Example Criteria Presented in RG 1.232	4-7

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
AEA	Atomic Energy Act
AEC	U.S. Atomic Energy Commission
AECLT	Atomic Energy Canada Limited Technologies, Inc.
ANS	American Nuclear Society
ARDC	Advanced Reactor Design Criteria
ASME	American Society of Mechanical Engineers
BDBE	Beyond Design Basis Event
CFR	Code of Federal Regulations
CRBR	Clinch River Breeder Reactor
DBA	Design Basis Accident
DC	Design Criteria
DG	Draft Regulatory Guidance
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
EC	Event Categories
ERA	Energy Reorganization Act of 1974
ERDA	Energy Research and Development Administration
EP	Emergency Planning
EPZ	Emergency Planning Zone
F-C	Frequency-Consequence
FFTF	Fast Flux Test Facility
FSV	Fort St. Vrain
GDC	General Design Criteria
GEM	Gas Expansion Module
HTGR	High Temperature Gas Reactor
HTR	High Temperature Reactor
IAEA	International Atomic Energy Agency
IAP	Implementation Action Plan
INL	Idaho National Laboratory
ISG	Interim Staff Guidance
LBE	Licensing Basis Events
LMP	Licensing Modernization Project
LMR	Liquid Metal Reactor
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MHTGR	Modular High-Temperature Gas Cooled Reactor
ML	ADAMS Accession No.
MST	Mechanistic Source Term
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission

NST	Non-Safety-Related with No Special Treatment
PBAPS	Peach Bottom Atomic Power Station
PDC	Principle Design Criteria
PRA	Probabilistic Risk Assessment
PRDC	Power Reactor Development Company
PRDP	Power Reactor Demonstration Program
PRISM	Power Reactor Innovative Small Module
PSC	Public Service Company of Colorado
PSID	Preliminary Safety Information Document
QA	Quality Assurance
R&D	Research and Development
RG	Regulatory Guide
RSS	Reserve Shutdown System
RVACS	Reactor Vessel Auxiliary Coolant System
SDA	Standard Design Approval
SECY	NRC Commission Paper
SEFOR	Southwest Experimental Fast Oxide Reactor
SER	Safety Evaluation Report
SFR	Sodium Fast Reactor
SMR	Small Modular Reactor
SNM	Special Nuclear Material
SR	Safety Related
SRM	Staff Requirements Memorandum
SSCs	Structures, Systems, and Components
TI-RIPB	Technology-Inclusive Risk-Informed Performance-Based
TRISO	Tristructural Isotropic Fuel
UK	United Kingdom

1 INTRODUCTION

1.1 Background

Over the past several years there has been significant interest in the development and licensing of advanced reactors that will be very different from the light water reactors (LWRs) that are currently used to generate electricity in the U.S. For example, some advanced reactors will use gas, liquid metal, or molten salt as a coolant, some will have a fast neutron spectrum (LWRs have a thermal neutron spectrum), and some will be much smaller in size than current generation LWRs. There are many possible applications for these reactors including electricity production, process heat, research and testing, isotope generation, or space applications.

To prepare for potential *non-LWR* application submittals, the U.S. Nuclear Regulatory Commission (NRC) has written a vision and strategy document [1-1] that outlines the tasks that must be undertaken to advance technical and regulatory readiness and related communications for these reactors. That document is supported by implementation action plans (IAPs) [1-2] that cover the near-term actions to be taken in the next five years based on six basic strategies.

Implementation action plan Strategy 1 involves acquiring and developing sufficient knowledge, skills, and capacity to perform reviews of non-LWR applications. Hence, the NRC staff (henceforth, the staff) “must be familiar with a range of potential technologies, must have adequate training support in place, [and] must have a non-LWR knowledge-base available,” ([1-2], p 8). In order to perform their regulatory function efficiently, the staff must be familiar with the current relevant policies and how the NRC has assured the safety of non-LWRs in the past.

Indeed, there is a long history of the NRC regulating non-LWRs. This has primarily been focused on gas cooled and liquid metal cooled reactors but in addition, there has been interaction with a heavy water reactor designer, with aqueous homogeneous reactor designers and recently, molten salt reactor vendors. To some extent, this has been chronicled in general histories of the NRC [1-3, 1-4, 1-5]. More recent sources of relevant regulatory information are found on the NRC’s advanced reactor website [1-6] and in an NRC knowledge management report on liquid metal reactors [1-7].

1.2 Objective

The objective of this report is to describe the NRC’s history with the licensing of non-LWRs and to provide a list of references where the reader can obtain more information. The focus is on regulatory policy and licensing issues that have arisen in the past so that NRC staff will be in a better position to deal with these matters in the future. It is not an objective to discuss in any detail the technology represented by non-LWR designs nor to provide a complete list of NRC documents related to non-LWR technology. The report does not address issues related to the fuel cycle that may arise if for example, high assay low-enriched uranium (greater than 5 weight percent and less

than 20 weight percent U-235) fuel is used or if the reactor has liquid fuel that is processed onsite to remove actinides.

This report fits into Strategy 1 of the IAP which involves acquiring and developing sufficient knowledge, skills, and capacity to perform reviews of non-LWR licensing applications. To this end, documenting the historical policy issues and licensing approaches is particularly important to NRC staff unfamiliar with non-LWRs and is the basis for the report's objective. Hence, the report is written as a tutorial rather than as an historical archive.

The subject is approached chronologically going from the early days of the Atomic Energy Commission to the formation of the NRC, then into the era when the regulatory structure newly developed for LWRs provided guidance for other types of reactors, and then into the era when policy was made specifically for advanced reactors including non-LWRs. This background provides the transition to current planning for future non-LWR licensing and policy.

1.3 Outline of Report

Chapter 2 discusses the many designs considered during the early days of the NRC and its predecessor, the Atomic Energy Commission. It is seen that as the industry focused on LWRs, regulatory policy was developed for that reactor type and non-LWRs were treated as exceptions or exemptions to regulations. Although Chapter 2 introduces early non-LWR designs, details of the technological and specific regulatory and licensing issues that arose with non-LWRs are discussed in Chapter 3, specifically for gas cooled, liquid metal cooled, heavy water, and aqueous homogenous reactors. Chapter 4 is the transition to what is now policy in the 21st century. The chapter discusses policy/regulations meant to be applicable to non-LWRs and those that are not specific to non-LWRs but are still applicable. It also includes input to the licensing regime from industry and international organizations. The organization of these chapters is such that a reader, depending on interest, can focus on only one of the chapters.

The Appendix contains tables with a chronological list of relevant documents—a subset of the references in every chapter. One table is for NRC documents and one for non-NRC documents. The listings include a brief summary of the document's contents. The list is meant to be comprehensive but not overwhelming. Regulatory documents not unique to non-LWRs, but still relevant (e.g., on severe accident policy), are not listed.

1.4 References

- 1-1. "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," U.S. Nuclear Regulatory Commission, 2016. (ML16139A812)

- 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," U.S. Nuclear Regulatory Commission, 2016. (ML16334A495)
- 1-3. "No Undue Risk, Regulating the Safety of Operating Nuclear Power Plants," NUREG/BR-0518, U.S. Nuclear Regulatory Commission, June 2014. (ML14181A493)
- 1-4. J. Samuel Walker and Thomas R. Wellock, "A Short History of Nuclear Regulation, 1946-2009," U.S. Nuclear Regulatory Commission, October 2010. (ML102980443)
- 1-5. <https://www.nrc.gov/about-nrc/history.html>
- 1-6. <https://www.nrc.gov/reactors/new-reactors/advanced.html>
- 1-7. G.F. Flanagan, G.T. Mays, and I.K. Madni, "NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors," NUREG/KM-0007, U.S. Nuclear Regulatory Commission, April 2014. (ML14128A346)

2 EARLY NON-LWR HISTORY

2.1 AEC Period (1946-1974)

The Atomic Energy Commission (AEC) was formed by Congress via the Atomic Energy Act (AEA) of 1946. Among its responsibilities was research and development that would lead to reactors to generate electricity and/or special nuclear material (SNM, specifically plutonium).

Toward this end, by the beginning of the fifties, the AEC began to get industrial participation and initiate work at national laboratories that led to the building of many research, test, and prototype or demonstration reactors.^a



Figure 2.1 President Truman Signing the AEA

Early non-LWRs

In the beginning many different types of non-LWRs were considered. It became clear that dual purpose (both electricity and SNM)

reactors were not the best path forward because of conflicting demands, for

example, power reactors need to operate for long periods and production reactors need to interrupt operations frequently to process fuel. Although it also became clear that water cooled reactors would become the favorite for the commercial sector for power production, this period did see development of non-LWRs as discussed below.

During the 1950s safety was regarded to be important but the corresponding regulatory infrastructure was minimal. Safety evaluations involved writing so-called hazards-summary reports which were evaluated by committees. However, since the reactors being built were essentially all experimental reactors, it was not surprising that there were many unforeseen technical problems that led to unsafe conditions and/or unexpected events with important lessons learned. For example, in 1955 there was a meltdown at the sodium cooled, fast spectrum, EBR-1 reactor due to a rapid increase in core reactivity from a positive reactivity coefficient. Although the reactor had been doing reactivity experiments at the time, the undesirable outcome was the result of miscommunication within the experimental team and could have been prevented [2-2].

^a References that focus on the history of the AEC include [2-1], [2-2], and [2-3]. All three discuss reactor development and [2-3] also discusses the AEC's responsibilities with respect to nuclear weapons.

The hazards-summary report required by 1951 “had to include a description of the reactor and the site, a detailed plan of operation, a schedule of chemical processing and disposal of reactor fission products, the methods of disposal of radioactive effluents, and a description of the safety mechanisms of the reactor... [The report] had to list all the known potentially hazardous features and include the experimental information, calculations, and assumptions used in evaluating those hazards. The report required information on steps taken to minimize the risks and an estimate, if a failure should occur, on the extent of any release of radioactive material and the damage to be expected.” [2-2]

Improved safety was one of the considerations in the revision of the Atomic Energy Act in 1954. The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory committee that was (and still is) to provide oversight on safety and report directly to the Commission. Prior to this, advisory committees existed on an ad hoc basis. The concerns over safety also led to the establishment of a separate Reactor Hazard Evaluation Staff within the AEC in 1955 (to become the Hazards Evaluation Branch). This enabled more effort in the review of hazards reports and the establishment of safety standards, guides, regulations, and codes. The law stipulated that licensing would be a two-step process with a construction permit coming after an initial review and an operating license before startup of the plant. In practice, the AEC also allowed for conditional or provisional permits if some technical information was still missing from the application. Although the process was defined, the evaluation of hazards was difficult due to all the technical uncertainties. For example, there was limited experience in how properties of materials changed with irradiation and high stress levels, or how water or other coolants would interact with metals at high temperature, or the impact of uncertainties in nuclear properties, etc.

Another important thrust of the 1954 act was to obtain more commercial involvement in the development of nuclear energy. In 1955 the Power Reactor Demonstration Program (PRDP) was initiated to help the private sector invest in nuclear energy while getting support from the AEC for research and development to reduce technical uncertainty.

Table 2.1 is a list of early non-LWRs considered for power production and includes designs with liquid metal, organic liquid, gas, and heavy water cooling. It also lists one short-lived experiment with liquid fuel in the form of an aqueous solution. More information on the licensing of some of these early reactors is provided in Chapter 3.

Table 2.1 Non-LWR Reactors Supported by the AEC

Reactor Type	Power	Reactor Name, Acronym	Startup Date
Sodium Cooled Fast Spectrum	1.7 MWe, 20 MWe	Experimental breeder Reactor, EBR-1, EBR-2	1951, 1962
Sodium Cooled Fast Spectrum	69 MWe	Enrico Fermi Atomic Power Plant	1962-63
Graphite Moderated Sodium Cooled	6.5 MWe	Sodium Reactor Experiment, SRE	1957
Graphite Moderated Sodium Cooled	75 MWe	Hallam	1962
Organic Moderated	5-6 MWt	Organic Moderated Reactor Experiment	1957
Organic Moderated	12.5 MWe	Piqua Organic Moderated reactor	1963
Heavy Water Cooled	70 MWt	Heavy Water Components Test Reactor	1962
Heavy Water Cooled	17 MWe	Carolinas-Virginia Tube Reactor, CVTR	1963
Gas Cooled	40 MWe	Peach Bottom Atomic Power Station, PBAPS	1967
Aqueous Homogeneous Reactor (Liquid Fuel)	150 kWe	Homogeneous Reactor Experiment, HRE-1, HRE-2	1952, 1957

Evolution of AEC as a Regulator of Nuclear Energy

During the AEC period it was recognized that there was an inherent conflict of interest in the role of the agency both as a promoter and as a regulator of nuclear energy. One situation which showed the importance of this issue was the result of the Power Reactor Development Company's (PRDC's) application for a construction permit for the Fermi plant, a non-LWR design, in 1956. The licensing process became contentious because of the lack of knowledge about sodium cooled fast spectrum reactors and because of what had happened at EBR-1. The ACRS concluded that there was insufficient information to ensure safe operation and urged that the AEC expand its experimental programs with fast breeder reactors to seek more complete data on the issues raised in the PRDC application. The plant nevertheless received a construction permit. The congressional Joint Committee on Atomic Energy (the oversight committee at the time) was at odds with the AEC and had legislation enacted to make licensing a more open process. The controversy, however, continued into the 1960s and during this period the AEC was frequently labeled as more interested in promoting nuclear power than ensuring nuclear safety.

Starting in the late 1950s and extending through the 1960s, the AEC organization evolved to separate the regulatory/safety function from the research/promotion function.

The staff involved in licensing, however, still relied on staff in the other branches that had expertise on reactor technology. At the same time the ACRS started its own evaluations as an independent advisory organization. There was also deliberation as to what an effective regulatory organization should include even if it was within the AEC, for example, should it have commissioners or be an agency headed by a single administrator. The separation of the two functions of the AEC became more apparent when a newly created position of Director of Regulation reported directly to the Commission in 1961 and when the regulatory organization moved into offices a good distance from the remainder of the AEC in 1962.

In 1962 the AEC issued a major report on civilian nuclear power [2-4]. The report discussed the limitations of fossil fuel supplies and stated that LWR technology was “on the threshold of being competitive with conventional power in the highest fuel cost areas.” To supplement the limited supply of U-235 needed for these reactors, the report called for a successful breeder-reactor program was needed. Although the report recommended further reactor development, it did acknowledge safety issues and the need to address them. Licensing and regulation were acknowledged in the report and the recent steps to improve that process were outlined. However, it was the promotional aspects of the report that elicited the most pro and con comments. According to one historian, [2-2] “Atomic regulation, by contrast, was largely invisible, intangible, and undramatic.”

A follow-up report in 1967 also emphasized the importance of adding to fuel supplies with a converter or breeder. The liquid metal fast breeder reactor (LMFBR) was the answer and two projects were put in place. The Clinch River Breeder Reactor (CRBR), was meant to be a demonstration plant and was launched in 1972. The Fast Flux Test Reactor was also designed during this period. These reactors are discussed in Section 3.2 as early non-LWR licensing experience. Experience with gas cooled reactor licensing during this period corresponds to the experience with LMRs and is discussed in Section 3.1.

Energy Reorganization Act and Creation of NRC

The Energy Reorganization Act (ERA) of 1974 marked the end of the AEC and the founding of the NRC to carry out the independent licensing and regulation of nuclear reactors. At the same time the Energy Research and Development Administration (ERDA) took over the role of promoting research, development, and deployment of reactors (in 1977 to become the Department of Energy, which had a broader energy-related mission). In addition to alleviating the obvious conflict of interest that had been a criticism of the AEC for many years, the change was also motivated by changes taking place within both the promotional and the regulatory side of the agency. For example, energy, not just nuclear energy, became of increasing importance after the Arab oil embargo in 1973. Additional regulatory procedures became necessary after a major judicial ruling—the so-called Calvert Cliffs decision in which a Federal Court of Appeals ruled that some AEC regulations were insufficient [2-3]. Furthermore, the AEC announced at the end of 1973 new requirements for emergency core cooling systems in

LWRs after two years of rule-making. It was clear by the time the ERA was passed and signed into law that a body for regulation of nuclear safety needed to stand on its own.

2.2 Early Non-LWR Licensing at NRC

After the creation of the NRC, regulatory activities continued along the same lines that had been started within the AEC. However, during the remainder of the 20th Century, the licensing process became more proscriptive with significant changes to licensing requirements. Some of these changes were motivated by incidents such as the fire at the Brown's Ferry plant (1975), the severe accident due to a loss-of-coolant at the Three Mile Island plant (1979), and the explosion at the Chernobyl plant in the Soviet Union (1986). Other changes were motivated by a better understanding of risk as a result of the use of probabilistic risk assessment techniques, the push for performance-based requirements by industry, and the tightening of international radiation standards. And other changes were the result of additional experience with LWRs—by the end of the 1980s, approximately 100 LWRs were operating in the U.S.

During this period there was also attention on non-LWR licensing. For example, during the 1980s the modular high temperature gas cooled reactor (MHTGR) was being reviewed (cf. Section 3.1) as were liquid metal cooled designs (CRBR and PRISM; cf. Section 3.2).

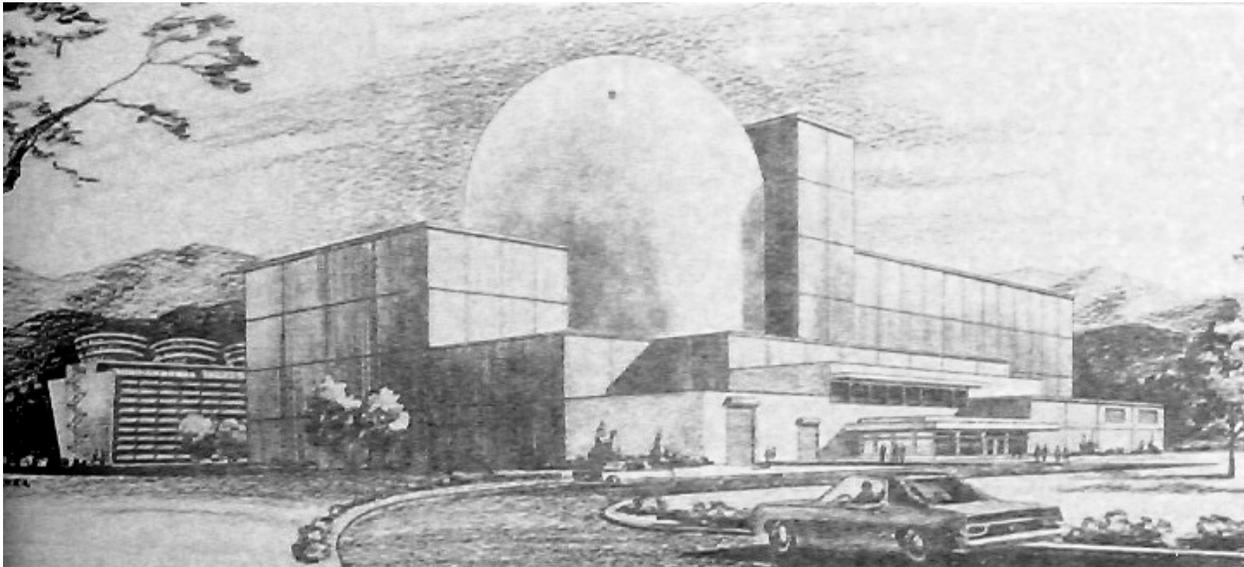


Figure 2.2 Artist Rendering of CRBR Plant

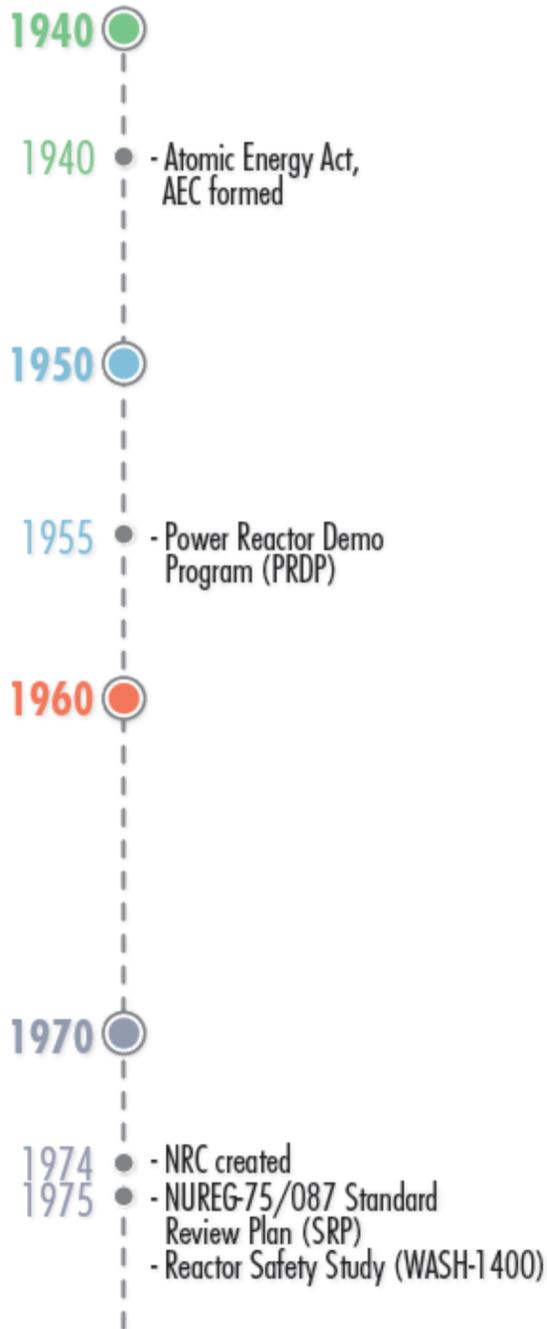
Although many licensing requirements were up to this time written for LWRs, it was always possible to have an exception or exemption from a rule. Based on 10 CFR 50.12, Specific Exemptions, “The Commission may, ... grant exemptions from the requirements of the regulations ...” However, it was clear that this was not the best approach and it was necessary for NRC to define requirements that more directly address non-LWRs, with corresponding policy, regulations, and guidance.

In 1986 the NRC issued an advanced reactor policy statement and subsequently more effort was put into how licensing reviews should be done. Advanced reactors were to include evolutionary LWRs, non-LWRs, and small modular light water reactors. This evolution into a licensing regime for non-LWRs is addressed in Chapter 4 and is demonstrated in the timelines shown in Figure 2.3. The figure shows relevant general regulatory events as well as those specific to non-LWRs.

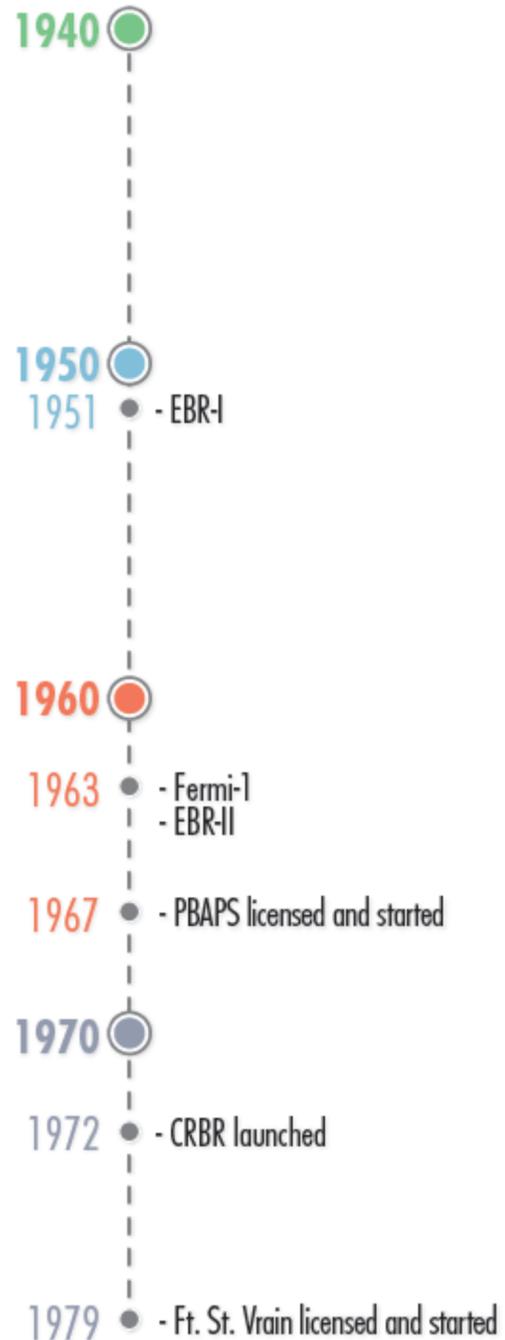
2.3 References

- 2-1. Wendy Allen, "Nuclear Reactors for Generating Electricity: U.S. Development from 1946 to 1963," R-2116-NSF, The Rand Corporation, June 1977.
- 2-2. George T. Mazuzan and J. Samuel Walker, "Controlling the Atom, The Beginnings of Nuclear Regulation 1946-1962," NUREG-1610, U.S. Nuclear Regulatory Commission, reprint date April 1997.
- 2-3. Alice Buck, "The Atomic Energy Commission," U.S. Department of Energy, July 1983.
- 2-4. "Civilian Nuclear Power, A Report to the President – 1962," U.S. Atomic Energy Commission, November 20, 1962.
- 2-5. J. Samuel Walker and Thomas R. Wellock, "A Short History of Nuclear Regulation, 1946-2009," U.S. Nuclear Regulatory Commission, October 2010. (ML102980443)
- 2-6. "No Undue Risk, Regulating the Safety of Operating Nuclear Power Plants," NUREG/BR-0518, U.S. Nuclear Regulatory Commission, June 2014. (ML14181A493)

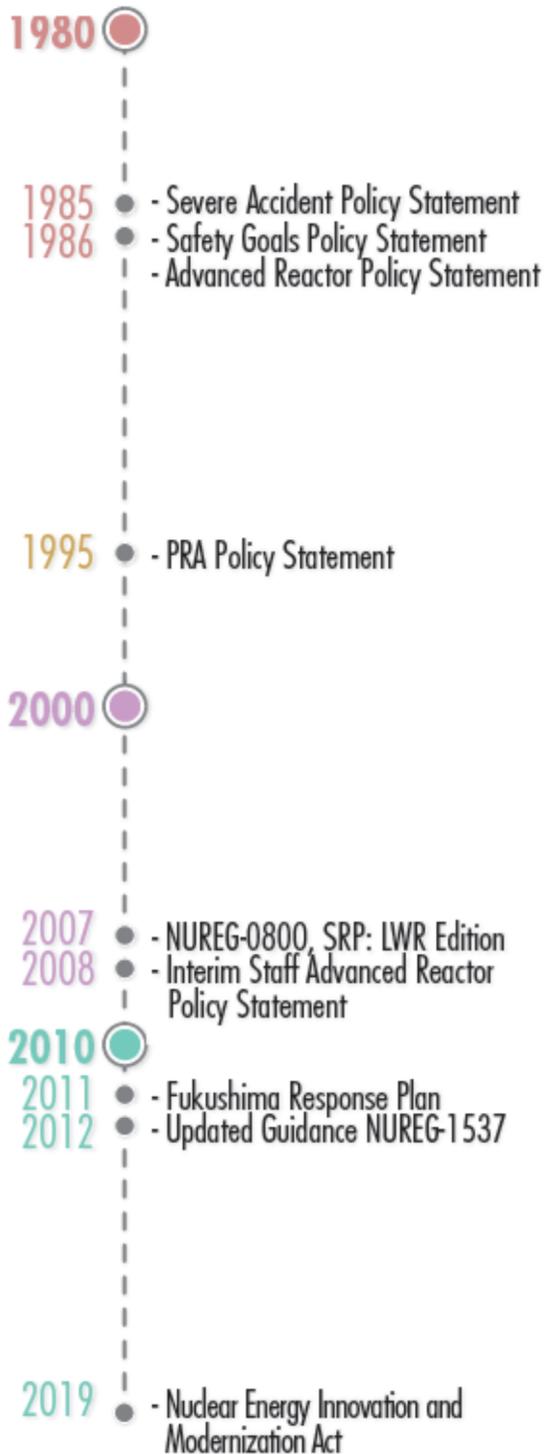
General Events



Non-LWR Events



General Events



Non-LWR Events

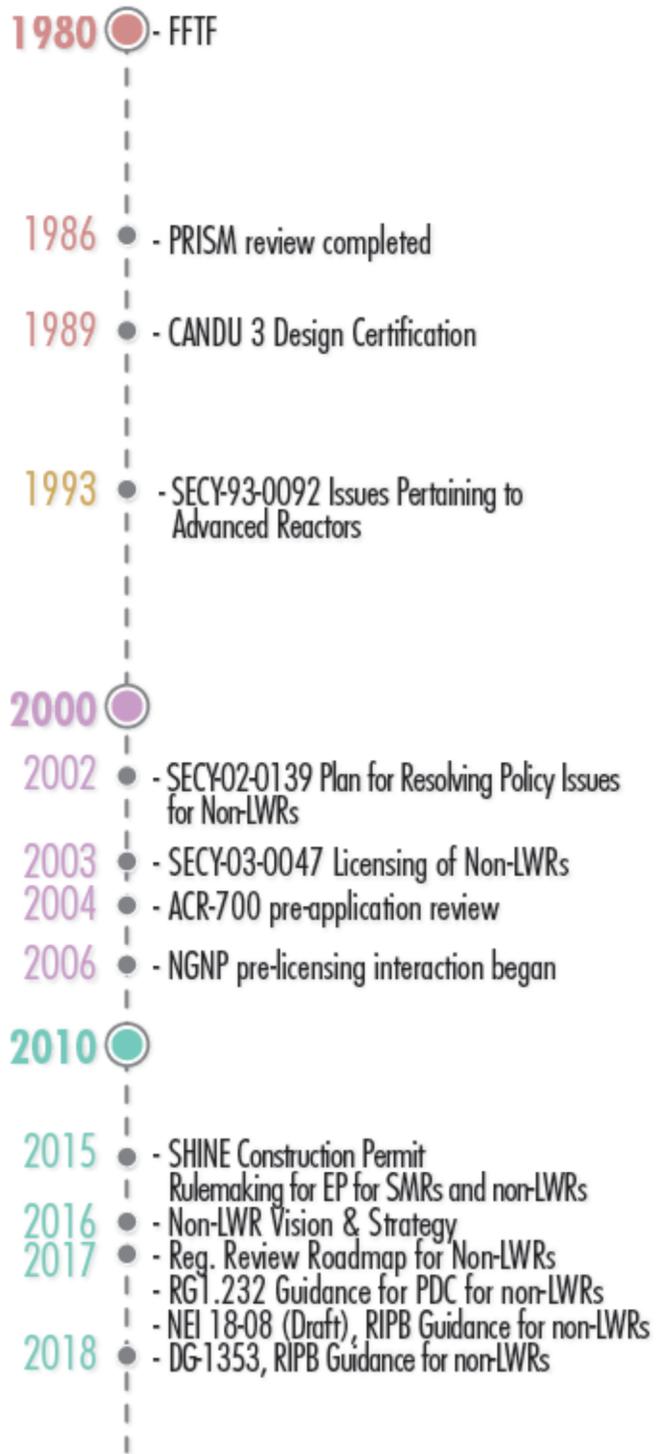


Figure 2.3 Timeline of Events in Nuclear Regulation and for Non-LWRs

3 NON-LWR REGULATORY AND LICENSING ISSUES

3.1 Gas Cooled Reactors

3.1.1 Introduction

Most gas cooled reactors use graphite as a neutron moderator and carbon dioxide or helium as a coolant. These types of reactors have been in operation since the 1950s, the first one being the Magnox reactor in 1956 in the United Kingdom (UK). Some advanced gas cooled reactors (Magnox successors) are still in operation today. High temperature gas cooled reactors (HTGRs) started with the 20 MWt Dragon test reactor in the UK and remain of interest since they can provide both efficient electricity and high-temperature process heat usable for various industrial applications. Major gas cooled reactors of different types are listed in Table 3.1.

Table 3.1 Gas Cooled Reactors Around the World

CO₂ Cooled, Graphite Moderated	CO₂ Cooled, Heavy Water Moderated
Magnox; UK, 1956-2015 (28 built)	Brennilis Nuclear Power Plant; France, 1967-1985
Advanced Gas Cooled Reactor (AGR); UK, 1962-present (Magnox successor, 15 built)	KS 150; Czechoslovakia, 1972-1979 (designed with Soviet Union/USSR)
UNGG reactor; France, 1956-1994 (10 built)	
He Cooled, Graphite Moderated Prismatic Block	He Cooled, Graphite Moderated Pebble Bed
Dragon; UK, 1964-1975	AVR; Germany, 1966-1988
Peach Bottom Atomic Power Station; U.S., 1967-1974	THTR-300; Germany, 1983-1989
Fort Saint Vrain Generating Station; U.S., 1979-1989	HTR-10; China, 2003-present
High-temperature engineering test reactor; Japan, 1999-present	HTR-PM; China, under construction
Gas Turbine Modular Helium Reactor (GT-MHR); U.S. (General Atomics design)	Pebble bed modular reactor; South Africa
Steam Cycle High-Temperature Gas Cooled Reactor (AREVA design)	He Cooled, Fast Spectrum
Next Generation Nuclear Plant (NGNP); U.S.	Energy Multiplier Module; U.S. (General Atomics design)

Following the Dragon reactor, the 115 MWt Peach Bottom Unit 1 (PBAPS) in the United States and the 46 MWt Arbeitsgemeinschaft Versuchsreaktor (AVR) in Germany were constructed for electricity generation. They were followed by the 842 MWt plant at Fort

St. Vrain (FSV) in the United States and the 750 MWt Thorium Hochtemperatur Reaktor (THTR) in Germany.

Modern gas cooled reactors use fuel particles (usually tristructural isotropic, i.e., TRISO) embedded either in prismatic graphite blocks or in (fist-sized) graphite pebbles. Small modular HTGR designs that allow the reactor to solely rely on inherent safety characteristics and design features instead of active engineered safety features are currently being pursued. For example, China is currently constructing two 250 MWt High Temperature Reactor - Pebble-Bed Modules (HTR-PM).

The following discussion focuses on HTGRs and the licensing and operating experience with PBAPS, FSV, and the Modular High-Temperature Gas Cooled Reactor (MHTGR). It then provides the current licensing and regulatory issues with gas cooled reactors.

3.1.2 Licensing of High Temperature Gas Cooled Reactors

Peach Bottom Atomic Power Station (PBAPS) Unit 1

Peach Bottom Unit 1 was the first HTGR built in the United States. It followed the first HTGR built at the UK Atomic Energy Authority Establishment by the Organization for Economic Cooperation and Development (OECD) High Temperature Reactor Project (commonly referred to as the Dragon project) [3-1]. The Peach Bottom HTGR was a 115 MWt, 40 MWe demonstration plant.^b It operated at 2.4 kPa primary system pressure with a core inlet temperature of 350°C and an outlet temperature of 750°C. The reactor went critical on March 3, 1966 and operated successfully until permanent reactor shutdown near the end of 1974. The goal of this plant was to demonstrate production of 538°C steam from a reactor with good neutron economy and high fuel burnup. Peach Bottom was closed when it completed its demonstration mission and was considered uneconomical because of its small size.

^b Section 202(1) and (2) of the Energy Reorganization Act of 1974 describe the term “demonstration nuclear reactor” as being “operated as part of the power generation facilities of an electric utility system, or when operated in any other manner for the purpose of demonstrating the suitability of commercial application of such a reactor.” The NRC does not have regulations specific to “demonstration reactor,” nor does it use this term in the licensing process. The AEC and the DOE did recognize “demonstration reactors” through its Cooperative Power Reactor Demonstration Program of 1955.

Fort Saint Vrain (FSV)



Figure 3.1 FSV Refueling Floor

rod pair; however, during the shutdown, numerous cracks were discovered in several steam generator main steam ring headers. The required repairs were determined by the PSC to be too extensive to justify continued operation, and operation was terminated on August 29, 1989.

Modular High Temperature Gas Cooled Reactor (MHTGR)

In the 1980s, HTGR designers at the German company INTERATOM (later Siemens), developed a new pebble bed design where the modified the reactor system design removed residual heat passively under all circumstances, and the need for active emergency core cooling systems was eliminated. The resulting design was called the HTR Module [3-2]. The HTR Module was not intended exclusively for electricity production; other possible missions were envisioned as well, including heat and power cogeneration, process heat and/or steam production, and district heating.

The U.S. modular HTGR concept began in 1984 when Congress challenged the HTGR industry to investigate the potential for using HTGR technology to develop a “simpler, safer” nuclear power plant design. The goal was to develop a passively safe HTGR plant that was also economically competitive. To maintain the coated-particle fuel temperatures below damage limits during passive decay heat removal, the core’s

As part of the U.S. Atomic Energy Commission (AEC) Power Reactor Demonstration Program, General Atomics constructed the reactor at Fort St. Vrain for the Public Service Company of Colorado (PSC). It was an HTGR that used Peach Bottom as a basis for the design. FSV was granted an operating license by the AEC on December 21, 1973; had its initial criticality on January 31, 1974; and went into commercial operation on July 1, 1979. The reactor operated at 842 MWt with an output of 330 MWe. High-temperature helium was used as the primary coolant to produce superheated and reheated steam at approximately 538°C. The helium entered the reactor at 404°C and left the reactor to the steam generator at 777°C. The reactor was contained within a prestressed concrete reactor vessel and used a prismatic block design for the fuel elements. The fuel used was a mixture of carbides of uranium and thorium with TRISO coatings (explained below). FSV remained in commercial operation for a little more than 10 years. Then, on August 18, 1989, the plant was shut down to repair a stuck control

physical size had to be limited; the maximum reactor power capacity was found to be about 200 MWt for a solid, cylindrical core geometry. However, this rating was projected to not be economically competitive for electric power generation. This judgment led to the development of an annular core concept to enable larger cores with increased power capacity. DOE and General Atomics developed a 350 MWt MHTGR using an annular prismatic block core arrangement. Licensing activities included preapplication interaction with the NRC and submittal of numerous documents including a Preliminary Safety Information Document [3-3].

Licensing Process and Considerations - PBAPS

The AEC's Division of Licensing and Regulation reviewed Philadelphia Electric Company's application for PBAPS. In a report to the Advisory Committee on Reactor Safeguards, it summarized the issues addressed to conclude that the proposed design provided reasonable assurance that the health and safety of the public will be adequately protected.

As part of the licensing review and interactions, significant deficiencies were identified, and modifications made to resolve the identified deficiencies. The report discussed the following aspects:

1. Design of fuel elements

The AEC review requested confirmation on several fuel characteristics and PBAPS made basic design changes to the fuel element and provided testing to demonstrate performance. The report concluded that the impervious graphite diffusion coefficient of $1 \times 10^{-6} \text{ cm}^2/\text{s}$ can be met, that the structural integrity against mechanical loads is adequate, that stress loads resulting from dimensional changes can be controlled by proper dimensional control, that the minimum specification for thermal conductivity of 15 Btu/hr/ft/ $^{\circ}\text{F}$ can be met and that the use of pyrolytic coatings on the fuel particles adequately prevents the mobility and improves the retention of fission products.

2. Fission product trapping

The AEC review required resolution of problems concerning the proposed fission product trapping system relating to proportional distribution of products in the fuel and various traps, the methods and materials to be used, the engineering arrangements to accommodate safe handling and storage, and the procedures for disposal of the trapped fission products. Design modifications provided several mechanisms to limit any fission products in the primary system. Also, the changed design assumed that each trap must be removable, with the external traps to be capable of operating for the entire plant life without replacement. Personnel access was provided where necessary within five days following shutdown.

3. Control systems

Uncertainties were identified relating to stability of control rod graphite under irradiation, possible effects of separation of the rod elements from their drives, and the requirement for a secondary shutdown system. A feasible control rod system design was provided with methods for indication of control rod separation and a backup shutdown system with adequate shutdown capability in the event of abnormal conditions, or loss of the normal control rods.

4. Core mechanical design

Assurance was obtained regarding adequate evaluation of the proposed core arrangement with respect to flow disturbances, lateral stability, interferences with control rod motion, and possible oscillatory movements.

5. Facility design

Three features were identified as requiring additional study to determine safety: a) provision of an emergency cooling system, b) precautions to prevent in-leakage to the core, and c) safeguards against accidents which would allow air to enter the primary system. An emergency cooling system was provided to protect against damaging results of core overtemperature due to decay heat. For addressing the effect of moisture on the graphite core materials, provisions were incorporated in the primary loop for rapid moisture detection and loop isolation. To prevent the possibility of rapid oxidation of graphite in the event of a primary coolant system rupture, the entire containment vessel, except for an isolated air room, was designed so that it is filled with a depleted oxygen atmosphere rather than air.

6. Safety analysis

The AEC required detailed identification of failures that could lead to on-site or off-site hazards resulting in safety analyses of the following:

- Incidents involving the reactor
 - reactivity accidents
 - loss of fission product barrier
 - loss of both main loops following rupture
- Incidents involving the fission product trapping system
 - loss of full cooling capacity
 - loss of system integrity
 - change in purge environment
- Safety of fuel handling
 - escape of fission products
 - stuck elements in large machine

- Plant behavior under abnormal conditions arising external to the plant
 - loss of power
 - earthquake, floods, landslides
 - fire
 - severe weather

- Environmental consequences of accidents
 - summary of accidents releasing activity to the containment
 - assumptions for dose calculations
 - discussion of consequences.

Licensing Process and Considerations - FSV

PSC applied for a construction permit and Class 104 License for FSV with the AEC in October 1966. They were granted an operating license in accordance with Section 104(b) of the Atomic Energy Act of 1954 (hereinafter referred to as the Act). The provisions of the Act allowed PSC ample leeway in its operation of FSV, and in fact, FSV was operated differently from other commercial nuclear power plants that were in operation at the same time. The Act allowed the AEC, “In issuing licenses under this subsection, the Commission shall impose the minimum amount of such regulations and terms of license as will permit the Commission to fulfill its obligations under this Act...”

When reviewing the operational experience at FSV as it may apply to the future licensing of new gas cooled reactor plants, it should be kept in mind that FSV was regulated not only in the past in a period of an evolving regulatory process but also under a somewhat different oversight structure than that used at its contemporary LWRs. Specifically, FSV was licensed under the provisions of *The Code of Federal Regulations*, Title 10—Energy, Part 50, Section 21, “Class 104 Licenses; for Medical Therapy and Research and Development Facilities” (10 CFR 50.21). The original AEC licensing officials commented verbally that FSV was considered by them to be a “research and development reactor that could be shut down immediately if there were any real safety problems.”

Prior to the 1970 Atomic Energy Act (AEA) amendment deleting the “practical value determination” previously required under Section 102 of the AEA, 10 CFR 50.21 had required that Class 104 licenses for the Power Reactor Demonstration Projects be converted to Class 103 licenses once the practical value determination had been made. This process required these nuclear power plants to anticipate the conversion of the license (to a Class 103 license). Following the amendment of the Act in 1970, the regulatory requirement to convert to a Class 103 license was dropped. Thus, the NRC allowed a rather wide latitude in regulatory interpretation of applicability to FSV consistent with the legal bases for its Class 104(b) operating license.

The period of FSV operation was one in which the NRC relied on the evolving regulatory process to deal with emerging safety issues such as the Browns Ferry fire, the Three Mile Island Action Plan, the need for environmental qualification of equipment

important to safety, the need to update and maintain current the design bases presented in the Final Safety Analysis Report (FSAR), standardization of technical specifications, etc. The Class 104(b) operating license issued for FSV and the NRC cognizant staff interpretation of the statutory basis for that license meant that FSV regulatory requirements were tailored to allow more flexibility than perhaps was afforded other contemporary plants that were licensed under Section 103 of the AEA.

Licensing Process and Considerations - MHTGR

NRC conducted and documented a preapplication safety evaluation of the MHTGR [3-4]. As stated in the safety evaluation, the general safety advantages of the MHTGR, like those of other HTGRs, were (1) its slow response to core heat-up events, because of the large heat capacity and low power density of the core and (2) the very high temperature that the fuel can sustain before the initiation of fission-product release (about 1600°C). Also, like other HTGRs, its major potential vulnerabilities derive from the need to protect metal components from continued exposure at elevated temperatures to hot helium during postulated transients and to prevent uncontrolled access of air or moisture to hot graphite and fuel particles. The safety of the MHTGR was, to a large extent, based on its proposed design features that utilized (1) passive removal of decay heat, (2) passive reactor shutdown with a modest temperature rise, and (3) high-integrity coated fuel particles. These fuel particles were to maintain their integrity during normal operation and at elevated temperatures under transient conditions or under conditions of chemical attack by steam and air and were proposed to function both as the initial fission-product barrier and primary reactor containment system. Accordingly, the staff's preapplication review of the MHTGR concentrated in those areas.

3.1.3 Operating Experience and Lessons Learned

The operating experience at Fort St. Vrain has been extensively studied and documented [3-5]. Lessons learned from both FSV and Peach Bottom (PBAPS) have been extensively studied [3-6] as part of the Next Generation Nuclear Plant (NGNP) program. The important operating experience and the lessons learned based on these documents is discussed below. Related experience with respect to materials and structural integrity for gas cooled (and sodium cooled) reactors is available in a recent review [3-7].

The lessons from the operating experiences at Peach Bottom, Unit 1 can be summarized as follows:

1. Fuel Design

Peach Bottom fuel kernels were originally coated with one layer of pyrolytic carbon (PyC), placed in prismatic annular elements, and used uranium and thorium carbides. After running for some time, the reactor coolant activity would continually increase because of failure of the fuel coating. The fuel was replaced with BISO fuel, which has an inner layer that acts as a buffer from recoiling fission products and an outer layer to retain the noble fission gases. BISO worked well at Peach Bottom but had shortcomings at higher temperatures. A layer of silicon carbide (SiC) was later added to the fuel coatings, now called TRISO fuel, which led to its use as fuel in FSV and showed much improved fission product retention.



Figure 3.2 TRISO Fuel Particle

2. Fission Product Trapping

A fission product trapping system was used at PBAPS to purify the primary coolant. Helium would enter the fission product trapping system after going through the fuel purge line in the reactor core. This helium would collect fission products from the core, such as krypton and tritium, and then travel through water- and Freon-cooled delay beds. From the delay beds, the helium would go through a series of fission product traps. One of the traps was a liquid nitrogen/charcoal trap, which would remove moisture, chemical impurities, argon, and krypton. Other components of the fission product trapping system were a dehydrator and an oxidizer. The fission product trapping system and the purge system worked efficiently but the trapping system did have a setback in retaining tritium. During reactor shutdown, the delay beds were regenerated and allowed to warm up. The adsorbed tritium gas was then desorbed. This happened because hydrogen is physically adsorbed on charcoal in increasing amounts with decreasing temperature. This is true for temperatures below 70°C. On the other hand, tritiated water molecules (a small portion of the total tritium) were permanently retained in the delay bed.

3. Oil leaks

Several oil leaks were found in the hydraulic components at static seal connections and piston seals associated with accumulators that provided the stored energy source for a reactor trip insertion. The piston seal leaks were determined to be caused by defects in the cylinder wall surface machining. During the initial preoperational testing of the

reactor control rod installation, the major problems encountered were in connection with sorting out external and internal oil leaks of the hydraulic system. A great deal of effort and time went into tracing and correcting this overall problem.

4. Oil Ingress in Compressors

The compressor circulates the helium through the primary system. Near the end of Core 1's life, there was concern about oil ingress. It was shown that the oil ingress originated in the compressor. More specifically, the ingress started from the oil demister/filter, which removes any oil vapor and oil mist from the discharge in the compressor, and the oil lubricant. Since the demister/filter was saturated with oil, it was speculated that oil was discharged into the reactor. The ingress of the oil lubricant in the main compressor would have originated from back diffusion through its helium buffer. Approximately 100 kg of oil entered the reactor. Evidence of the oil ingress was found by observing carbon deposits in the primary circuit metallic surfaces and persistent hydrogen and methane impurities. These deposits did not have any negative effects on the heat exchangers nor the metallurgy. Cesium plateout occurred near these carbon deposits. This ingress did cause a failure of the moisture monitor cells.

5. Containment

During power operations, the containment vessel for Peach Bottom had to be inerted. Several problems developed in this inert nitrogen space in the form of steam and water leaks. The plant had to be shut down to repair these leaks since the nitrogen area had to be de-inerted to ensure maintenance personnel's safety was not put at risk. Most of the leaks were repairable with the plant at full power, were it not for the nitrogen containment.

The reactor's design at Fort St. Vrain employed many of the same fundamental principles that formed the basis of PBAPS, but they were different. The most significant differences were that (1) PBAPS had a 40-MWe power rating versus a 330-MWe power rating for FSV; (2) PBAPS used a steel reactor vessel, and FSV had a prestressed concrete reactor vessel; (3) PBAPS used long, annular fuel elements with a solid graphite spine, and FSV used prismatic-block graphite fuel elements; and (4) PBAPS used electric-motor-driven helium circulators with oil-lubricated bearings, and FSV used steam and water turbine-driven helium circulators with water-lubricated bearings. The lessons learned from the operating experience at Fort St. Vrain can be summarized as follows:

1. Moisture Ingress

Moisture ingress caused a number of problems in the reactors, especially at FSV. Water from the circulator bearings in FSV was a source of moisture ingress into the primary system. Some of this moisture was absorbed by insulation or small cracks in welds. This moisture was then released later during various reactor operations. The moisture intrusion events that occurred at FSV show the effects of moisture ingress on

reactivity control. The effects of the moisture intrusion ranged from a change in reactivity to swelling and corrosion. Not only did the moisture affect the reactivity control system, but it also directly affected the reactivity by acting as a moderator and absorbing neutrons. Further, moisture can cause graphite to oxidize and other components to corrode. Moisture can also be out-gassed from graphite when it is heated, referred to as drying out. A related problem associated with the moisture ingress is the leaching of volatile chlorides from various sources within the reactor and their deposition throughout the primary system.

At FSV, a control rod cable broke and jammed in its guide tube during a test of the control rod drive. This was also attributed to moisture ingress. The reserve shutdown system at FSV was also degraded over a time period that probably exceeded two years. The degradation of the reserve shutdown system was traced back and was also shown to result from small amounts of moisture in contact with the shutdown material. This failure to completely guarantee a plant shutdown when required represents a significant safety hazard for plant operations. Other impurities could contaminate the primary helium coolant (such as air or oil) which could cause damage and/or corrosion.

2. Primary Coolant Flow issues

The primary coolant was shown to not always flow as predicted. In several cases, the helium would flow through the gaps within the core, known as bypass flow. Bypass flow has been shown to cause high stresses in the fuel elements and temperature fluctuations in the core. Helium impurities resulted in plateout on the heat transport surfaces and reduced their effectiveness. Bypass flow and helium impurities altered the efficiency of the heat exchangers and caused some walls to operate at temperatures that could cause material creep.

3. Fuel Performance and Fission Product Release

FSV experienced fuel cracks that propagated through two stacked fuel elements. Based on calculation models, it was concluded that high tensile stress and irradiation stresses resulted from incompatible peak factors in high stresses on the interregional faces of the two cracked fuel elements. Load tests indicated that even with cracks, the fuel elements' strength was essentially unaffected. Additionally, post irradiation examination of the cracked fuel element webs indicated that controlling key parameters, such as peaking factors, during plant operation would limit the cracking phenomenon (self-arresting).

Experience with fuel design, development, and manufacture for FSV provided the basis for the fuel technology used for the GT-MHR and guided subsequent fuel quality and performance improvements. For FSV, 2,448 hexagonal fuel elements, 7.1 million fuel compacts, and 26,600 kg of TRISO fuel particles were produced. The fuel was irradiated at temperatures greater than 1300°C to a maximum burnup in the fissile particles of 16% fissions per initial metal atom and to a maximum fast neutron fluence of

4.5×10^{25} n/m² with no evidence of significant in-service coating failure. FSV provided invaluable fuel performance, fission product release, and plateout data that have been used for validation of General Atomics' design methods.

4. Reserve Shutdown System (RSS) Operation

The Reserve Shutdown System at FSV experienced both inadvertent actuation of and failure to inject sufficient borated graphite balls (aka boron balls). The RSS was inadvertently actuated, and the Region 27 RSS boron balls were injected into the core. The licensee first observed a slight power tilt on the core outlet thermocouples. A follow-up investigation confirmed that the boron balls had been injected into the core. The boron balls were removed during an extended maintenance outage; the accidental injection of the borated graphite balls went undetected for almost a month, since there was no indication that the hopper door had failed or was open. The failure to deploy was due to moisture ingress which was discussed above. Instrumentation of the RSS to indicate when the system has actuated could have been helpful.

5. Graphite Dust

Fuel damage can cause graphite dust to form and transport fission products throughout the primary loop, especially for pebble bed reactors. Dust is also a concern for prismatic designed HTGRs due to movement and shifting of blocks during operations.

3.1.4 **Regulatory and Licensing Issues**

Reviewing the licensing and regulatory experience of PBAPS, FSV, and the MHTGR provided insights for reviewing later license applications. These insights are given below based on Appendix A of reference [3-5] and reference [3-8].

1. Safety Analysis Reports

An applicant needs to be clear in their safety analysis reports about (1) the selection of the principle design criteria (e.g., for the reactor core, cooling systems, reactivity control, decay heat removal, containment function) and how these bridge to and accommodate meeting the safety functions underlying the NRC's General Design Criteria (GDC), (2) the required seismic and environmental qualifications for the cooling systems and equipment to be relied upon as safety-related, and (3) the instrumentation and surveillance mechanisms that will be used to apply Technical Specifications to the equipment so as to satisfy the appropriate criteria for selecting Limiting Conditions for Operation and their associated Surveillance Requirements in 10 CFR 50.36(c)(2) and (3).

2. Industry Codes and Standards

Industry codes and standards should be applied in a consistent manner to the new and innovative designs; reasonable exceptions to be acknowledged, explained and

documented. Regulatory requirements of 10 CFR 50.55a and the guidance in applicable regulatory guides should be followed. At FSV, safety evaluations do not clearly document whether a Class 104(b) exception was being granted making it difficult to understand the reasoning behind the regulatory decisions.

3. Water Ingress

As discussed in Section 3.1.3, water ingress caused a number of significant issues at FSV. The possibility of getting water into the reactor primary system of an HTGR should be avoided. Also, ingress during shutdown conditions can pose high risk with subsequent moisture hide-out occurring due to water being absorbed into the graphite and in-vessel insulation. Many designs can operate with high-pressure helium in the primary side and low-pressure cooling water in the secondary side heat rejection system. Risk for water ingress during shutdown should be evaluated with attention to detection and monitoring of heat exchangers in a shutdown cooling system. If a helium purification system is used, then small helium lines providing purge gas should be designed to avoid both corrosion and stress corrosion cracking so as to preclude line blockages, carrying of corrosion debris to bearings or other moving parts, and tube cracking leading to small helium leaks or loss of purge flow to essential equipment such as the control rod drive mechanisms.

The detailed metallurgy on all boundaries of the primary coolant system including the safe shutdown cooling system, the helium purification system, and other systems needs to be well understood with regard to its ability to maintain integrity during normal operations and upset conditions. Fatigue, corrosion, creep rupture and all other threats to integrity must be addressed.

4. Level of Documentation and Experimental Data

Licensing of any new reactor should have a program based on Regulatory Guide 1.68 [3-9] requiring detailed documentation of how calculations are done, how measurements are made (with all uncertainties accounted for), and how analytical and experimental results are reconciled. The Class 104(b) license at FSV didn't require such documentation and didn't contribute to relevant data for reactor physics validation for future gas cooled reactors.

5. Fire Protection Program

The fire protection program and the mechanisms for responding to a fire to achieve hot and cold safe shutdown should be demonstrated to be consistent with regulatory requirements, and be reflected in selecting Principal Design Criteria, equipment qualification requirements, and the Technical Specifications. In the case of FSV, for complying with 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," the method of "safe shutdown" cooling following the design basis fire was defaulted to the FSV Design Basis Accident No. 1 (DBA-1) with fire water cooling the prestressed concrete reactor vessel to

maintain reactor vessel/primary containment integrity. In this scenario, fuel damage would occur during the resulting transient but would be contained to prevent and mitigate off-site doses resulting from the fuel damage and “safe shutdown” would not be achieved within the 72 hours as required in the regulations. This regulatory latitude is not expected to be granted for future plants.

Lastly, the preapplication safety review of the MHTGR [3-4] identified four policy issues that would require Commission review and guidance for resolution. These policy issues developed because of the differences in design from that of a light water reactor and the way in which the MHTGR proposed to accomplish the safety functions. These policy issues are discussed below.

1. Selection of Events that Must be Considered in the Design

In reviewing the MHTGR, the staff developed definitions for four event categories (ECs) corresponding to, in general, traditional LWR event categories:

EC-I	abnormal operating experiences
EC-II	design-basis accidents
EC-III	severe accidents
EC-IV	emergency planning basis events

The events in each of the categories were expected to be formalized with further information and reexamination.

2. Siting Source Term Calculation and Use

Mechanistic means of determining radioactive release to the environment in the performance of safety analyses were proposed. The development of a reactor design that supports consideration of mechanistic siting source term was a major departure from both the LWRs and earlier HTGR designs.

3. Adequacy of the Containment Concept

The MHTGR challenged the need for a containment structure. The basis of the proposal was that if a mechanistic analysis is used to calculate the release of radioactive material from the plant under all postulated events studied in ECs defined earlier, the dose guidelines of 10 CFR part 100 can be met without a containment structure. The staff argued that the establishment of a mechanistic source term could be a safety enhancement, even if a conventional containment becomes required for MHTGR.

4. Adequacy of Emergency Planning

The MHTGR preapplication proposed that the offsite emergency plan need not require elements of preplanned public notification, evacuation, or sheltering. This was based on

the contention that credible accidents in the MHTGR will not lead to offsite doses in excess of the protective action guidelines of the U.S. Environmental Protection Agency [3-10]. The NRC staff argued that this proposal will depend on, but may not necessarily directly follow from, the acceptance of the mechanistic source term.

3.2 Liquid Metal Cooled Reactors

3.2.1 Introduction

Liquid metal cooled reactors (LMRs) are fast-spectrum designs using sodium (or sodium-potassium) or lead (or lead-bismuth) as coolant. The reactors are further divided into two major plant configurations: loop designs--similar to light water reactors with secondary and tertiary coolant loops external to the reactor vessel--and pool designs in which the intermediate heat exchangers and primary pumps are contained within the reactor vessel. The initial interest in these reactors was to enable a fuel cycle that "bred" plutonium, thereby extending the available uranium resources severalfold. Later designs emphasized the inherent safety characteristics of these plants relative to LWRs and their ability to transmute actinides greatly reducing the volume of high-level nuclear waste. These types of reactors have been in operation since the 1950s, with EBR-I credited as being the first reactor to generate electricity. Major LMRs of different types with the country of origin and period of operation or start date (where applicable) are listed in Table 3.2^c.

Several innovative liquid metal reactor designs are being developed in the U.S. and internationally and deployment of large scale designs continues in Russia, China and India. The regulatory and licensing history in the U.S. is mainly drawn from the experience with the Fermi-I, EBR-II, SEFOR, and FFTF. The experience with EBR-I, CRBR (which was issued a construction permit but not built), along with the PRISM and Toshiba 4S design reviews also inform the discussion. All are discussed in the following section.

3.2.2 Licensing of Liquid Metal Reactors

EBR-I

EBR-I was a 1.4 MWt test reactor, which began operation in 1951. A loop design, it used electromagnetic pumps in the primary loop. It was cooled by a eutectic alloy of sodium-potassium (NaK) and used a metal fuel. The reactor had a small power conversion plant which was used to power four light bulbs in the first demonstration of electric power production by a nuclear reactor. The plant also demonstrated that a "breeding cycle" was feasible in that it produced Pu-239 at a rate at least equal to U-235 consumption. The plant suffered a partial core meltdown in 1955 during a series of

^c Older LMRs are reviewed in reference [3-11].

reactivity tests; the reactor was unstable under certain flow conditions. A second core was designed and installed which addressed the stability problems and was used until the program was terminated in 1963.

Table 3.2 Liquid Metal Cooled Reactors Around the World

Sodium Cooled Loop Reactors	Sodium Cooled Pool Reactors
EBR-I (Na-K); USA, 1951-1963	EBR-II; USA, 1961-1991
CRBR; USA, (some components built)	SAFR; USA
FFTF; USA, 1980-1993	PRISM; USA
Rhapsodie; France, 1967-1983	4S; Japan-USA
Fermi 1; USA, 1963-1972	PHENIX; France, 1973
SEFOR; USA, 1969-1972	SuperPhenix; France, 1985-1997
BOR-60; Russia, 1968	PFR; UK, 1974-1994
BN-350; Kazakhstan, 1972-1999	BN-600; Russia, 1980
Dounreay (Na-K); UK, 1959-1977	BN-800; Russia, 2015
JOYO; Japan, 1977	FBTR; India, 1985
MONJU; Japan, 1994	CEFR; China (Russian design), 2010
Lead Cooled Reactors	Lead Bismuth Cooled Reactor
SSTAR; USA	SVBR; Russia
BREST; Russia	

EBR-II

EBR-II was designed and built as a follow-on to the EBR-I project. The plant was a pool design with metal fuel and used centrifugal pumps augmented by a single electromagnetic pump.

The plant was rated at 62.5 MWt (20 MWe) and operated between 1961 and 1991, which was a record for sodium cooled reactors at that time. Designed and built by Argonne National Laboratory (as was EBR-I), the plant initially focused on further refining the “breeding cycle.” Later, it was used to demonstrate the inherent safety of the design in a series of safety tests in



Figure 3.3 EBR-II Control Room

which the plant was cooled by natural circulation without having a reactor trip. At the end of its life, EBR-II was used to test advanced metal fuel which was to be reprocessed using a pyro-processing scheme intended for an advanced integral fast reactor design. EBR-II was permanently shut down in 1994 after 30 years of operation which was a record for sodium cooled reactors at that time.

Fermi 1

The Enrico Fermi plant was a three-loop sodium cooled fast reactor designed for a nominal power of 300 MWt (100 MWe). It was operated by the Power Reactor Development Corporation; a consortium specifically formed to operate the plant). The power level was intended to ultimately be 430 MWt but was originally licensed to a power level of only 200 MWt (66 MWe). The fuel was a uranium-molybdenum alloy placed in 105 core (or driver) subassemblies and 531 radial blanket subassemblies. The design also included upper and lower blanket regions in each subassembly. The plant was designed with isolation valves in the primary loop so that loops could be isolated for maintenance while the plant continued operation with one loop shut down. Construction of the plant began in 1956 with initial criticality in 1963. The plant then underwent a series of increasing power steps which were terminated in 1966 when the plant suffered a partial meltdown of two subassemblies (at 31 MWt) when a zirconium plate tore loose from the lower plenum and blocked flow to two channels. The plant underwent a 42-month shutdown in which the damaged subassemblies were removed, and the fuel replaced (the zirconium plates were also removed). In 1972, the PRDC made the decision to shut down the reactor as the core was approaching end-of-life.

SEFOR

The Southwest Experimental Fast Oxide Reactor (SEFOR) was a 20 MWt reactor fueled with mixed oxide fuel. The plant was built and operated by a consortium that included Southwest Atomic Energy Associates (a utility group), General Electric, and German nuclear institutes. It was built as a test reactor to obtain data at operating conditions, and mainly to experimentally measure the Doppler reactivity coefficient which is an important contributor to the overall negative power coefficient in fast sodium cooled reactors. A two-loop design, the plant did not have a tertiary loop for power generation but instead used air cooled heat exchangers. The plant also included a 1.0 MWt auxiliary coolant loop to remove decay heat when the main loops were not operational. The plant operated from 1969 until 1972, when the test program was completed [3-12].

FFTF

The Fast Flux Test Facility (FFTF) was the last sodium cooled fast reactor built and operated in the United States. A loop design, the 400 MWt reactor operated for 10 years from 1982 to 1992. Unlike the EBR-II design, no tertiary loop with power generation capability was installed. Instead, heat was removed from the secondary sodium loop through a series of dump heat exchangers directly to the atmosphere.

Although its primary mission was irradiation of materials for advanced reactors, much data was obtained on safety tests conducted as part of the program, including natural circulation decay heat removal and transients and loss-of-primary-coolant flow without reactor trip. Although the plant was not licensed by the NRC, a review was conducted by the NRC and the Advisory Committee on Reactor Safeguards and a formal Safety Evaluation Report was written.

CRBR

The Clinch River Breeder Reactor (CRBR) was a 1000 MWt (350 MWe) reactor that was to be constructed and operated by Commonwealth Edison and the Tennessee Valley Authority under contract to the AEC (later ERDA and DOE). The plant was opposed by the Carter Administration and work was halted in 1979 but later revived in 1981 under the Reagan Administration. A safety evaluation report related to the application for a construction permit for the CRBR was issued in March 1983 (NUREG-0968) [3-13]. The review was limited to those subjects that could have affected the decision to issue a construction permit. A key part of the review, which was conducted taking into account the NRC standard review plan (NUREG-75/087) [3-14]), was a review of the proposed design criteria. Because of the extremely conservative nature of the principal design criteria, the staff concluded “that core disruptive accidents can and must be excluded from the design-basis accidents for the plant.” The project was expected to receive a construction permit in 1983, but work was stopped when the project was cancelled in October 1983. Memorandum of Findings, issued by the Atomic Safety Licensing Board in lieu of a construction permit in January 1984, resolved all outstanding issues regarding the construction permit. [3-13]

SAFR

After the cancellation of the CRBR, the Department of Energy pursued three other LMR designs: the Larger Scale Prototype Breeder (LSPB), SAFR (Sodium Advanced Fast Reactor) and PRISM (Power Reactor Innovative Small Module). SAFR consisted of four identical reactor modules which were coupled to four steam turbine-generator sets. The NRC staff was in the process of reviewing the SAFR conceptual design when the DOE terminated the work in September 1988. The NRC staff documented the review that had been done up to that time in NUREG-1369 [3-15]. The staff emphasized that the preliminary conclusions on all matters in the Safety Evaluation Report (SER) were not final and that the SER should be used with caution and no conclusions on the overall acceptability of the SAFR concept should be drawn from the SER.

PRISM

The PRISM design was the only one that was developed to the point that a safety review was conducted. The PRISM design evolved over the years and eventually included a number of variations ranging in power from 425 MWt to 1000 MWt, with the standard design being 840 MWt. [3-16] The reactor was a pool design using metallic fuel, with solid oxide fuel as a backup design. The design has different arrangements of

fuel, driver, and blanket elements depending on whether the core is optimized for breeding, actinide burning, plutonium burning, or long life (so-called breakeven). All designs have two intermediate heat exchangers which connect to a single steam generator. Depending on the reactor thermal output, two or three steam generators connect to a single turbine generator set for electricity production. Decay heat is normally removed through the turbine condenser, but the design has two additional decay heat removal systems: an auxiliary cooling system which removes decay heat through the steam generator to the atmosphere and a reactor vessel auxiliary coolant system (RVACS) which removes heat directly through the containment vessel to atmosphere. The RVACS is on continuously so there is always a heat loss to atmosphere in this design.

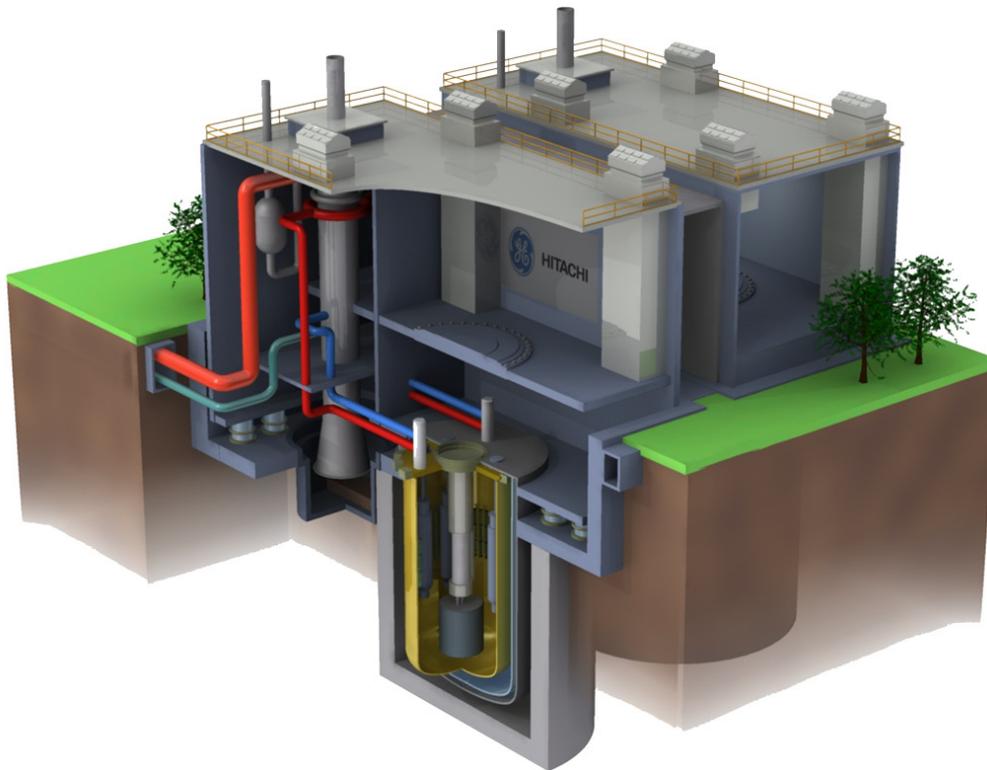


Figure 3.4 PRISM Plant

The NRC conducted a thorough review of the 475 MWt design between 1986 and 1994. A Preliminary Safety Information Document (PSID) was submitted by the Department of Energy in November 1986 for NRC review in accordance with NRC's "Statement of Policy for the Regulation of Advanced Nuclear Power Plants" [3-17] published in July 1986. The staff reviewed the PSID in accordance with NUREG-1226 [3-18], "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants" and documented the review in NUREG-1368 [3-19]. In that review, the NRC staff identified eight areas where the design deviated from LWR guidance. Of these eight areas, only one (Control room and Remote Shutdown Area Design) was considered not eligible for a departure from LWR regulations. The staff's

preliminary findings were reviewed with the ACRS. After the review, several revisions to the conceptual design were made and were also reviewed by the NRC staff. The staff, with ACRS concurrence, concluded that there were “no obvious impediments to licensing the PRISM design.”

Toshiba 4S

Toshiba proposed a small compact reactor with a long-life core for remote locations. The Toshiba 4S (Super-Safe, Small, and Simple) was a 30 MWt (10 MWe) pool type reactor designed for remote locations with small grids. The reactor was designed with a long-life core (30 years with no refueling) and utilized metallic fuel. A single loop, with electromagnetic pumps, was used for steam generation to a single turbine. Like the PRISM design, an RVACS and IRACS (Intermediate reactor auxiliary coolant system) are meant to operate passively for decay heat removal under loss-of-power conditions.

Safety analyses were carried out in accordance with NUREG-0800 [3-14]. The results, along with additional documentation, were submitted to the NRC, which held a series of preapplication public review meetings in 2007-2008. A key document was a set of principal design criteria developed by Toshiba based on the General Design Criteria for light water reactors. The NRC commented that it was the role of independent organizations to develop such criteria. In 2008, the NRC issued a policy statement on the licensing of advanced reactors which included several attributes that would assist in establishing the licensability of advanced reactor designs. From 2008 to 2013, Toshiba continued to submit a series of technical reports that responded to this policy statement and other NRC guidance. [3-20] NRC ceased its review of the Toshiba 4S in 2013 without issuing any review documents.

3.2.3 Operating Experience and Lessons Learned

There is over 50 years of operating experience with liquid metal cooled fast reactors in the U.S. The operating experience and lessons learned from EBR-II (30 years) and FFTF (10 years) have been extensively studied and documented. Although neither of these reactors were under NRC regulation, there were important operating experiences and lessons learned that informed the NRC review of more advanced designs [3-16]. In addition, there were safety lessons learned from the global experience [3-8], [3-21].

1. Metallic Fuel Performance

The testing and use of metallic fuels in EBR-II and FFTF provides most of the data underlying their safe usage in advanced designs such as PRISM and Toshiba 4S. Metallic fuels have several safety advantages including high thermal conductivity, ease of manufacture, and the ability to accommodate high burnups. The passive safety advantages of metallic fuels are discussed in Section 3.2.3.

EBR-II was designed to operate with metallic fuels. Over the thirty-year operating life, four different types of driver fuel were used with varying fuel composition (uranium

mixed with fission products in the early designs; uranium mixed with zirconium in the later designs) and cladding materials. In addition, numerous test pins (over 15,000) were irradiated providing a large data base of metallic fuel performance.

The FFTF was designed and built with an oxide fuel core. However, an extensive irradiation testing program of metallic fuels was conducted at the facility. The program focused on irradiation testing of metal fuel pins for use in sodium fast reactors, including PRISM. In addition, in the late 1980s there was a proposal to convert the FFTF core from oxide fuel to metal fuel. In order to support this re-design, a series of irradiation tests of metallic fuel pins that were prototypic of the driver fuel for the new core were completed. In general, the test demonstrated the expected performance of the fuel pins with no cladding breaches or other failures detected. The metal fuels were not part of the FFTF passive safety tests discussed below.

2. Decay Heat Removal through Natural Circulation Cooling

Both FFTF and EBR-II performed tests that demonstrated the ability of sodium cooled reactors to remove decay heat through natural circulation cooling.

FFTF conducted tests as part of its startup test program. The NRC in its review of the FFTF stated that its conclusion that the design complied with the intent of the existing rules depended on demonstration of the ability of the plant to remove decay heat through natural circulation. The tests included gathering in-core data from two instrumented fueled test assemblies. The tests demonstrated that natural circulation was a viable mechanism for decay heat removal and the data collected was subsequently used for validation of computer codes.

EBR-II conducted similar tests in the late 1980s. These tests also included instrumented in-core assemblies. The data generated in these tests was also used to validate codes used in the design of CRBR and in FFTF safety studies. Similar to FFTF, the tests demonstrated that natural circulation cooling would be established under a wide range of conditions.

Both the FFTF and EBR-II used centrifugal pumps which had significant coast down times which contributed to the successful transition to natural circulation. The PRISM and Toshiba 4S designs incorporate electromagnetic pumps which do not naturally have a coast down curve. In the 4S design, the coastdown is accomplished by having a dedicated motor generator set power the pump which duplicates the inertial coastdown of the centrifugal pumps.

3. Passive Shutdown/Negative Power Reactivity Coefficient

An important characteristic of sodium cooled fast reactors is a strong negative power coefficient with a strong negative temperature coefficient a major contributor. Both EBR-II and FFTF conducted tests in which the reactor was intentionally not scrammed under simulated accident conditions.

The EBR-II tests included the shutdown heat removal tests. EBR-II used metal fuel which operates closer to the coolant temperature than the oxide fuels used in FFTF. The tests included simulating a loss of electric power without scram, loss of primary heat sink, and an overcooling event (which leads to a power increase in this design without reactor scram). In all cases the reactor reached a state in which the fuel temperature was far below the limit for fuel damage.

FFTF carried out a similar unprotected series of tests in 1986 at partial powers up to 50%. The main purpose of the tests was to demonstrate the effectiveness of passive negative reactivity insertion through Gas Expansion Modules (GEMs) placed in the reflector region of the core. The GEMs introduce negative reactivity through the expansion of argon gas in the modules under loss of flow conditions (which decreases overall system pressure) thereby increasing neutron leakage in the reflector region. A second series of tests involved a reduced number of GEMs (three) and an attempt to determine the structural expansion feedback experimentally. The GEM system is incorporated in some PRISM designs.

4. Sodium Leaks and Sodium Water Interactions

The primary sodium coolant can be radioactive in this design. Therefore, loss of coolant in the primary system presents an additional hazard. The EBR-II was a pool design in which all the primary reactor components (pumps, heat exchanges, refueling systems, etc.) are contained in a large tank which has an inert cover gas (argon). The EBR-II tank was double walled, with an additional steel liner between the tank and the supporting structure. This design allowed the steel liner to serve as a guard system such that if a leak occurred the sodium level could not drop below the height of the core and decay heat removal system levels, thereby maintaining cooling under leak conditions. However, EBR-II did not incorporate passive air cooling of the guard vessel and instead relied on a set of small in-core heat exchangers for direct decay heat removal. The intermediate sodium is not radioactive, however, there is a possibility of energetic sodium water reactions in the steam generator if a leak occurs. To minimize this possibility, EBR-II incorporated a double walled tube design in the steam generator. This design proved to be highly successful, although there were other problems with the steam generator system. Guard vessels and double walled steam generator tubes have been incorporated into the PRISM and Toshiba 4S designs.

FFTF did not have a steam generation capability and used dump heat exchangers to the atmosphere as the heat sink. FFTF was a loop design so the primary components were not all located in a single large vessel. Instead, the components containing radioactive sodium were placed in cells inerted by nitrogen and lined with steel. As with EBR-II, all primary and secondary piping was constructed to high standards, in this case, ASME Boiler and Pressure Code. These standards led to a high reliability of the systems during the 10 years of power, with only one sodium leak near an auxiliary electromagnetic pump.

3.2.4 Regulatory and Licensing Issues

In 2012, Sandia National Laboratory led a Sodium Fast Reactor Safety and Licensing Research Plan [3-22] which proposed “potential research priorities for the Department of Energy with the intent of improving the licensability of the Sodium Fast Reactor (SFR).” The authors chose to only address metal fueled reactors since all the future designs being considered only use metal fuel. The report summarized the work of five expert panels that identified “potential safety related gaps in available information, data, and models” needed to support licensing of sodium fast reactors. The five areas and identified gaps are summarized below. The report did not address three other areas which impact licensability because it was felt they fell outside the primary scope of the project: instrumentation and control, in-service inspection and under-sodium viewing, and codes and standards. The report recommended that in all areas a structured knowledge management program was needed to effectively maintain and access the operational knowledge obtained during the U.S. fast reactor program prior to 1994.

1. Accident Sequences and Initiators

The expert panel concluded that there were “no major technology gaps” that would prevent the licensing of sodium cooled fast reactors as the design stayed “with known technology.” The panel did identify ten gaps in the knowledge base but of differing degrees, some associated with the advanced supercritical CO₂ power cycle that is under development for some designs. Other gaps were associated with special fuel cycles, such as minor actinide burning, that were not part of the earlier fast reactor programs. Of particular concern was the adequacy of the existing simulation codes (SAS-4A/SASSYS-1) as licensing tools since these codes were not developed to support a regulatory case.

2. Sodium Technology

This gap analysis did not address the performance of the primary or intermediate cooling loop components but focused on accident conditions caused by leaks in these systems and the ability to model the effects of leaks including sodium fires, gas production, and sodium interactions with concrete and drip liners. The panel looked at three broad accident areas, sodium leaks at high pressure, sodium leaks at low pressure, and leaks from the power conversion cycle (water or supercritical CO₂). Twenty-six gaps were identified and grouped into six topical areas. This panel did not directly address the issue of licensability but did identify a similar need for state-of-the-art computer codes to support the needed analysis in this area.

3. Fuels and Materials

The fuels and materials panel concluded that the knowledge base for sodium fast reactor fuels and structural materials (both in-core and ex-core) was “sufficient for designing and licensing of a sodium fast reactor” within the existing database (burnup of approximately 10%). The panel did raise some issues regarding the quality and

retrievability of the existing data. In agreement with the accident sequence panel, this panel pointed out that the database to support minor actinide burning is too weak to support licensing at this time. The panel identified twenty gaps in knowledge requiring additional research, but did not identify a need for improved computer codes for licensing tools

4. Source Term

The panel identified twenty gaps (grouped into five topical areas) in the mechanistic model of source terms for sodium cooled fast reactors. The panel determined without additional experimental data that the “mechanistic modeling of the source term would be judged by the experts as seriously deficient and potentially unreliable.” Although specific computer code recommendations were not made, some of the knowledge gaps identified by the panel were recommended to be addressed.

5. Codes and Models

The codes and models panel reviewed the sodium fast reactor computer analysis tools and their ability to support a license application. The group identified thirteen gaps which were consolidated into five topical areas. In its review, the group identified SAS4A/SASSYS-1 as a “central tool” and as “adequate to support these activities for licensing,” a conclusion not shared by the accident sequences panel. In addition, the panel noted that two codes: MELCOR(LMR) and LIFE-METAL required additional work to be suitable to support the licensing of sodium cooled fast reactors. It also recommended that all three codes be upgraded to improve performance on modern parallel computing platforms. Consistent with the other panels, this panel recommended an effort to preserve the existing knowledge base.

3.3 Heavy Water Reactors

3.3.1 Introduction

Although no heavy water reactor vendors are currently part of the advanced reactor stakeholders interacting with the NRC, it is still useful to consider NRC’s experience with them herein because they are very different from LWRs. However, the emphasis is on the regulatory and policy issues and not on the associated unique technical issues.

There have been two preapplication reviews of a heavy water power reactor. During the period from 1989 to 1995 the NRC reviewed documents from Atomic Energy Canada Limited Technologies, Inc. (AECLT) for the CANDU-3 reactor. During the period 2002-2005 there was a preapplication review of the AECLT ACR-700 design. Both reactor designs were based on the CANDU reactors that had been built, and successfully operated in Canada and other countries. The ACR-700 had one major difference from previous CANDUs in that light water, rather than heavy water, was used as the coolant while moderation outside the fuel channels was heavy water, as in all other CANDUs.

CANDU designs differ from LWRs not only because of the use of heavy water but also because the fuel elements are horizontal, in pressurized channels rather than a pressure vessel, and are removed and added while the reactor is operating. There were unique licensing issues, discussed below, that needed to be resolved at the time the applications were withdrawn. These issues were more important than the use of heavy water per se as the NRC licensed the research reactors at the National Institute of Standards and Technology and the Massachusetts Institute of Technology, the former using heavy water for both moderation and cooling and the latter for moderation in the reflector.

3.3.2 Regulatory and Licensing Issues

CANDU-3

The CANDU-3 preapplication activity was during the period when the NRC was creating advanced reactor policies (as discussed in Section 4.1). These were relevant although the staff did not consider CANDU-3 as truly an “advanced” reactor due to the fact that there were existing similar CANDUs in operation. The staff, very soon after AECLT said it wanted to submit an application for Design Certification, identified features that might present unique challenges, both technically and from a regulatory perspective. [3-23] Indeed, several years later these challenges were still before the NRC.

The staff documented the policy issues in a SECY [3-24] for this reactor along with those for several advanced reactors; namely, PRISM, MHTGR, and PIUS. These were issues where the applicant was proposing to deviate from current LWR guidance. For each issue, the staff provided the current LWR regulations, the pre-applicants position and the staff’s considerations and recommendations. The vendors’ responses to the draft report and the staff’s resolution where there was a disagreement are also documented in the SECY. There was no Safety Evaluation Report completed before AECLT withdrew the application.

The five CANDU-3 issues discussed in the SECY were:

- accident evaluation
- source term
- containment performance
- control room and remote shutdown area design
- positive void reactivity coefficient

The first four issues were common to all the reactors considered in the SECY and for CANDU-3 the technical issues could mostly be resolved by following guidance for LWRs, providing more information, and consideration of the licensing approach in Canada. For example, the accident evaluation would need to explain in the future how acceptance criteria were related to the probability of a particular accident.

The issue of the positive void reactivity coefficient was unique to CANDU-3 (and PRISM). This was a regulatory issue because General Design Criterion 11 requires that prompt inherent nuclear feedback should compensate for a rapid increase in reactivity. CANDU-3 had a very small negative power reactivity coefficient. However, the large positive void coefficient could lead to a damaging power excursion if there was a large-break loss-of-coolant accident and insufficient shutdown capability. The CANDU-3 was designed with two independent, diverse, fast acting, safety grade, shutdown systems to preclude this problem. The staff concluded that the positive void reactivity coefficient in itself would not disqualify the design, but the staff's concern would have to be addressed with more deterministic and probabilistic analysis to understand the consequences of voiding events.

Advanced CANDU Reactor 700 MWe (ACR-700)

The preapplication submittals for the ACR-700 were to address 13 focus topics. The staff in their safety assessment report [3-25], based on AECLT documents and face-to-face meetings, concluded that the applicant will need to pursue many technical issues to reach satisfactory conclusions for design certification and that the "policy, regulatory, and technical issues involved are complex." The potential policy and regulatory issues were far fewer than the technical issues and in only several of the focus topics as discussed below. Although they were identified, there was no need to resolve the issues due to AECLT deciding not to submit an application for design certification.

The focus topics for which there were no significant regulatory or policy issues were:

- Class 1 pressure boundary design
- Computer codes and validation adequacy
- Severe accident definition and adequacy of supporting research and development
- Design philosophy and safety-related systems
- Distributed control systems and safety critical software
- Preparation of standard design certification docketing
- ACR technology base

The other focus topics with regulatory/policy issues are discussed below:

1. Design-Basis Accidents and Acceptance Criteria

For certain loss-of-coolant accidents (LOCAs), considered to be design-basis, the vendor showed that there is localized fuel/cladding melting. However, 10 CFR 50.46 fuel performance acceptance criteria do not allow for cladding oxygen embrittlement, or fuel or clad melting, to occur during a LOCA. For this to be acceptable, the Commission would have to allow for an exemption to the rule.

Another staff concern was that the design-basis accident source terms were "major deviations from the 10 CFR 50.34 requirements." AECLT had used a mechanistic

source term and the staff thought that the issue would “require Commission consultation.”

2. Canadian Design Codes and Quality Assurance Standards

The ACR-700 had been designed using Canadian codes and standards. In general, vendors use codes and standards frequently for their own benefit and not to fulfill a regulatory requirement. The NRC allows for international codes and standards to be used after review by the staff. However, one policy issue identified by the staff would be the use of the approval of a code or standard by the Canadian regulatory agency without NRC review. The NRC requirements for quality assurance (QA) are spelled out in Appendix B to 10 CFR 50. After an extensive comparison of these requirements with those in the equivalent Canadian QA standards it was determined that there was no need for further Commission guidance.

3. On-Power Fueling

The inserting and removing of fuel assemblies during normal reactor operation is unique to CANDU designs. AECLT analysis of accidents during fueling provides the probability of fuel damage in the channel being refueled. “The policy issue involves application of the Commission’s safety goals and the principle of defense-in-depth to these initiating events. These events pose little risk of damage to a significant portion of the core or of a large early release of radioactive material. However, these fuel damage event sequences may have higher frequencies than more severe fuel damage event sequences evaluated for existing licensed LWRs. The staff recommends developing acceptance criteria for these potential limited fuel damage events provided that the overall risk is equivalent in terms of frequency times consequences.”



Figure 3.5 Darlington CANDU Plant

event sequences evaluated for existing licensed LWRs. The staff recommends developing acceptance criteria for these potential limited fuel damage events provided that the overall risk is equivalent in terms of frequency times consequences.”

4. Confirmation of Negative Void Reactivity

The ACR-700 had considered several fuel designs but the final design was to have negative void feedback. This was to be determined by calculations and experiments in

a critical facility, which were continuing into the future. (As noted above for the CANDU-3, a reactor with a positive void reactivity coefficient can be licensed.) The only regulatory issue identified by the staff was that “there may be a need for more specific guidance on the confidence levels to be considered in assessing the impact of reactivity coefficient uncertainties in the contexts of (1) establishing compliance [with GDC 11] and (2) best-estimate accident analysis methods.”

5. ACR Probabilistic Risk Assessment Methodology

The potential policy issue in this topic relates to how to apply the Commission’s risk objectives for advanced reactors. The staff provided the specific objectives that needed to be considered (e.g., the core damage frequency risk objective).

6. CANFLEX Fuel Design

The final fuel design for the ACR-700 was still undergoing fuel qualification and hence, the staff’s review and approval of the design were not expected to be possible as early as during design certification. The staff identified this as a potential policy issue requiring Commission guidance.

3.4 Liquid-Fuel Reactors

3.4.1 Introduction

Liquid-fuel reactors for power generation, most likely using molten salts, have not yet been considered for licensing by the NRC. However, in the early days of the NRC, there were twelve aqueous homogeneous reactors with thermal power levels of from 5 W to 50 kW licensed [3-26]. More recently, there have been two licensing activities for aqueous liquid fuel for isotope generation and this experience might prove to be relevant to liquid-fuel molten salt reactors. It may also be applicable to “micro-reactors” in the future as the liquid fuel was at low thermal power. One of these experiences was the 220 kWt Aqueous Homogeneous Reactor (AHR) for which Babcock & Wilcox Technical Services Group submitted preapplication material in 2010. The second was the application for a construction permit in 2013 by SHINE Medical Technologies for an accelerator with an aqueous target. In both cases an aqueous solution of uranyl sulfate with low-enriched uranium (less than 20 weight percent U-235) was to be used. The objective of these projects was primarily to generate the fission product Mo-99, an extremely useful medical isotope, which would be separated from the fuel at the plant site.

3.4.2 Licensing of Liquid-Fuel Reactors

When the AHR design was submitted, it was recognized that the existing NRC guidance for format and content of a safety analysis report for non-power reactors, and the corresponding standard review plan (NUREG-1537 [3-27]), would only be partially applicable to a liquid-fuel design. Hence, a panel was convened to study the technical

issues associated with normal operation and potential transients and accidents of this design. The panel produced the requisite AHR licensing guidance for three chapters of NUREG-1537: Chapter 4, Reactor Description, Chapter 5, Reactor Coolant Systems, and Chapter 13, Accident Analysis. This guidance takes into account the unique features of an AHR such as the fuel being in solution; the fission product barriers being the vessel and attached systems; the production and release of radiolytic and fission product gases and their impact on operations and their control by a gas management system; and the movement of fuel into and out of the reactor vessel [3-28].

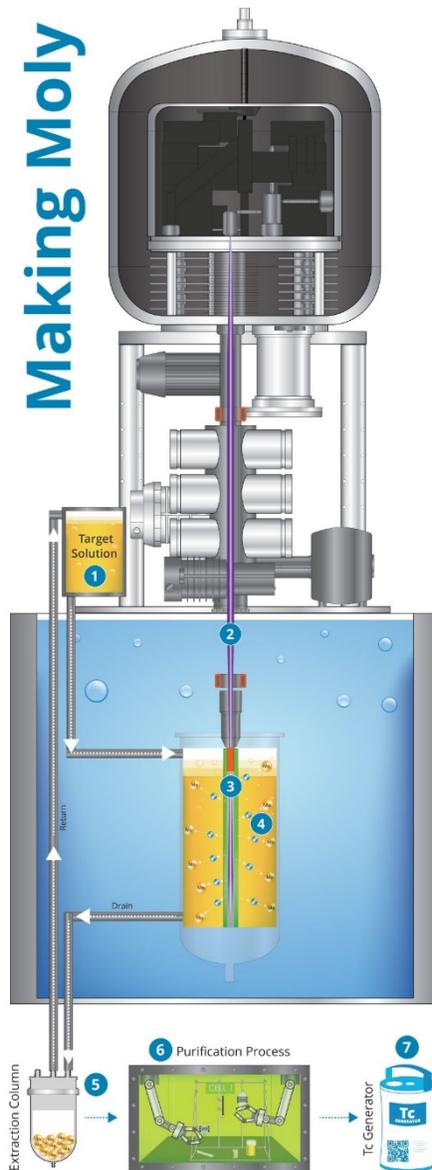


Figure 3.6 SHINE Isotope Generator

The isotope production aspects of the plant were not addressed in the three chapters cited above. This was addressed when NUREG-1537 was further modified and significant changes were made to the following chapters: Chapter 6, Engineered Safety Features, Chapter 7, Reactor Instrumentation, Chapter 12, Conduct of and Operations, and Chapter 14, Technical Specifications. Section 12.12 containing guidance for environmental reports was eliminated and, because of the importance of the subject, a new Chapter 19, Environmental Review was added.

The result was the Interim Staff Guidance (ISG) for “Radioisotope Production Facilities and Aqueous Homogenous Reactors” [3-29]. Specifically, the ISG covered any non-power reactor (including an AHR) as a utilization facility and the following types of production facilities used for the separation of byproduct material from fuel:

- “targets irradiated in a non-power reactor
- the core of an AHR
- the content of a subcritical multiplier solution tank or reactor vessel containing special nuclear material and fission products resulting from incident accelerator-generated neutrons.”

The ISG was applicable to the SHINE facility which applied for its construction permit after it was written. Although the reaction vessel (the accelerator target in the facility) is not a nuclear reactor, “its safety analysis must consider phenomena analogous to those of an AHR. The reaction vessel can achieve relatively high power levels from the fission process. The production of reasonable and practical quantities of radioisotopes on a commercial scale may require operating power levels on the order of 50 to 75 kilowatts. While the

assembly is maintained subcritical, it will have to be operated very much like an AHR with controls for managing temperature and pressure of the fuel solution, maintaining radiolytic gases at safe levels, and containing fission products, some of which are volatile, in the solution. It will have to have the same protective structures, systems, and components that are required for an AHR.” Many of the hazards and concerns associated with AHRs addressed in the ISG also apply to the SHINE reaction vessel subcritical neutron multiplier and to the associated radioisotope separation and purification processes involved in the radioisotope production process.

As is true of all ISGs, the resulting NUREG-1537 provides guidance but is not a regulation. Alternate approaches are possible provided they can be found acceptable by the NRC. The ISG does, however, state that an application must be prepared and submitted in accordance with several regulations (e.g., 10 CFR 50.34(b) Final Safety Analysis Report).

3.4.3 Regulatory and Licensing Issues

The AHR never submitted a license application so the NRC never did a formal review of the reactor. The SHINE facility received a construction permit after the NRC staff issued a Safety Evaluation Report (SER) [3-30] and after an Environmental Impact Statement was written by the NRC. The review was based on the ISG for NUREG-1537 taking into account what is needed for a construction permit as opposed to an operating license. The SER lists a few construction permit conditions that mostly relate to more information about criticality safety. The staff and the ACRS also received commitments from the vendor for more information on many other subjects. These topics are to be discussed in the Final Safety Analysis Report, submitted for an operating license, which will be more detailed than the Preliminary Safety Analysis Report submitted for the construction permit.

Two relevant research projects were mentioned in the SER: irradiation and corrosion testing at Oak Ridge National Laboratory to study mechanical performance of materials; and studies at Argonne National Laboratory to ensure precipitation of uranyl peroxide in the target solution will not occur.

The review by the ACRS highlighted two topics that needed further input from the vendor: One was the facility’s ability to withstand potential aircraft impact and the other was the facility’s layup capability. The ACRS noted that, “nuclear chemical processing facilities need to have built-in capability to support layup following unexpected process interruptions. It must be possible to stop the process, safely remove materials within the system, clean the system, and place it in a safe condition for an extended period in a way that does not challenge the facility piping systems and chemical reactors.”

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4 TRANSITION TO THE PRESENT AND BEYOND

Regulation of nuclear power plants has advanced significantly in recent decades forming the current regulatory paradigm for non-LWRs. Following the accident at Three Mile Island (TMI), the TMI Action Plan [4-1] was developed making improvements in many aspects of regulation. The ground-breaking Reactor Safety Study [4-2] addressing the risk of operation was followed by Individual Plant Examinations (IPEs) for internal and external events [4-3, 4-4], and severe accident risk studies (e.g., NUREG-1150 [4-5]). The Probabilistic Risk Assessment (PRA) studies completed as part of the IPEs and their application in various regulatory issues eventually led to risk-informed and performance-based regulatory concepts being increasingly used in nuclear power plant regulation.

Although the emphasis was initially on LWR regulation, the Commission issued a number of Policy Statements that will influence the regulation of non-LWRs. These were on Advanced Reactors [4-6], Safety Goals [4-7], Severe Accidents [4-8], and PRA [4-9]. They are integral to the regulatory process for non-LWRs currently in development, as discussed in this chapter. Also discussed in this chapter are influences from outside the NRC, namely from industry in the form of the Licensing Modernization Project and from the international community.

4.1 Advanced Reactor Policy

On July 8, 1986, NRC published “Regulation of Advanced Nuclear Power Plants; Statement of Policy.” [4-6] The latest revision to this policy statement was published on October 14, 2008 [4-10] to include consideration of security and continues to provide the overall guidance of all activities relating to advanced nuclear power plants.

The stated primary objectives of the policy statement are:

- “Encourage earliest possible interaction of applicant, vendors, and government agencies, with the NRC;
- Provide all interested parties, including the public, with the Commission’s views concerning the desired characteristics of the advanced reactor designs; and
- Express the Commission’s intent to issue timely comment on the implications of such designs for safety and the regulatory process.”

The Commission defined its expectation for advanced reactors as part of the policy statement: “Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the environment and public health and safety and the common defense and security that is required for current generation LWRs. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.”

The policy statement also addresses the attributes that could assist in establishing the acceptability or licensability of a proposed advanced reactor design, and therefore should be considered in advanced designs. In addition, the Commission expects that the safety features of these advanced reactor designs will be complemented by the operational program for emergency planning (EP). This EP operational program, in turn, must be demonstrated by inspections, tests, analyses, and acceptance criteria to ensure effective implementation of established measures. The Commission also expects that advanced reactor designs will comply with the Commission's Safety Goal Policy statement [4-11] and the policy statement on conversion to the metric measurement system [4-12].

Details about the development and utilization of the Policy Statement on the regulation of advanced reactors can be obtained in NUREG-1226 [4-13], published following the issuance of the original policy in 1986. NUREG-1226 discusses the staff's plans for utilization and implementation of the guidelines contained in the policy statement, including staff information needs and the approach to be used in the review of advanced reactor concepts. These plans have, in general, been followed and provide useful background information.

Considering the provisions of the policy statements and the related policies and regulations, the following guidance applies in seven areas, as discussed in NUREG-1226:

Information Needs for Review of Advanced Reactor Concepts

The information to be provided for a review of an advanced reactor concept includes:

- description of the plant design and its proposed design, safety, and licensing criteria, including analysis of major accident scenarios demonstrating acceptable plant response (see Section 4.2 for further discussion of design criteria),
- probabilistic risk analysis of the plant following applicable standards and guidelines,
- description of those applicant sponsored research and development (R&D) programs considered necessary to support development and licensing of the design.

The staff evaluation should delineate the key safety issues, provide assessment of the R&D programs and the applicable licensing criteria, and identify any impediment to the licensing of the design, if applicable.

Enhanced Margins of Safety Over Current Generation LWRs

The policy statement states that enhanced margins of safety are expected in these designs compared to current generation of LWRs. The staff's evaluation is expected to be based on a judgment of the design addressing:

- the extent to which the designs incorporate those attributes listed as desirable in the policy statement,
- the uncertainties associated with the safety analysis and supporting base technology for the designs,
- the extent to which margins and defense-in-depth are employed to account for these uncertainties,
- the capability and margin included in the design to prevent and mitigate severe accidents, including compliance with the Commission's severe accident and safety goal policies, and
- the previous operating experience, existing technology and proposed R&D supporting the design.

Defense-in-Depth

The staff's position has been that when considering reactor types for which there is significantly less design, construction, and operating experience, the use of engineering judgment and defense-in-depth philosophy is an essential element in accounting for uncertainties in the design. The application of defense-in-depth may take various forms, such as:

- requirements to prevent accidents, such as high reliability, redundancy and/or diversity in systems, structures and components,
- requirements to mitigate accidents, such as long response times, multiple barriers, or safety systems,
- requirements to contain radioactive materials.

Probabilistic Risk Assessment

The Commission's Severe Accident Policy requires completion of a probabilistic risk assessment (PRA) at the conceptual design stage and consideration of the severe accident vulnerabilities that the PRA exposes, along with the insights that it may add to the assurance that there is no undue risk to public health and safety. The Commission's Safety Goal Policy establishes goals that broadly define an acceptable

level of radiological risks to the public from nuclear power plant operation and are expected to be used, whenever appropriate.

The conduct of the PRA should be consistent with the PRA standards being developed and utilization of the PRA should follow the Commission's PRA Policy Statement. In general, PRAs performed for advanced reactor concepts should cover the whole plant, should address internal and external events as well as various plant operating states (full power, low power, refueling, etc.), and should confirm the bases for component and system selections, confirm the adequacy of overall plant design, be used to identify and correct any areas of high risk, and confirm the adequacy of plant response to severe accidents and mitigation measures. Any PRA must also estimate and factor in the uncertainties associated with it. These uncertainties must be factored into decisions which utilize PRA results.

Supporting Technology

The Advanced Reactor Policy statement addresses the role of supporting technology on several aspects requiring advanced reactor designers to provide information on their applications on them. The use of supporting technologies in the areas of operating experience, technology development, foreign information and data, and prototype testing has been discussed in NUREG-1226 and is summarized below.

The available sources of operating experience should be used wherever possible as they provide the most direct, least expensive and preferred means of demonstration of licensability of reactor concepts. Advanced design, at the conceptual design stage, is expected to develop a "technology development plan" or equivalent documentation to present the scientific and engineering data that will be developed to support the design and safety analysis of the advanced reactor concept. As stated, the scientific and engineering data could include laboratory research, component development and testing, verification during plant preoperational testing or startup, periodic testing and/or inspection during plant operation, and the use of a reactor prototype test.

The experience base associated with advanced reactor concepts being less than what it is for LWRs, the use of applicable foreign reactor experience is acceptable. The Advanced Reactor Policy statement does not require a priori that a prototype reactor be constructed and operated; however, it does state that "The Commission favors the use of prototypical demonstration facilities as an acceptable way of resolving many safety related issues." Accordingly, advanced reactor designers are expected, from the conceptual design stage, to describe their plans for the construction, testing and operation of a prototype plant to support design certification.

Industry Codes and Standards

Over the years a large body of industry codes and standards have been developed and used for licensing LWRs. The use of these standards, wherever applicable, is encouraged in advanced reactor designs rather than proposing specialized unique

approaches. As advanced reactor designs progress, additional committees are expected to be formed providing codes and standards that may uniquely be needed for these designs.

Treatment of Sabotage

The Commission's Severe Accident Policy statement [4-8] recognizes sabotage as a contributor to severe accidents and accordingly, advanced reactors will be required to analyze and address these risks. They are expected to be treated as special considerations in the design and in the operating procedures developed for the plant. Advanced reactor designers are expected to address, from the conceptual design stage, the advantages and disadvantages their design provides in protection from insider and outsider sabotage.

4.2 Principal Design Criteria

4.2.1 Background

The Code of Federal Regulations (CFR) requires applications for a construction permit, design certification, combined license, standard design approval, or manufacturing license to include principal design criteria (PDC) for the facility. Regulatory Guide 1.232, Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors [4-14], provides recent guidance for developing PDC for non-LWRs.

The PDC for light water power reactors are derived from the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50. The GDC requirements are a key part of the regulatory requirements for LWRs and they support the design of the current nuclear power plants. Appendix A to 10 CFR 50 states:

“These General Design Criteria establish minimum requirements for the principal design criteria for water cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.”

The GDC in 10 CFR Part 50 Appendix A recognize that they are not regulatory requirements for non-LWRs but provide guidance for defining the PDC for non-LWR designs. Recognizing the specific needs for non-LWRs, NRC staff developed RG 1.232 to provide insights into staff's views on how the GDC could be interpreted to address non-LWR design features.

4.2.2 Design Criteria for Non-LWRs

RG 1.232 presents non-LWR design criteria addressing two specific design concepts, sodium cooled fast reactors (SFRs) and modular high temperature gas cooled reactors

(MHTGRs), as well as general advanced reactors. The criteria are generally applicable to six different types of non-LWR technologies (SFRs, lead cooled fast reactors, gas cooled fast reactors, MHTGRs, fluoride-salt high-temperature reactors, and liquid-fuel molten salt reactors). Applicants/designers may use the advanced reactor design criteria (ARDC) in Appendix A of RG 1.232 to develop all or part of the principal design criteria and may choose among the ARDC, SFR-DC (Appendix B), or MHTGR-DC (Appendix C) to develop each PDC. Applicants/designers may also develop entirely new PDC as needed to address unique design features in their respective designs.

The criteria are presented in tabular form in six sections as in 10 CFR Part 50 Appendix A:

- Section I Overall Requirements
- Section II Multiple Barriers
- Section III Reactivity Control
- Section IV Fluid Systems
- Section V Reactor Containment
- Section VI Fuel and Radioactivity Control

Each criterion of the GDC in 10 CFR Part 50 Appendix A was considered for its applicability to non-LWR designs and the applicable criterion (modified, as required) was provided. NRC staff's rationale for the adaptations are included (in a separate column). In many cases, the rationale refers to the changes made to the language of the GDC. If no changes are needed, it is denoted as "Same as GDC."

Examples of the way the criteria are presented in RG 1.232 are shown in Table 4.1 for Section II, Multiple Barriers. To fully understand the rationale and the changes made to the GDC in 10 CFR Part 50 Appendix A, the reader should refer also to the GDC.

4.2.3 Development of RG 1.232

The development of RG 1.232 is also discussed in the guide. It was a joint initiative of the NRC and the U.S. Department of Energy (DOE) and was carried out in two phases. Phase 1 of the DOE-NRC initiative consisted of reviews and evaluations of applicable technical information and an assessment of the existing GDC to determine their applicability to non-LWR designs. As part of the assessment DOE proposed a set of ARDC which could serve the same purpose for non-LWRs as the GDC serve for LWRs. In addition to the technology-inclusive ARDC, DOE proposed two sets of technology-specific design criteria; one for SFRs and one for MHTGRs. DOE developed the technology-specific design criteria to demonstrate how the GDC could be adapted to specific technologies in which there was some level of maturity and documented design information available. [4-15]

In Phase 2 of the initiative, NRC developed its own version of the ARDC, SFR-DC, and MHTGR-DC. NRC assembled a multidisciplinary team to review the Phase 1 DOE report, other pertinent references and NRC documents and prepared the final version following public comments and discussions.

Table 4.1 Example Criteria Presented in RG 1.232

A. Example from Advanced Reactor Design Criteria		
Criterion	ARDC Title and Content	NRC Rationale for Adaptation to GDC
15	<i>Reactor coolant system design.</i> The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	“Reactor coolant pressure boundary” has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.
B. Example from Sodium Cooled Fast Reactor Design Criteria		
Criterion	SFR-DC Title and Content	NRC Rationale for Adaptation to GDC
15	<i>Primary coolant system design.</i> The primary coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the primary coolant boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	“Reactor coolant pressure boundary” has been relabeled as “primary coolant boundary” to conform to standard terms used in the liquid metal reactor industry. The use of the term “primary” indicates that the SFR-DC are applicable only to the primary cooling system, not the intermediate cooling system. The cover gas boundary is included as part of the primary coolant boundary.

4.2.4 Applicability, Intended Use, and Key Assumptions

As stated in RG 1.232, applicants may use this RG to develop all or part of the PDC and are free to choose among the ARDC, SFR-DC, or MHTGR-DC to develop each PDC after considering the underlying safety basis for the criterion and evaluating the rationale for the adaptation described in the RG. In each case, it is the responsibility of the designer or applicant to provide not only the PDC for the design but also supporting information that justifies to the NRC how the design meets the PDC submitted, and how the PDC demonstrate adequate assurance of safety. In instances where a GDC or non-LWR design criterion PDC is not proposed, the designer/applicant must provide a basis and justify the omission from a safety perspective. The key assumptions in developing the non-LWR design criteria are provided in RG 1.232.

4.3 Additional NRC Statements on Advanced Reactor Policy

NRC staff policy relating to non-LWR issues are issued as SECY documents for the benefit of the Commission (and stakeholders) and frequently are followed by Staff Requirements Memoranda (SRMs) from the Commission. Both provide useful insights on staff activities and Commission thinking on policy issues. In the following some of the documents most important to non-LWRs are discussed.

SECY-93-0092 [4-16], issued in 1993, is based on the preliminary review of the advanced reactors, PRISM, MHTGR, and PIUS, and the CANDU 3 design. In it, the NRC staff identified eight issues for which departures from current regulations should be considered. The staff presented a summary of the issues, current LWR regulations, pre-applicant positions, staff considerations, and recommendations for staff action. The eight issues were:

1. How should appropriate event categories, associated frequency ranges, and evaluation criteria for events that will be used to assess the safety of the advanced reactors and CANDU 3 designs be identified?
2. Should mechanistic source terms be developed in order to evaluate the proposed designs?
3. Should the proposed advanced and CANDU 3 reactor designs be allowed to employ alternative approaches to traditional “essentially leak-tight” containment structures provided for the control of fission product release to the environment?
4. Should advanced reactors with passive design safety features be able to reduce emergency planning zones and requirements?
5. Should the NRC accept a reactivity control system design that has no control rods?
6. Should advanced reactor designs be allowed to operate with a staffing complement that is less than currently required by the LWR regulations?
7. Should advanced reactor designs that rely on a single completely passive, safety-related residual heat removal system be acceptable?
8. Should a design in which the overall inherent reliability tends to increase under specific conditions or accidents be acceptable?

In SRM-SECY-93-0092 [4-16], the Commission agreed to the staff’s positions, except for issue 4, emergency planning. The Commission stated that it was premature to reach a conclusion on emergency planning for advanced reactors. It asked the staff to use existing regulatory requirements for ongoing review purposes. In SRM-SECY-15-0077, the Commission initiated a rulemaking, as discussed below.

In SECY-02-0139 [4-17], the NRC staff identified seven issues with policy implications resulting from the preapplication activities on non-LWR designs of the Pebble Bed Modular Reactor and the Gas Turbine-Modular Helium Reactor. These issues were considered applicable for future non-LWR designs and recommended positions for resolving the issues were addressed in SECY-03-0047 [4-18]. The Commission approved the recommendations of the following four issues:

1. Should specific defense-in-depth attributes be defined for non-LWRs?
2. To what extent should a probabilistic approach be used to establish the plant licensing basis?
3. Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability?
4. Under what conditions, if any, can emergency planning zones be reduced, including a reduction to the site exclusion area boundary?

Additionally, the Commission shared their thoughts on expectations of enhanced safety for non-LWRs.

On the issue of containment vs confinement, the Commission stated: “The staff should develop containment performance requirements and criteria working closely with ... stakeholders ... taking into account such features as core, fuel, and cooling systems design. The staff should pursue the development of functional performance standards and then submit options and recommendations to the Commission on this important policy decision.” The staff’s work in this area at that time is documented in reference [4-19]. The most recent information on the staff’s methodology to establish functional containment performance criteria is given in reference [4-20].

Other policy issues also continue to be addressed in different SECYs. For example, policy issues relating to accident source terms and siting, emergency preparedness, physical security, and risk-informed regulation.

SECY-16-0012 [4-21], “Accident Source Terms and Siting for SMRs and Non-LWRs,” presented an assessment of the policy issues associated with the use of mechanistic source terms (MSTs) in design-basis accident (DBA) dose analyses and siting. It discussed the use of MSTs for DBAs for small modular reactors (SMRs) which, given the expected smaller amount of fuel and unique and passive nature of these designs, is expected to result in reduced source terms compared to LWRs. The staff stated its belief that a mechanistic approach could also be applied to non-LWR designs subject to availability of adequate tools and analysis approaches. This would allow future applicants to consider reduced distances to exclusion area boundaries and low population zones, and potentially increased proximity to population centers. This, in turn, may need additional direction from the Commission if an applicant were to propose

a site that is significantly closer to a very densely populated center than previously approved.

SECY-15-0077 [4-22], “Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies,” proposed a consequence-based approach to establishing requirements, as necessary, for offsite EP. It proposed revising NRC regulations and guidance through rulemaking to require SMR license applicants to demonstrate how their proposed facilities achieve Environmental Protection Agency Protection Action Guide dose limits at specified emergency planning zone (EPZ) distances, which may include the site boundary. In the SRM to the SECY, the Commission approved the staff’s recommendation to initiate rulemaking to revise regulations and guidance for EP for SMRs and other new technologies, such as non-LWRs and medical isotope facilities.

SECY-18-0076 [4-23], “Options and Recommendation for Physical Security for Advanced Reactors,” recommended limited-scope rulemaking for physical security of advanced reactors, including SMRs and non-LWRs, which was approved by the Commission. The limited scope rulemaking effort would evaluate possible performance criteria and alternative security requirements for advanced reactors that have incorporated the reactor attributes defined in the NRC’s Policy Statement on the Regulation of Advanced Reactors [4-6, 4-10], specifically designs that incorporate “enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.” The alternative physical security requirements and related guidance would support efforts to better address security concerns within the design process, and thereby reduce reliance on armed responders.

SECY-18-0060 [4-24], “Achieving Modern Risk-Informed Regulation,” discusses four significant staff initiatives one of which is an optional performance-based, technology-inclusive regulation for non-LWRs. In this SECY, the staff recommended creation of a new rule for the licensing of non-LWRs that includes an optional performance-based, technology-inclusive set of safety criteria for licensing the design and operation of advanced reactor technologies. Subsequently, Section 103 of the “Nuclear Energy Innovation and Modernization Act” (Public Law 115-439) signed on January 14, 2019, requires the NRC to complete a rulemaking, by December 31, 2027, to establish a “technology-inclusive regulatory framework” for optional use by applicants for new commercial advanced nuclear reactor licenses.

4.4 Next Generation Nuclear Plant (NGNP) Interactions with NRC

The mission of the NGNP project was to develop, license, build, and operate a prototype modular high temperature gas-cooled reactor (HTGR) plant that would generate high-temperature process heat for use in hydrogen production and other energy-intensive industries while generating electric power at the same time. The project was established for DOE by the Energy Policy Act of 2005 and was carried out



Figure 4.1 NGNP Reactor and Power Conversion Vessels

by Idaho National Laboratory (INL). The Act directed DOE to develop the NGNP prototype for commercialization and provided the licensing authority to NRC. DOE and NRC jointly developed a licensing strategy and carried out activities that provided useful input for the regulatory basis for non-LWRs discussed in Section 4.5. DOE decided in 2011 not to proceed into the detailed design and the license application phase of the NGNP project was not pursued.

Pre-licensing activities for the NGNP prototype began with the development of the NGNP Licensing Strategy Report to Congress [4-25] issued in August 2008. Subsequent NRC interactions with DOE and INL centered primarily on the NRC's review and assessment of a series of NGNP white paper submittals that described the approach that DOE and INL proposed to pursue in establishing the technical safety bases and criteria

for licensing the prototype. The NGNP white papers addressed the following aspects:

- defense-in depth approach [4-26]
- high temperature materials [4-27]
- fuel qualification [4-28]
- mechanistic source terms [4-29]
- licensing basis event selection [4-30]
- structures, systems, and components safety classification [4-31]
- determining appropriate emergency planning zone size and emergency planning attributes for an HTGR [4-32]
- probabilistic risk assessment [4-33]

In addition, NGNP carried out and submitted documentation to NRC on the following aspects:

- modular HTGR safety basis and approach [4-34]
- NGNP project regulatory gap analysis for modular HTGRs [4-35]
- key issues for NGNP licensing [4-36]

NRC staff conducted assessments of NGNP white papers and key licensing issues and provided detailed feedback, as documented in the NRC website [4-37].

NUREG-1860 [4-38] studied the feasibility of a technology-neutral risk-informed and performance-based process for licensing of future nuclear power plants. This study, along with the approach developed for licensing a prototype HTGR under the NGNP

project informed NRC’s vision and strategy and the regulatory roadmap discussed in Section 4.5 below. Also, industry’s Licensing Modernization Project’s licensing basis event selection; structures, systems, and components classification; and defense-in-depth approaches (all also discussed in Section 4.5 below) were informed by NUREG-1860 and the corresponding NGNP activities.

4.5 Planning for the Future

The NRC has developed two documents to guide its future regulation of non-LWRs. One explains the vision and strategy for the future (Section 4.5.1) and the other is a roadmap (Section 4.5.2). In addition, the NRC has been interacting with an industry-led group (The Licensing Modernization Project) to develop new regulatory guidance for the future (Section 4.5.3).

4.5.1 NRC’s Vision and Strategy for Regulating Non-Light Water Reactors

In preparing to review and regulate a new generation of non-LWRs, NRC realized that it needed to develop a flexible framework that can address different types and varieties of non-LWRs and allows different applicants to pursue different paths toward regulatory reviews and decisions. To address this need, NRC developed its vision and strategy for mission readiness in assuring safe, effective, and efficient licensing of non-LWRs [4-39]. The vision and strategy report was preceded by three non-LWR readiness assessments since the early 2000s discussed in SECY-01-0188 [4-40] and two reports to congress [4-41, 4-42].

Currently, review and licensing of non-LWR applications requires submittals under 10 CFR Part 50 or 52 that are reviewed against existing LWR criteria, developed largely based on experience with LWR technology, and would necessitate the use of regulatory exemptions and imposition of new requirements where design-specific review, analysis, and additional engineering judgment is required. The vision and strategy, when implemented, is developed to address these potential inefficiencies and provide regulatory certainty for non-LWR applicants.

Figure 4.2 shows the organization of the vision and strategy, called the “NRC Non-LWR Mission Readiness Roadmap.”

Three strategic objectives were defined to focus NRC’s effort to assure readiness to efficiently and effectively review and regulate non-LWRs; namely, enhance technical readiness; optimize regulatory readiness; and optimize communications. These strategic objectives are explained as follows:

Enhance Technical Readiness: NRC staff’s specific technical knowledge, skills, and tools are intended to be enhanced to improve the efficiency and effectiveness of review and regulatory capabilities of non-LWRs as part of this objective. Activities addressed for this objective include training; knowledge capture and knowledge management; development of analytical tools; staff capacity planning; and long-range staff

development. Identification and resolution of policy issues applicable to non-LWRs are also addressed.

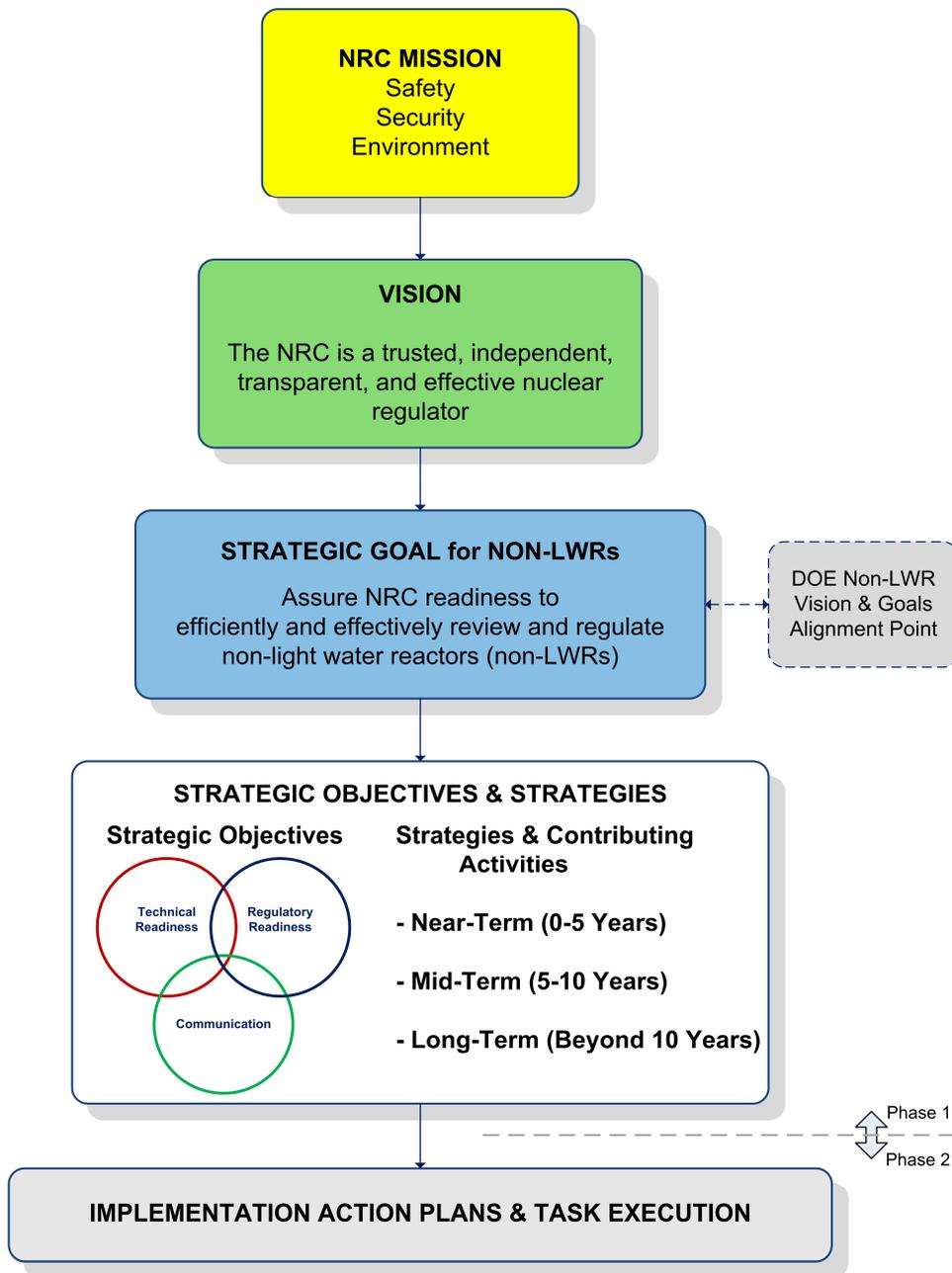


Figure 4.2 NRC Non-LWR Readiness Roadmap [4-39]

Optimize Regulatory Readiness: Regulatory readiness includes the clear identification of NRC requirements and the effective and timely communication of those requirements to potential applicants in a manner that can be understood by stakeholders with a range of regulatory maturity. Regulatory review processes are optimized when the resources

of the NRC and potential applicants are efficiently and effectively used in a way that meets NRC requirements in a manner commensurate with the risks posed by the technology, that maximizes regulatory certainty, and that considers the business needs of potential non-LWR applicants.

Optimize Communication: As part of this objective, the NRC will optimize its communication with non-LWR stakeholders by disseminating clear expectations and requirements for non-LWR regulatory reviews and oversight. Stakeholder feedback paths to the NRC will also be optimized to ensure that feedback is received, considered, and addressed in a timely manner, as appropriate.

The vision and strategy report is a planning tool that describes: 1) what work must be done to achieve non-LWR readiness, 2) how the work should be sequenced, 3) how to prepare the work force to do the work, and 4) consideration for work execution for maximum effectiveness and efficiency. It is a long-range planning document that is broken down into three periods: near-term (0-5 years), mid-term (5-10 years), and long term (greater than 10 years). The near-term actions have been further developed using Implementation Action Plans (IAPs) [4-43]. Specific, actionable tasks to achieve NRC's non-LWR strategic objectives have been identified.

4.5.2 Regulatory Review Roadmap for Non-LWRs

As part of the NRC'S vision and strategy for non-LWRs, NRC published a white paper for public discussion, on a regulatory review roadmap for non-LWRs [4-44]. This document reflects the design development lifecycle and appropriate points of interaction with the NRC and references appropriate guidance to staff reviewers and applicants.

The regulatory roadmap used DOE Order 413.3B, "Program and Project Management for the Acquisition of Capital Assets," which defines different stages of project development and critical decisions for non-LWRs, to define the associated interactions between the NRC staff and designers.

The roadmap defines flexible non-LWR review processes, including interaction during the conceptual design phase, preliminary design reviews, and standard design approvals (SDAs), to define possible stages of reviews for designs or parts of designs at various levels of completion or maturity (i.e., across a spectrum of regulatory readiness levels).

Figure 4.3 shows the alignment of various regulatory applications (e.g., construction permit, operating license, SDA, design certification, combined license) and preapplication interactions (e.g., meetings, topical reports, white papers, conceptual design reviews) with different stages of the design process. The regulatory roadmap provides discussions of different design stages, interactions, licenses, certificates, and approvals.

The roadmap also supports and discusses the development of the technology-specific or design-specific regulatory engagement plans. Parties designing non-LWRs or wishing to construct and operate a non-LWR should prepare a regulatory engagement plan as an early step in the overall program to develop and deploy a new reactor technology. The regulatory engagement plan is expected to include the following:

- technology readiness level of the reactor design, including innovative features, and the related R&D activities
- interactions with the NRC staff to reach mutual agreement on the desired outcomes of defined interactions and estimated cost and schedules for defined reviews
- near term activities needed to support the critical decision process
- longer term licensing and construction strategies for commercial units
- development of submittals and NRC review plans.

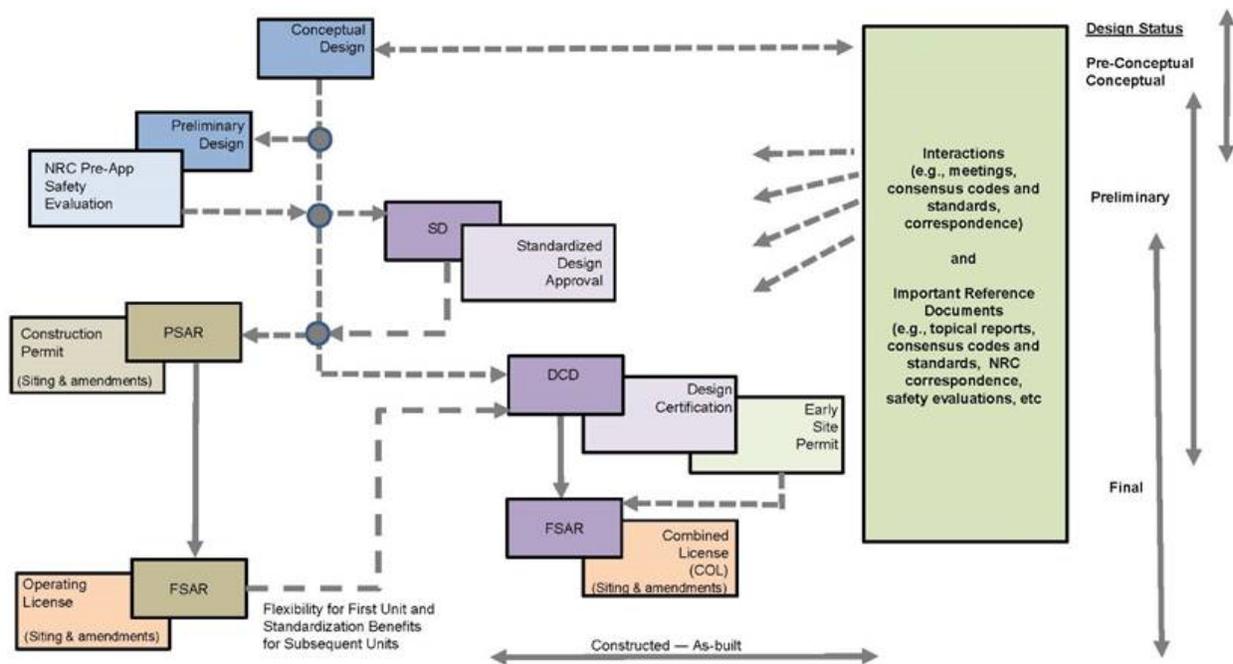


Figure 4.3 NRC Licensing-Related Processes [4-44]

As stated in the regulatory roadmap, the regulatory engagement plans for non-LWRs progressing into the preliminary design process have a number of options for applying for licenses, certifications, or approvals to support the design processes and potential commercial deployment of a non-LWR design. The designers may submit information on the preliminary design of a plant or key systems before a formal application. For NGNP design documents (Section 4.4) the DOE used this type of preliminary design review by the staff.

Designers may elect to submit an application for standard design approval as a means of progressing in the regulatory area as design decisions are made and the overall program advances. An SDA, in combination with other reference documents, can be used to support a license or certification under either 10 CFR Part 50 or 10 CFR Part 52. The use of the available combinations of preapplication interactions, creation of reference documents, and SDA is sometimes referred to as a staged licensing process. The use of a staged licensing process can reduce the degree to which applicants fail to address regulatory risks until late in the preliminary or final design processes.

Plans for the overall deployment of non-LWR designs might include multiple projects involving critical decisions related to research and test reactors, first-of-a-kind large scale plants, and subsequent commercial plants. The NRC's existing processes and practices are flexible and support interactions related to this wide variation in design development, recognizing that the NRC staff may in some cases be providing feedback and developing regulatory positions in parallel with designers assessing various alternatives during the conceptual design process. The roadmap uses an enclosure to discuss nuclear power reactor testing needs and prototype plants, as they may be needed for advanced reactor designs. The following aspects are addressed in the enclosure:

- the regulations governing the testing requirements for the licensing, approval, or certification of a proposed standard plant design for advanced reactors
- the process for determining testing needs to meet the NRC's regulatory requirements
- clarification when a prototype plant might be needed and how it might differ from the proposed standard plant design
- licensing strategies and options that include the use of a prototype plant to meet the NRC's testing requirements.

The process for determining the type of demonstration facilities that may be needed for the certification-by-test approach under 10 CFR Part 52 is discussed in SECY-91-074 [4-45] and reproduced as an appendix in the roadmap.

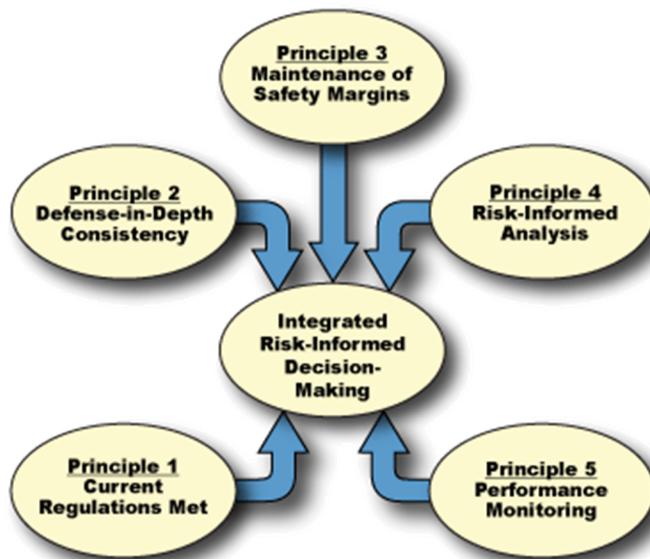
4.5.3 Licensing Modernization Project

The Licensing Modernization Project (LMP), which is led by the Southern Company, coordinated by the Nuclear Energy Institute (NEI), and cost-shared by the DOE, is developing a technology-inclusive, risk-informed, and performance-based (TI-RIPB) regulatory guidance for licensing non-LWRs for the NRC's consideration and endorsement.

The LMP developed a risk-informed, performance-based guidance document, NEI 18-04 [4-46] which describes a general methodology for identifying an appropriate scope

and depth of information to be provided in applications for licenses, certifications, and approvals. The methodology is called TI-RIPB because it includes processes that:

- utilize insights from a probabilistic risk assessment of the design in an iterative manner along with prescriptive rules to account for the uncertainties in the risk assessment, and
- defines quantifiable performance metrics for different licensing parameters (e.g., licensing basis events (LBEs), and structures, systems, and components (SSC)) to evaluate effectiveness considering the desired outcomes to be achieved.



The TI-RIPB approach is considered an alternative to the prescriptive nature of the current LWR-centric guidance that has developed over the years. This TI-RIPB methodology and the associated process does not exempt any reactor designer from existing regulations. Rather, it informs an approach to safety design that can be applied to demonstrate the compliance with the regulations applicable to the design. The NEI guidance document [4-46] describes the following TI-RIPB processes:

Figure 4.4 NRC Approach to RIPB Decision-Making

- systematic definition, categorization, and evaluation of event sequences for selection of licensing basis events, which include anticipated operational occurrences, design basis events and accidents, and beyond design basis events
- systematic safety classification of SSCs, development of performance requirements, and application of special treatments
- guidelines for evaluation of defense-in-depth adequacy.

These processes are intended to be used to:

- develop logical, coherent, and complete bases for the development of the safety design and evaluation of the safety design based on the specific technology and design.

- apply a sound PRA, including appropriate probabilistic models based on available standards, to develop and evaluate the safety design outcomes for a design.
- answer the following broad questions:
 - What are the plant Initiating events and event sequences that are associated with the design?
 - How does the proposed design and its SSCs respond to initiating events and event sequences?
 - What are the margins provided by the facility’s response, as it relates to prevention and mitigation of radiological releases within prescribed limits in the protection of public health and safety?
 - Is the philosophy of defense-in-depth adequately reflected in the design and operation of the facility?

The NRC staff has released its working draft regulatory guidance DG-1353, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” [4-47]. The RG plans to endorse, with clarifications as detailed in the RG, the principles and methodology in the industry TI-RIPB guidance as one acceptable method for determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs.

As part of developing the guidance document, the LMP produced four white papers on the following topics, which were reviewed by the NRC.

- selection of licensing basis events [4-48]
- probabilistic risk assessment [4-49]
- safety classification and performance criteria for structures, systems, and components [4-50]
- risk-informed performance-based evaluation of defense-in-depth adequacy [4-51].

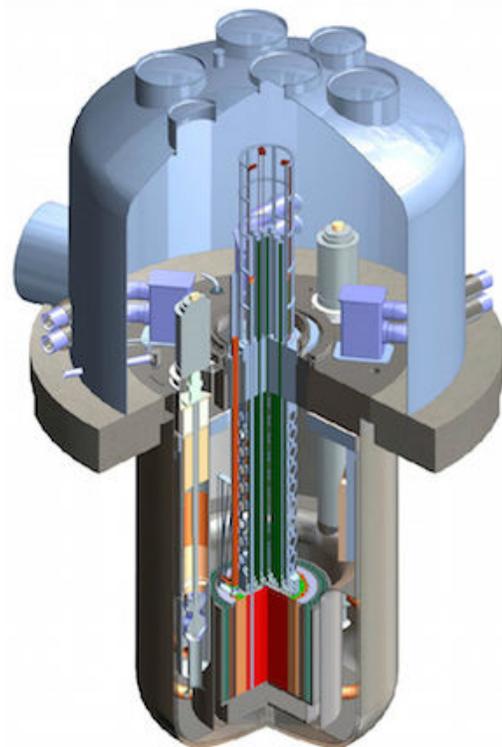


Figure 4.5 Proposed Traveling Wave Reactor

These documents, discussed below, involved a refinement of the NGNP methodologies (cf Section 4.4) to reflect interactions with the NRC, feedback from industry, broadening relative to technology-neutral applications to ensure applicability to various non-LWR technologies and were used to develop NEI 18-04.

Licensing Basis Event (LBE) Selection Process

The LBE selection in the LMP is an iterative process involving design development, PRA, selection of LBEs, and evaluation of LBEs. It begins in the conceptual design phase when many design details are not available, the PRA effort has not begun, and the safety design is just being formulated. The initial set of LBEs is proposed based on engineering judgment.

The scope and the level of the PRA is expanded as different parts of the PRA mature at different times to provide input to the LBE selection process. The selected set of LBEs undergo deterministic and probabilistic evaluations which include evaluation of LBEs against a Frequency-Consequence (F-C) target, developed as part of the LMP program.

In addition, the integrated risk of the entire plant, including all the LBEs, is evaluated against the quantitative health objectives and 10 CFR Part 20 requirements. The LBE process allows the determination of risk significant LBEs and SSCs and the evaluation of defense-in-depth adequacy. In some applications, if the design basis accidents and SSC classification steps are completed prior to the application of the LMP methodology, the evaluation of LBEs, as discussed in [4-48], can be viewed as a means of confirming or refining prior selections in formulating the design and licensing bases.

Probabilistic Risk Assessment and its Role in the LMP Licensing Basis Document

PRA plays a significant role in the LMP licensing basis development. In fact, the major difference in the LMP from the current approach to LWR licensing is the early development of PRA and the consistent, integrated use of PRA in different facets of the licensing basis. A full scope, Level 3 PRA will be used for the following aspects:

- evaluation of design alternatives using risk insights based on the PRA
- probabilistic input in selecting LBEs, including evaluation of LBEs against F-C targets
- PRA-driven process for selection of safety-related SSCs
- PRA-derived process for defining special treatment and design requirements regarding performance, capability, and reliability of SSCs used in the prevention and mitigation of event sequences and accidents
- basis for risk-informed determination and evaluation of DID.

The technical adequacy of the non-LWR PRA will be demonstrated through the use of ASME/ANS PRA Standard for Advanced non-LWR PRA RA-S-1.4 [4-52]. A trial use PRA Standard for Advanced non-LWR Nuclear Power Plants is currently available and a final version is expected to be issued. The Advanced non-LWR PRA Standard is similar to the ASME/ANS PRA Standards that are being developed for at-power PRAs, low power and shutdown PRAs, and Level 2 and Level 3 PRAs.

Since the PRA for non-LWR Advanced Reactors is planned to be used starting at the pre-conceptual or early stages of the conceptual design, the initial PRA will be simplified

compared to the full scope PRA enough to meet applicable PRA standards that will evolve by the final application stage. However, the PRA in the early stages incorporates traditional techniques for system analysis (e.g., failure modes and effects analysis, process hazard analysis, single failure analysis, etc.) which were used to define LBEs for currently licensed LWRs and ensures that early stage evaluations are systematic, reproducible, and as complete as the current stage of design permits.

As stated in NEI 18-04, PRA development is a continuum and provides a more frequent, integrated plant performance check that is otherwise missing in the conventional design process and can also provide risk insights to help design decisions. This application is, however, associated with the following technical issues and challenges, as noted in the LMP white paper on PRA:

- PRA treatment of multi-reactor module plants
- sufficiency of relevant PRA data
- treatment of inherent and passive safety features
- new risk-informed applications for non-LWR PRAs

Currently, the technical adequacy of the risk-informed decisions is confirmed by NRC review of license amendment requests which are subject to full NRC review resulting in an amended license. For non-LWRs, any risk-informed licensing decisions that are supported by the PRA would be subject to full licensing review by including the justification for the decision as part of the license application or supporting topical report. For the LMP approach for non-LWRs, many risk-informed decisions are expected at different stages and a process for NRC review and regulatory guidance will apply.

Safety Classification and Performance Criteria for SSCs

The LMP defines an approach to SSC safety classification and defines requirements for SSC performance of safety functions in the prevention and mitigation of LBEs that are modeled in the PRA, based on the classification. This approach uses many features of the existing risk-informed SSC classification approaches developed for current and advanced LWRs as part of 10 CFR 50.69, Risk-Informed Categorization and Treatment for Structures, Systems, and Components in Nuclear Power Reactors [4-53].

The proposed classification system is defined as follows for SSCs that are “safety-significant:”

Safety-Related (SR):

- SSCs that are available to perform the required safety functions to mitigate the consequences of design basis events to within the LBE F-C target, and to mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 [4-54] using conservative assumptions

- SSCs relied on to perform required safety functions to prevent the frequency of BDBE with consequences greater than the 10 CFR 50.34 dose limits from increasing into the design basis event region and beyond the F-C target

Non-Safety-Related with Special Treatment (NSRST):

- non-safety-related SSCs relied on to perform risk-significant functions. Risk-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the F-C target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.
- non-safety-related SSCs relied on to perform functions requiring special treatment for DID adequacy

The proposed classification system for SSCs that are “not safety-significant” (NST) are “non-safety-related with no special treatment” and include all other SSCs.

As can be noted, safety-significant SSCs include all those classified as SR or NSRST. None of the NST SSCs are classified as safety-significant.

The process of SSC classification starts as an SSC function classification because only SSC functions that prevent or mitigate events represented in the LBEs are taken into consideration. Safety-significant SSCs include those that perform risk-significant functions and those that perform functions that are necessary to meet DID criteria. Thus, the SSC classification process defines both risk-significant and safety-significant SSCs. As discussed above, the scope of the PRA is expected to cover all SSCs that are responsible for preventing or mitigating the release of radioactive materials, and accordingly, LBEs include all the relevant SSCs for prevention and mitigation functions.

NEI 18-04 discusses the applicability of special treatment requirements for SSCs and provides a table (Table 4.1 of NEI 18-04) summarizing the category of special treatments that are applicable to each group of SSCs (SR, NSRST, and NST). SSC classification along with DID adequacy evaluations were considered to provide the general guidance. In an actual analysis, a case-by-case evaluation will be necessary.

Evaluation of Defense-in-Depth (DID) Adequacy

In the LMP approach, DID is considered and implemented in defining design requirements, developing the design, evaluating the design from both deterministic and probabilistic perspectives, and defining the programs to ensure adequate public protection. An integrated decision panel is formed to guide the overall design effort (including development of plant capability and programmatic DID features), conduct the DID adequacy evaluation of the resulting design, and document the DID baseline. The role of the panel is similar to that in currently operating plants to guide risk-informed changes to the licensing basis, such as risk-informed safety classification under 10 CFR

50.69. The panel includes a cross-functional team responsible for the design, operations, and maintenance program development and for performing the necessary deterministic and probabilistic evaluations as identified in the integrated process for incorporation and evaluation of defense-in-depth defined in NEI 18-04.

The three elements of the DID adequacy evaluation process, discussed in detail in NEI 18-04, are defined as follows:

Plant Capability Defense-In-Depth – This element is used by the designer to select functions, SSCs, and their bounding design capabilities to assure safety adequacy.

Programmatic Defense-In-Depth - This element is used to address uncertainties when evaluating plant capability DID as well as when protective strategies are being defined. It provides means to incorporate special treatment (i.e., those requirements that provide increased assurance beyond normal practices that SSCs perform their design-basis function) while designing, manufacturing, constructing, operating, maintaining, and inspecting the plant and the associated processes to ensure that there is reasonable assurance that the predicted performance can be achieved throughout the lifetime of the plant.

Risk-informed Evaluation of Defense-In-Depth - This evaluation is performed by a risk-informed integrated decision-making process to assess sufficiency of DID and to enable consideration of different alternatives for achieving commensurate safety levels at reduced burdens combining the plant and programmatic elements. The outcome of this element also establishes a DID baseline for managing risk throughout the plant lifecycle.

For advanced non-LWRs, the DID framework defined in the LMP is a comprehensive, integrated process consisting of 18 different tasks which address many activities. Detailed descriptions are provided in NEI 18-04 and in the white paper for DID evaluation.

4.5.4 International Activities

There are currently two important international organizations considering the safety and licensing of non-LWRs that may impact the future of NRC licensing: the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development and the International Atomic Energy Agency (IAEA). NRC participates in activities in both organizations, being able to contribute and to benefit from these efforts.

The NEA has a “Working Group on the Safety of Advanced Reactors” (WGSAR) [4-55] that has as a part of its mandate: “The WGSAR will provide regulatory perspectives through the issuance of technical reports containing discussions of areas in which additional or revised regulatory framework and licensing approaches, including safety research, may be needed to facilitate effective regulation of advanced reactors and to develop common understanding and approaches.” Toward that end the group has

already finished or has in draft form several reports on sodium cooled fast reactors; namely, on severe accident prevention and mitigation, analytical codes and methods, fuel qualification, and neutronics and criticality safety. More generically, in 2018 they initiated efforts to prepare a report on regulatory approaches on fuel qualification for all advanced reactor types.

According to SECY-18-0011 [4-56]: IAEA, in collaboration with the International Project on Innovative Nuclear Reactors and Fuel Cycles and the Generation IV International Forum, established the Sodium-Cooled Fast Reactor Task Force. This task force collaborates with international designers, governmental organizations, and regulators to develop safety design criteria and safety design guidelines for SFRs [4-57]. IAEA also has a coordinated research activity on modular HTGR safety design criteria.

There is an IAEA SMR Regulators Forum, wherein interested regulators identify and address key regulatory challenges that may emerge in future SMR regulatory discussions. This forum focuses on issues that are applicable to both light water cooled and non-LWR reactors, such as emergency planning and defense-in-depth. In addition to specific technical areas, IAEA and organizations such as the World Nuclear Association publish useful summaries of non-LWR designs and the status of programs in various countries.

4.6 References

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- 4-8. "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," U.S. Nuclear Regulatory Commission, August 1985. (ML003711521)
- 4-9. "Use of Probabilistic Risk Assessment methods in Nuclear Regulatory Activities, PRA Policy Statement," 60 FR 42622, US Nuclear Regulatory Commission, August 1995.
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- 4-44. "A Regulatory Review Roadmap for Non-Light Water Reactors," U.S. Nuclear Regulatory Commission, December 2017. (ML17312B567)

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- 4-46. "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," Draft Report Revision N, NEI 18-04, Nuclear Energy Institute, September 28, 2018. (ML18271A172)
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APPENDIX – KEY HISTORICAL REGULATORY DOCUMENTS

Table A.1 NRC Documents

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Regulation of Advanced Nuclear Power Plants, Statement of Policy	July 8, 1986 51 FR 24643 ML082750370	This policy statement reinforces the Commission's current policy regarding advanced reactors and includes new items to be considered during the design of these reactors, including security, emergency preparedness, threat of theft, and international safeguards. (See also revision below.)
Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants	Peter M. Williams & Thomas L. King, June 1988 NUREG-1226 ML13253A431	Overview of the significant changes from the proposed Policy Statement to the final Statement, and of the Commission's response to questions contained in the proposed Policy Statement. Discusses the definition for advanced reactors, the establishment of an Advanced Reactors Group, the staff review approach and information needs, and the utilization of the Policy Statement in relation to other NRC programs
Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements	April 8, 1993 SECY-93-092 ML040210725	Discusses 10 issues where there may need to be departures from regulations developed for LWRs and how the staff is addressing those issues. The corresponding SRM (July 1993, ML003760774) gives approval, or suggestions for further study, on each issue.
Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Part 1, Format and Content; Part 2, Standard Review Plan and Acceptance Criteria	1996 NUREG-1537	Guidance provided in 18 chapters on all subjects needed for non-power reactors.
Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactor Designs,	July 22, 2002 SECY-02-0139 ML021790610	This SECY is an early discussion of policy issues and is based on what was learned from experience with the Pebble Bed Modular Reactor. Nevertheless, the issues discussed are still relevant.

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Policy Issues Relating to Licensing Non-Light Water Reactor Designs	March 28, 2003. SECY-03-0047 ML030160002	Contains recommendations for Commission consideration on seven technical policy issues identified in the pre-application reviews to date on non-LWR designs. The seven issues involve the approach to licensing on key aspects of reactor design and operation which relate to Commission policy and practice and which could impact the viability of future non-LWR designs.
Status of Response to the June 26, 2003, Staff Requirements Memorandum on Policy Issues Related to Licensing Non-Light Water Reactor Designs	June 23, 2004 SECY-04-0103 ML041140521	Staff's response to the Commission direction on two policy issues in the subject SRM. These two issues include (1) the integrated risk posed by multiple reactors, and (2) containment versus confinement.
Feasibility Study for a Risk-Informed and Performance-Based Structure for Future Plant Licensing	December 2007 NUREG-1860 ML073400763 Appendices: ML080440215	Establishes the feasibility of developing a RIPB regulatory structure for future licensing. Documents a "framework" that could be used to develop a set of requirements that could be an alternative to 10 CFR 50. Does not represent a staff position but rather is a research project.
Updated Policy Statement on Regulation of Advanced Reactors and "Policy Statement on the Regulation of Advanced Reactors,"	Sept. 11, 2008 SECY-08-0130 and October 14, 2008 73 FR 60612 ML082750370	Reinforces the Commission's current policy regarding design considerations for advanced reactors, early interactions with advanced reactor designers, and timely comment on potential technical and policy issues. In addition, the revised policy statement includes new items to be considered during the design of these reactors, including security, emergency preparedness, threat of theft, and international safeguards
Potential Policy, Licensing, and Key Technical Issues for Small Modular Reactor Designs	March 28, 2010 SECY-10-0034 ML093290268	Issues included are implementation of defense-in-depth; source term, dose calculations and siting; and operator needs for small or multi-unit plants. NGNP project also enters into discussion.
Report to Congress on Advanced Reactor Licensing	August 2012 ML12153A014	The status as of 2012 of proposed licensing activities with respect to both advanced LWRs and non-LWRs.

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Final Interim Staff Guidance Augmenting NUREG-1537, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Part 1, Format and Content; Part 2, Standard Review Plan and Acceptance Criteria, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors	October 17, 2012 ISG NUREG-1537 Part 1: ML12156A069 Part 2: ML12156A075	Updates and expands the content of NUREG-1537 to provide guidance in preparing a license application and for the NRC staff in evaluating the application and issuing a license for an aqueous homogeneous reactor or subcritical system for isotope production or for similar facilities
NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors	G.F. Flanagan, G.T. Mays, & I.K. Madni April 2014 NUREG/KM-0007 ML14128A346	Explains NRC's knowledge management program and captures information on the technical and licensing aspects of sodium-cooled fast reactors. It is based on the NRC's experience with FERMI-1, PRISM, and CRBR.
Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications	January 8, 2015. SECY-15-0002 ML13281A382	This request is to assure consistency with respect to any way an application is received (e.g., 10 CFR Part 50 or Part 52) and is meant to apply to any type of reactor.
Options for Emergency Preparedness for Small Modular Reactors and other New Technologies	March 29, 2015 SECY-15-0077 ML15037A176	Requests approval to move forward with rule on consequence-based emergency preparedness taking into account Environmental Protection Agency Protection Action Guides
NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness	2016 ML16139A812	Written to assure readiness to effectively and efficiently review and regulate non-LWRs. Discusses activities needed in near-, mid-, and long-term.
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	2016 ML16334A495	Vol 1 explains the six strategies needed to assure readiness for non-LWRs. Vol 2 provides details of the activities needed for each strategy.

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Accident Source Terms and Siting for SMRs and Non-LWRs	February 7, 2016 SECY-16-0012 ML15309A319	Status report on staff activities and assessment of potential policy issues associated with the use of mechanistic source terms in dose analyses.
Rulemaking Plan on Emergency Preparedness for Small Modular Reactors and Other New Technologies	May 31, 2016 SECY-16-0069 ML16020A388	Based on the Commission's approval of SECY-15-0077, this is an update on the plan for rule making.
A Regulatory Review Roadmap for Non-Light Water Reactors	December 2017 ML17312B567	Provide options available for NRC review of pre-application information and of formal applications. Enclosure describes testing needs and prototype plants.
Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors	2018 Draft Regulatory Guide 1353 ML18264A093	Proposed guidance on using a TI-RIPB methodology to inform the content of applications for licenses, certifications, and approvals for non-LWRs. The selection of licensing basis events; classification and special treatments of structures, systems, and components; and assessment of defense in depth are of fundamental interest.
Advanced Reactor Program Status	January 25, 2018 SECY-18-0011 ML17334B199 (two enclosures) ML17334B184 ML17334A907	Provides an update on all activities related to the six strategies in the IAP through Dec. 2017. Details are given in enclosure 1 while enclosure 2, "Non-LWR Landscape," explains related activity, for example, what is relevant at DOE, in industry, and internationally.
Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," U.S. Nuclear Regulatory Commission,	April 2018 RG 1.232, Rev. 0 ML17325A611	How the general design criteria (GDC) may be adapted for non-LWR designs. This guidance may be used by non-LWR reactor designers, applicants, and licensees to develop principal design criteria (PDC) as required by the applicable regulations. It also describes the NRC's proposed guidance for modifying and supplementing the GDC to develop PDC for sodium-cooled fast reactors and modular high temperature gas-cooled reactors.

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Achieving Modern Risk-Informed Regulations	May 23, 2018 SECY-18-0060 ML18110A187 (+8 enclosures)	Requests consideration of expanding the use of risk and safety insights, revising 10CFR50.59 to allow licensees greater flexibility, developing a performance-based (PB), technology-inclusive regulation for non-LWRs and a regulation to define high-level PB-based I&C design.
Options and Recommendation for Physical Security for Advanced Reactors	August 1, 2018 SECY-18-0076 ML18052B032 (+1 enclosure)	Relevant to both light water SMRs and non-LWRs, it includes a recommendation for limited-scope rule making.
Functional Containment Performance Criteria for Non-Light-Water-Reactors	September 28, 2018 SECY-18-0096 ML18115A157	Staff's proposed methodology for establishing functional containment performance criteria for non-LWRs.

Table A.2 Non-NRC Documents

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors	Jim Kinsey and Mark Holbrook Idaho Nat'l. Lab. December 2014 INL/EXT-14-31179, Rev. 1 ML14353A246	A set of draft Advanced Reactor Design Criteria is proposed for consideration by NRC in the establishment of guidance for use by non-LWR designers and NRC staff. They were developed to preserve the underlying safety bases expressed by the original GDC and recognizing that advanced reactors may take advantage of various new passive and inherent safety features different from those associated with LWRs.
Selection of Licensing Basis Events, Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors (Draft Report)	Southern Company April 2017 ML17145A574	Presents a technology-inclusive, risk-informed, and performance-based approach to identifying licensing basis events (LBEs). In this paper, the LMP is seeking: 1) NRC's approval of the proposed approach for incorporation into appropriate regulatory guidance; and 2) identification of any issues that have the potential to significantly impact the selection and evaluation of LBEs.
Probabilistic Risk Assessment Approach, Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors (Draft Report)	Southern Company June 2017 ML17158B543	Outlines the approach to develop a PRA for advanced non-LWR plants in support of risk-informed and performance-based applications including: evaluation of design alternatives and incorporation of risk insights into early and continuing development of the design; input to the selection of LBEs; input to the safety classification of SSCs.

Title	Author (if present) / Date / ID / ADAMS Accession No.	Description
Safety Classification and Performance Criteria for Structures, Systems, and Components, Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors (Draft Report)	Southern Company October 2017 ML17290A463	Outlines the following tasks that impact SSC performance requirements: complete the process of SSC safety classification by defining additional categories beyond safety-related; describe the LMP approach to the definition of risk significant SSCs; describe the LMP approach for defining safety significant SSCs in terms of their risk significance and role in supporting DID; provide guidance for the development of special treatment requirements, functional design criteria, and performance requirements for the reliability and capability of SSCs in the prevention and mitigation of LBEs.
Risk-Informed Performance-Based Evaluation of Defense-in-Depth Adequacy, Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors (Draft Report)	Southern Company December 2017 ML17354B174	Proposed DID framework is technology-inclusive, risk-informed, and performance-based (TI-RIPB). The approach to establishing DID adequacy incorporates DID attributes into the plant capabilities and programmatic elements of DID. The integrated evaluation of DID adequacy includes both quantitative elements to incorporate RIPB considerations and qualitative elements that address uncertainties and limitations in the quantitative models and supporting data.
Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development, Draft Report Revision N	Nuclear Energy Institute Sept. 28, 2018 NEI 18-04 ML18271A172	Presents a, technology-inclusive, risk-informed, and performance-based process for selection of licensing basis events; safety classification of structures, systems, and components and associated risk-informed special treatments; and determination of defense-in- depth adequacy for non-LWRs.