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NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors

Office of Nuclear Regulatory Research

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ABSTRACT

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research has been actively engaged in an effort to develop and compile information on liquid-metal-cooled reactors (LMRs), particularly sodium-cooled fast reactors (SFRs), as part of a concerted knowledge management (KM) program for LMRs. The objective of this program is to apply KM principles to capture and retain technical knowledge related to LMRs that NRC staff might need to support evaluations as part of licensing activities as related to future LMR applications for design certification and combined operating licenses. In support of this objective, the NRC has focused its efforts on documenting previous NRC licensing activities associated with Fermi 1, Power Reactor Inherently Safe Module, and the Clinch River Breeder Reactor, as well as on international research and development, safety analyses, and licensing activities associated with LMRs. This includes information and documentation on LMR severe accidents, operational issues, and analysis tools and codes that would be relevant for licensing purposes. In addition to capturing historical information and discussing recent and current activities, a second objective was to develop informational tools to facilitate the compilation and access to this information such as an LMR Desk Reference and an SFR Technology Course that are described in this report. Much of the information compiled and collected for LMRs has been added to the NRC's Knowledge Center, which is one of the agency's key information technology applications for capturing and sharing knowledge.

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The authors also want to thank Ms. Linda Dockery of the Oak Ridge National Laboratory, now retired, for her outstanding efforts in preparing and successfully formatting this document as we included material from several informal project reports and other sources.

The following individuals provided project leadership and management oversight in the development of this report:

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ACRONYMS

4S	Super Safe, Small and Simple
ACRS	Advisory Committee on Reactor Safeguards
ACS	auxiliary cooling system
AEC	U.S. Atomic Energy Commission
AI	Atomics International
ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
ANL–W	Argonne National Laboratory–West
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARC	advanced reactor concept
ASME	American Society of Mechanical Engineers
aSMR	advanced small modular reactor
ASTM	American Society for Testing and Materials
BDBA	beyond-design-basis accident
C	Celsius
CFR	*Code of Federal Regulations*
COL	combined operating license
CoP	communities of practice
CP	construction permit
CRBR	Clinch River Breeder Reactor
CRBRP	Clinch River Breeder Reactor Project
D&D	decontamination and decommissioning
DBA	design-basis accident
DC	design certification
DFR	Dounreay Fast Reactor
DOE–NE	U.S. Department of Energy–Nuclear Energy
DR	desk reference
EBR-I	Experimental Breeder Reactor-I
EBR-II	Experimental Breeder Reactor-II
ERDA	U.S. Energy Research and Development Administration
F	Fahrenheit
FFTF	Fast Flux Test Facility
FSAR	final safety analysis report
GE	General Electric
GIF	Generation IV International Forum
HLW	high-level waste
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency

IEEE	Institute of Electrical and Electronics Engineers
IFR	Integral Fast Reactor
IHX	intermediate heat exchanger
IRACS	intermediate heat exchanger auxiliary cooling system
JCN	Job Control Number
KC	Knowledge Center
KM	knowledge management
KMSC	KM Steering Committee
LMFBR	liquid-metal fast breeder reactor
LMR	liquid-metal-cooled reactor
LWR	light-water reactor
MOX	mixed oxide fuel
MWe	megawatts electric
MW-s	megawatt seconds
MWt	megawatts thermal
NDE	non-destructive examination
NRC	U.S. Nuclear Regulatory Commission
OECD/NEA	Organization of Economic Co-operation and Development/Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
OSTI	Office of Scientific and Technical Information
PFR	Prototype Fast Reactor
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
PSAR	preliminary safety analysis report
PSER	preapplication safety evaluation report
PWR	pressurized-water reactor
R&D	research and development
RD&D	research, development, and deployment
RDT	reactor development and technology
RES	Office of Nuclear Regulatory Research
RI	Rocketdyne International
RVACS	Reactor Vessel Auxiliary Cooling System
SAFR	Sodium Advanced Fast Reactor
SEFOR	Southwest Experimental Fast Oxide Reactor
SER	safety evaluation report
SFR	sodium-cooled fast reactor
SMiRT	Structural Mechanics in Reactor Technology
SMR	small modular reactor
S-PRISM	Super-PRISM
SS	stainless steel

SSC series	Super System Code series
SSCs	systems, structures, and components
TMI	Three Mile Island
TREAT	Transient Reactor Test Facility
TVA	Tennessee Valley Authority
TWG-FR	Technical Working Group on Fast Reactors

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) has been actively engaged in an effort to develop and compile information on liquid-metal-cooled reactors (LMRs), particularly sodium-cooled fast reactors (SFRs), as part of a concerted knowledge management (KM) program for LMRs. The objective of this program is to apply KM principles to capture and retain technical knowledge related to LMRs that NRC staff might need to support evaluations as part of licensing activities as related to future LMR applications for design certification (DC) and combined operating licenses. In support of this objective, efforts have focused on documentation of previous NRC licensing activities associated with Fermi 1, Power Reactor Innovative Small Module (PRISM), and the Clinch River Breeder Reactor (CRBR), as well as international research and development, safety analyses, and licensing activities associated with LMRs. This includes information and documentation on LMR severe accidents, operational issues, and analysis tools and codes that would be relevant for licensing purposes. In addition to capturing historical information and discussing recent/and or current activities, a second objective was to develop informational tools to facilitate the compilation and access to this information such as an LMR Desk Reference and an SFR technology course, which will be discussed in more detail later in this report. While a number of review articles and documents on international LMR operating experience were included in the collection of documents reviewed for this program and some references are made to international LMR experience in this report, the primary focus of this document is on U.S. LMR programs and activities.

This report is not intended as a guide for licensing LMRs rather it is a reference tool from a knowledge management perspective that provides relevant information for NRC reviewers on LMR technology specifics, differences between LMRs and light-water reactors (LWRs), historical results from previous safety-related research and development (R&D) programs, and past licensing experience of LMRs.

NRC RES contracted with the Oak Ridge National Laboratory (ORNL) under NRC Job Control Number (JCN) N6472, "Knowledge Management Support for Liquid Metal Reactors," to support the various LMR KM efforts and to document them in this NUREG report. An important element of the work performed under JCN N6472 involved developing an LMR-specific section of NRC's Knowledge Center (KC) as the principal tool for storing and organizing the information collected under this project. Much of the information compiled and collected for LMRs has been added to the KC, which is one of the NRC's key information technology applications for capturing and sharing knowledge.

1.1 Background and Basis for Establishing an LMR KM Program

The United States had an active research and development (R&D) program focused on commercial demonstration and development of an LMR from 1950–1989. The Atomic Energy Commission and its successor, the U.S. Department of Energy (DOE), principally funded this program. It involved collaboration among the national laboratories, reactor vendors, utilities, and the NRC. The program resulted in (1) the design and operation of three test reactors (the Experimental Breeder Reactor-I (EBR-I), Experimental Breeder Reactor-II (EBR-II), and the Fast Flux Test Facility (FFTF)), and (2) the design and issuance of a construction permit for a commercial demonstration plant, the Clinch River Breeder Reactor Project (CRBRP). The U.S. Government in 1983 canceled construction of this plant. The commercial joint government and industry development program ended in 1989 with the preliminary design of two Advanced Liquid-Metal Reactors (ALMRs), the Sodium Advanced Fast Reactor (SAFR) and the Power Reactor Innovative Small Module (PRISM), which were given preliminary safety evaluation reviews by the NRC. Because of the declining interest in construction of new reactors in the United States following the Three Mile Island (TMI) accident, the DOE continued the LMR program as a research and technology development effort without focusing on a commercial development program. As a result, with no prospects for an application, LMR activities at the NRC were reduced considerably. In the intervening 23 years following the termination of the LMR development program, many of the NRC staff who had knowledge of LMR technology have retired, left the agency, or are working in other areas of the NRC. The situation is exacerbated by the lack of young scientists and engineers becoming familiar with this type of reactor design and related technology and the fact that LMR technology is no longer a part of the curriculum in most U.S. nuclear engineering departments.

However, recent focus on small modular reactors (SMRs) in the United States and the international Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) program interest in Generation IV reactors indicate a possible renewed interest in LMRs for high-temperature use and reduction of actinide inventories in used nuclear fuel. It was determined that a KM effort was needed for LMR technology within the NRC to provide the staff with basic information on this technology in preparation for future licensing activities.

1.2 Objective of This Report

The objective of this report is two-fold: (1) to document the LMR KM activities conducted under N6472 and (2) to integrate the products from this project into one useful resource that will provide NRC staff with a knowledge base and a reference that is relevant to support potential future LMR design certification and licensing activities undertaken by new staff or staff who may not be as familiar with LMRs as they are with light-water reactors (LWRs). In some instances, information is directly included in this document, while in other instances, this document refers the reader to other resources and tools accessible to NRC staff.

1.3 Summary of LMR KM Project Activities Related to JCN N6472

The first two phases of the project focused on identification and capturing relevant information on research and development for the design and safety of LMRs, and relevant information from a licensing and regulatory perspective, including key information on PRISM and CRBR licensing experience. Operating experience information was compiled for those LMRs that have operated in other countries. Key accomplishments included:

- Developing an LMR taxonomy for categorizing and organizing LMR technical information that was entered into the NRC's KC and is described in Section 1.5.
- Identifying, categorizing, and uploading about 125 full documents and technical reports to the NRC's KC.
- Preparing an LMR "Desk Reference Guide" on LMR design information, safety issues, operating experience.
- Obtaining information from LMR subject-matter experts and organizations with LMR experience and expertise.
- Identifying three LMR experts and coordinating the development of three white papers and three corresponding presentations by these experts as part of the NRC's RES

seminar series. These presentations were video recorded as part of RES archived information

The later phases of the project focused on the development of an SFR technology course composed of nine modules, complete with module objectives, discussion questions, and annotated slides.

More information on each of these LMR KM activities is provided in Chapters 3 through 6. Appendix A presents a table of the titles and primary taxonomy categories for the 125 documents that were added to the LMR section of NRC's KC. Appendix B provides in tabular form a list of LMR reactors that either operated or were designed worldwide, sorted alphabetically, including design summary information for each. The three aforementioned LMR white papers are included in their entirety in Appendix C, while Appendix D contains the 3-day agenda developed for the SFR Technology Course. Appendix E contains an example from a report on SFR Codes. Appendix F is a paper by Imtiaz Madni providing input on potential research on LMRs for NRC's advanced reactor research plan. This paper provided input to both developing the course as described in Section 6 and evaluating which SFR codes to evaluate as discussed in Section 7.

While not a part of N6472, another complementary project related to N6472 involved compiling information for a select number of SFR accident analysis codes and models. This project (JCN N6975) was entitled "Sodium-Cooled Fast Reactor (SFR) Codes." The objective of this project was to characterize and summarize the features and status of SFR computer codes for which information could be obtained. The basis for this project came from a white paper that RES was developing, which summarized a cross section of existing models/codes and development needs for SFR transient and accident analyses, to ultimately determine whether an independent analysis capability is needed to support the review of future SFR design-certification applications. Several SFR codes were listed in that white paper along with their attributes. It was also indicated in the white paper that some legacy SFR codes (e.g., SSC, SIMMER, and SAS4A/SASSYS1) developed under NRC/DOE sponsorship a number of years ago may be evaluated to determine whether they can be used to provide that analytical capability.

This project on SFR Codes expanded on RES's white paper by providing characterization of attributes for several SFR codes, including the computing platform on which the code was developed to run, verification and validation activities, current operating status, availability, operating platforms, phenomena modeled, etc. This project did not attempt to review all the SFR codes, but only a sufficient number to identify the accident sequences as we now know them, to provide NRC staff with the necessary background that would be required for conducting licensing review, including information on what types of codes may be needed and examples of such codes. This includes the capability of analyzing anticipated operational occurrences, design-basis accidents (DBA), and beyond-design-basis accidents (BDBA) including severe accidents. The results were documented in an informal report to the NRC project manager.

1.4 Overview of the NRC's Knowledge Management Program

It is important to include background information on the purpose and development of NRC's KM program to understand how some of the information developed under this LMR KM effort is integrated into the overall NRC KM program. NRC recognized the importance of KM as a discipline and as a tool for capturing and transferring knowledge as part of its human capital management process. KM programs and activities support agency objectives to maintain core competencies and meet the future needs of headquarters and regional offices. As the NRC has added new staff in recent years, KM programs have supported the transfer of knowledge from staff members who have many years of licensing and regulatory experience to new staff, not only to assist in the licensing of new plants but also to continue the oversight of the safe operation of existing plants.

Discussion in SECY-06-164 (Ref. 1) succinctly ties the NRC's overall approach for implementing its KM program to a set of specific business objectives:

> The NRC is a knowledge-centric agency that relies on its staff to make the sound regulatory decisions needed to accomplish the agency's mission. In the recent past, the agency has enjoyed a stable workforce and a climate of slowly-evolving technologies that has allowed it to meet its performance goals by using an informal approach to KM. That environment has changed and the agency must now institute a systematic approach to KM that can support the faster rate of knowledge collection, transfer, and use needed to accommodate increased staff retirements, mid-career staff turnovers, the addition of new staff, and the broader scope of knowledge needed to expand the agency's knowledge base to support new technologies and new reactor designs.

The NRC has identified four principal categories of initiatives within its KM program aimed at retaining NRC's knowledge base. These four categories include:

1. human resource processes, policies, and procedures
2. knowledge-sharing practices
3. knowledge-recovery practices
4. information technology applications to acquire, store, and share knowledge

Figure 1.1 (Ref. 1) presents the overall structure and framework for its KM program and illustrates how the program supports successfully accomplishing the above-noted initiatives.

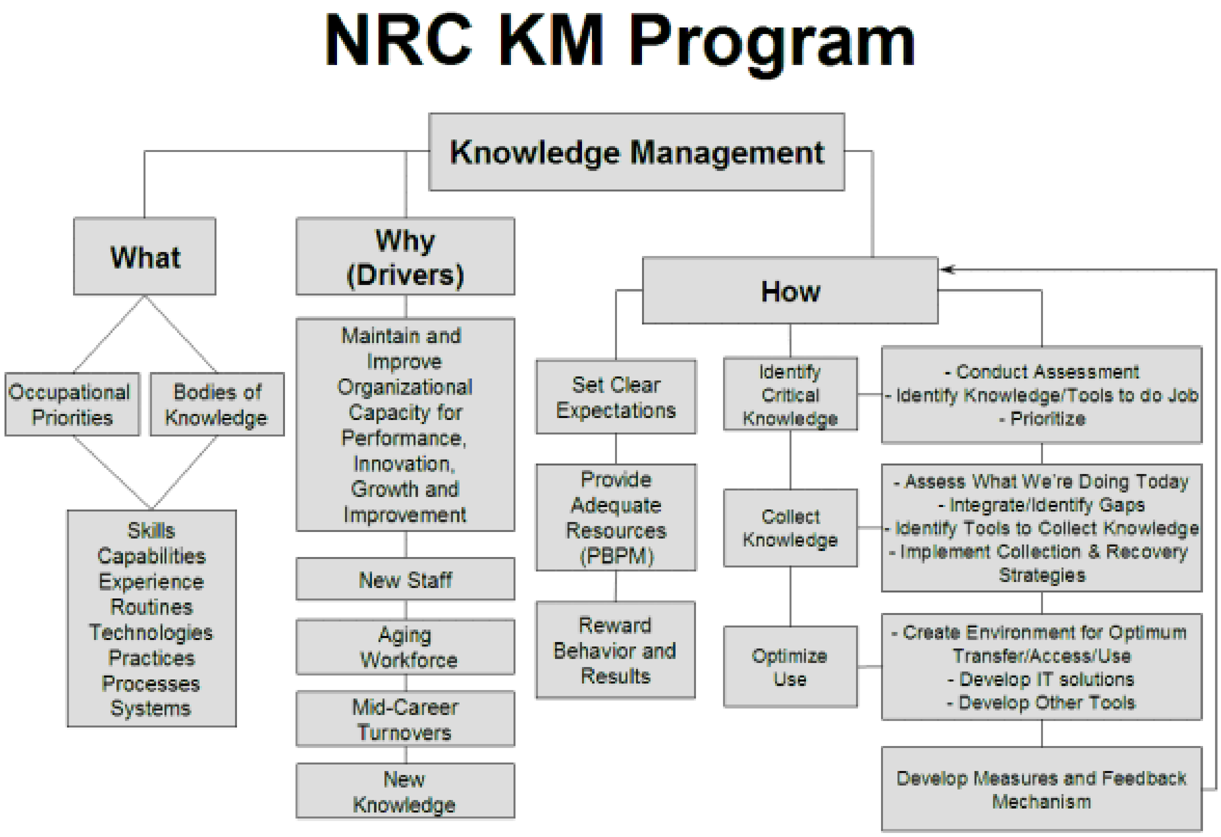

Figure 1.1 Framework of NRC's KM program

An excellent summary of the NRC's KM program is provided in a paper presented at an International Atomic Energy Agency (IAEA) meeting by J. Linehan of NRC, "A Framework for Knowledge Management at the U.S. Nuclear Regulatory Commission," (Ref. 2).

NRC established a KM Steering Committee (KMSC) in late 2006 to monitor the program's progress and serve as the governing body for KM. The KMSC has a chairman who is designated as the "NRC KM Champion." SECY-07-138, "NRC's Knowledge Management Program Status Update," provided a summary of progress through July 2007 from the time the program was officially started in July 2006 (Ref. 3).

At the 2009 NRC Regulatory Information Conference, Martin Virgilio and Patricia Eng gave a presentation entitled "Knowledge Management at NRC" (Ref. 4) that described the history of the KM Program and the drivers for maintaining agency knowledge. Various factors (e.g., increased retirements, workforce mobility, mid-career transfers, etc.) can contribute to overall agency knowledge loss. Impacts from loss of knowledge have the potential of adversely affecting the successful accomplishment of NRC objectives. Thus, applying KM techniques involving knowledge recovery, knowledge sharing, and information technology tools are being used to preclude or minimize these impacts.

1.5 NRC Knowledge Center

An integral part of NRC's KM Program is its web-based portal, "NRC Knowledge Center" (KC) that was developed as part of an NRC pilot program. The NRC KC provides considerable functionality that enables users to share and access a wide variety of information posted by members of the several communities of practice (CoPs) in a number of technical areas. The KC allows registered members to view posted documents, presentations, and other information as well as post similar information, ask questions of NRC subject-matter experts, conduct online discussions, rate the value of specific pages and content, establish user preferences, identify other CoP participating members, etc.

1.5.1 Access Protocols and Communities of Practice

After accessing the home page of the NRC KC as presented in Figure 1.2, the new user can request an account to have access to the overall KC. After receiving approval for general access to the KC Portal itself, the new user can proceed to join CoPs of interest. By clicking on the “NRC’s Knowledge Center” category under the green box (Figure 1.2), one can “browse” available topics in the KC as presented in Figure 1.3 to list primary categories such as materials, reactors, cross-cutting issues, etc. Under reactors, one sees that LMRs (and HTGRs—high-temperature gas-cooled reactors) are listed as CoPs.

Figure 1.2 NRC KC Portal home page (Ref. 4)

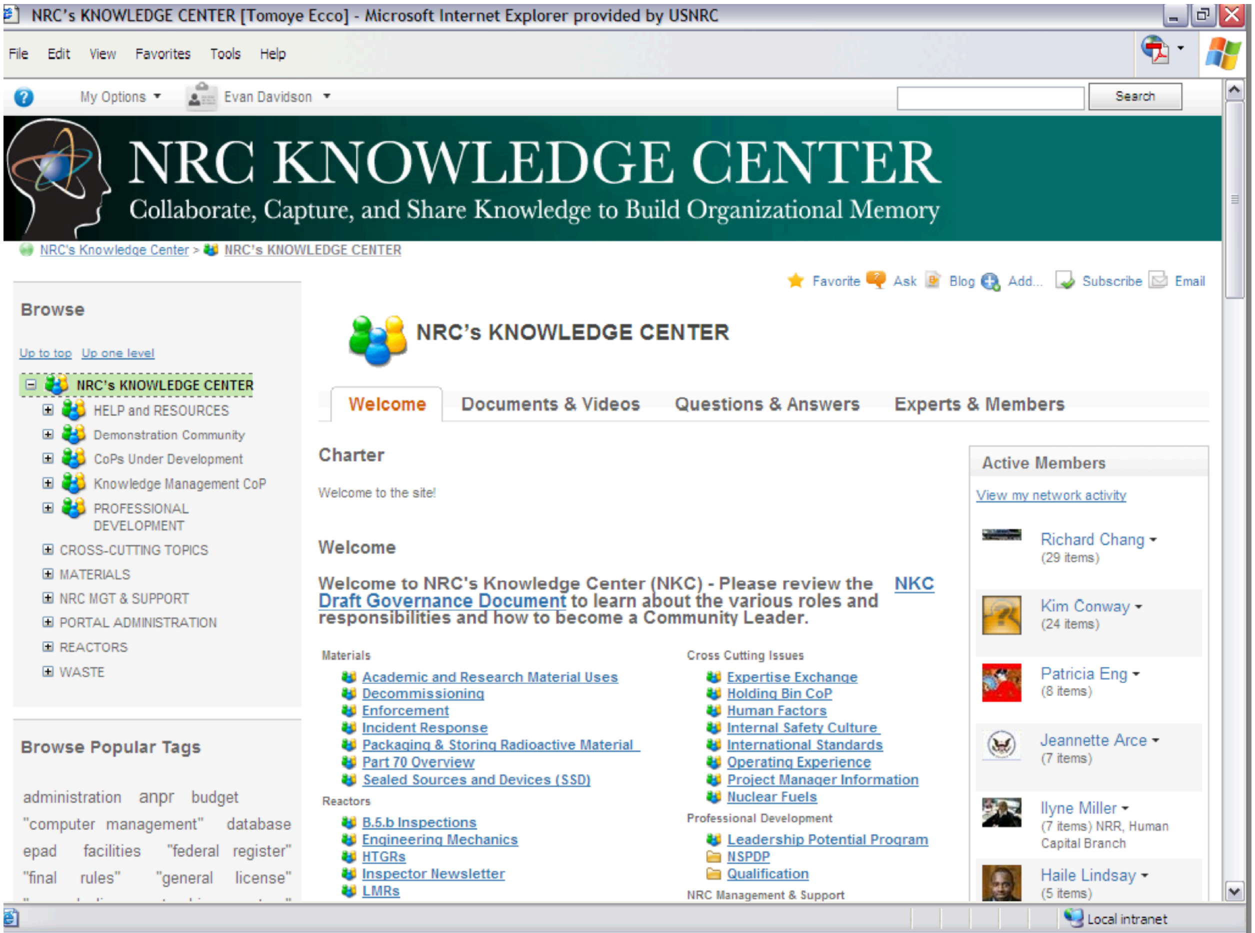

Figure 1.3 Accessing LMR information on the NRC KC (Ref. 4)

1.6 Basic KM Principles and Terms

There are many and varied definitions of KM. Most all typically include how companies and organizations create, identify, obtain, acquire, analyze, and share knowledge to leverage past and current knowledge to accomplish a given business objective. In NRC's situation, the intelligent implementation of KM practices can facilitate and strengthen the timely decisionmaking process that is NRC's responsibility in (1) ensuring the safe operation of existing reactors and (2) evaluating and certifying the safety of the designs of new plants.

The following key definitions are extracted from Attachment 1 to SECY-06-164, "NRC's Knowledge Management Program" (Ref. 1). For the purposes of this document, the definitions of these terms are considered sufficient for an introductory exposure to KM.

> Much of the current literature in the field of KM classifies knowledge as being either **explicit, implicit, or tacit** knowledge. **Explicit knowledge** implies declared knowledge (i.e., knowledge that is conscious to the knowledge bearer). Explicit knowledge is easily codified, which is why it is not a problem for the employee to tell about rules and obviously learned facts. Very often this knowledge is already written down in books, procedures, or training materials.
>
> In contrast to such relatively accessible information, **implicit knowledge** is fact based but difficult to reveal, but it is still possible to be recorded. Usually

knowledge bearers cannot recall this knowledge by themselves, because the information is too obvious to them. When people are asked what they are doing in the morning they might answer “getting up, taking a shower, having a coffee, going to work, checking their e-mail...” without first thinking about their having had to get undressed to take a shower; without thinking about the multiple steps involved in making coffee; and, without thinking about their having had to switch on the computer before being able to read their e-mail. It is generally feasible to convert implicit knowledge into explicit knowledge through documentation.

The third type of knowledge, **tacit knowledge**, is the most difficult to recall and, thus, to transfer. Tacit knowledge includes cognitive and experience-based knowledge about topics such as how to ride a bicycle or how to talk. These examples describe knowledge everybody just has. However, every individual possesses a large amount of tacit knowledge. Employees, for example, tacitly know how they persuade other people, how to behave in different situations, or how to organize a meeting. Such knowledge cannot be completely explained, since it is wholly embodied in the individual, rooted in practice and experience, expressed through skillful execution, and transmitted by apprenticeship and training through watching and doing forms of learning. Tacit knowledge can be observed; however, it is doubtful that all of this knowledge can be converted to explicit knowledge. This fact is why it is said, “We know more than we know that we know.”

One other term of interest by way of introduction of KM is ***Community of Practice (CoP)***. CoPs are groups or collections of “KM practitioners” who are involved in and share a particular area of interest or competence. Typically, these CoPs will share their experiences, contacts, and knowledge. Within the NRC KC, several CoPs have been established as noted in Section 1.5.1.

1.7 About This Report

As background information, Chapter 2 presents general design features of both loop and pool configurations for LMRs, as well as fuel types; discusses their general safety, technical, and operational issues; and summarizes the development of LMRs in the United States. Information is provided on the design, operational history, and purpose of the several LMR test reactors, experimental facilities, operating reactors, and LMR designs that never operated. Chapter 3 discusses the types of documents—safety, licensing, and operational experience—that are included in the LMR portion of the NRC KC as well as presenting the taxonomy that was developed to categorize these documents for retrieval. Chapter 4 summarizes the LMR Desk Reference, an electronic tool designed to provide background and introductory material for NRC staff members who may not be as familiar with LMRs as a resource at such time that an application for an LMR might be submitted to NRC. In Chapter 5, synopses of three white papers prepared by LMR subject-matter experts are provided as a KM and preservation activity on the early development of LMRs, while Chapter 6 outlines the contents of a SFR training course developed under this LMR KM project. Chapter 7 briefly summarizes a companion project to N6472 that focused on compiling and characterizing SFR safety-analysis computer codes that have been previously developed. To complement the “backward look” from a KM perspective, Chapter 8 is a snapshot of current activities associated with LMRs in progress at NRC, DOE, IAEA, standards organizations, and the Generation IV International Forum (GIF). Finally, Chapter 9 presents a summary of this document.

2. GENERAL INTRODUCTION TO LIQUID-METAL REACTORS WITH FOCUS ON SODIUM-COOLED LMRS

2.1 Distinctive Characteristics of LMRs

LMRs have several particularly defining characteristics, including:

- They use no deliberate neutron moderators, resulting in a “fast” or “hard” neutron energy spectrum compared to light-water reactors (LWRs).
- The technology takes advantage of high-energy fission cross sections and smaller parasitic capture cross sections at high neutron energies.
- Proper design allows reactors to breed fissile material from fertile material for resource use or to convert actinide material into short-lived fission products for waste management.
- For their coolant, they use liquid metal, typically sodium, which is about 100 times as effective a heat-transfer medium as water.
- Nominal operating conditions for sodium as a coolant are far below its boiling point and, along with its low vapor pressures, allow low operating pressure.
- Higher operating temperatures result in greater efficiency than LWRs.

2.2 Design Aspects of LMRs

Two reactor design types or configurations have been built and operated in the United States and worldwide—the pool design and the loop design. Characteristics of each are listed below.

Characteristics of pool reactors are:

- The primary heat-transfer fluid (sodium) is kept within the reactor vessel (including the intermediate heat exchanger (IHX)).
- It has a large primary vessel.
- It reduces the impact of a primary pipe break or leak.
- It is currently preferred in the United States and France.
- Its primary vessel is surrounded by a guard vessel.

PRISM and 4S (Super Safe, Small, and Simple) are pool reactors. Experimental Breeder Reactor (EBR)-II was also a pool reactor.

For loop designs, key characteristics are:

- The primary coolant is allowed to leave the reactor vessel, and the IHX is located in the containment area outside the vessel.
- It has reliability improvements—easier to isolate the loop and do maintenance on the IHX.
- The primary vessel is surrounded by a guard vessel.
- It usually requires double-walled piping in areas outside the vessel.
- It is preferred in Japan.

The Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor Project (CRBRP) were loop plants.

Figure 2.1 through Figure 2.3 below compare the two design configurations.

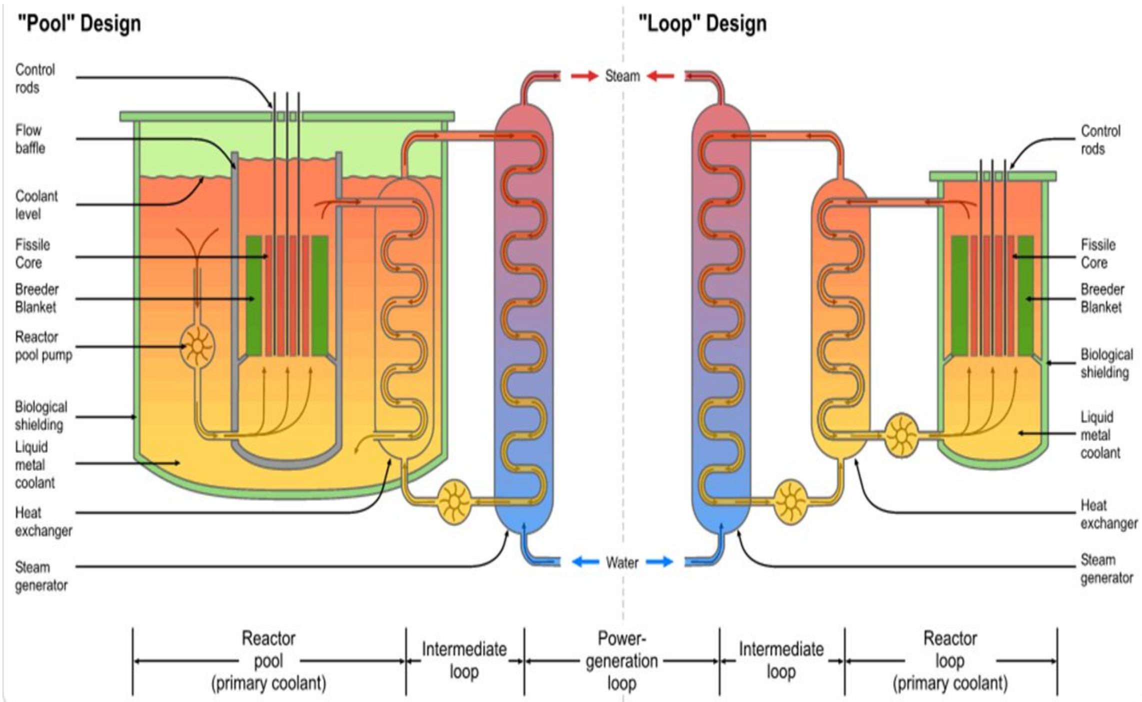

Figure 2.1 Comparison of a pool vs. loop design

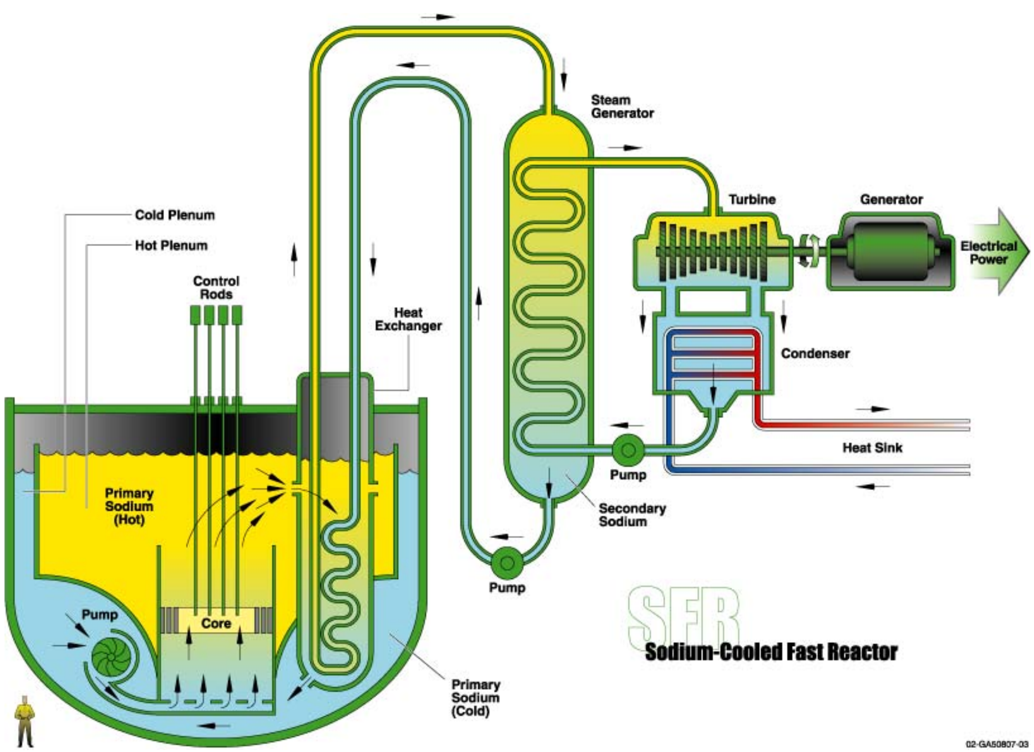

Figure 2.2 Pool reactor layout

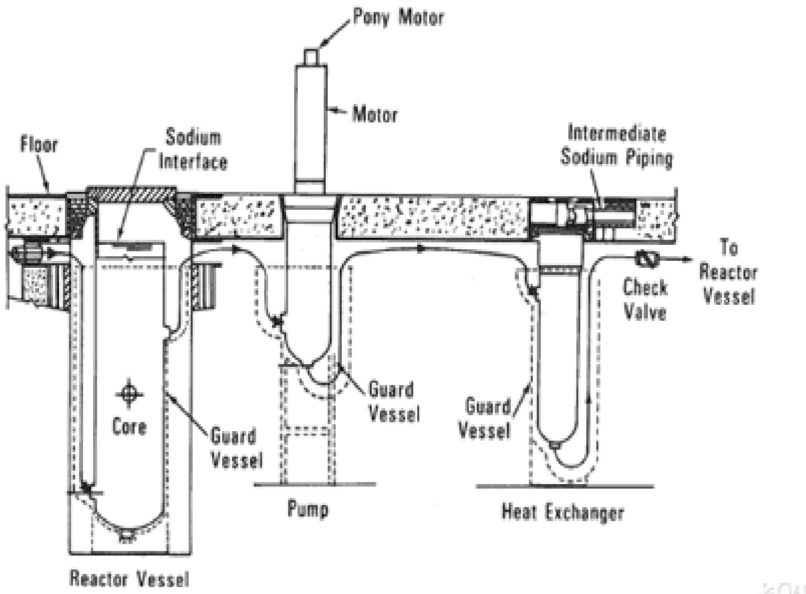

Figure 2.3 Loop reactor layout

Listed below are test reactors that have been built in either a pool or loop design.

Test reactors in pool configuration

- EBR-II
- Phénix (France)

Test reactors in loop configuration

- FFTF
- BOR-60 (Russia)
- Rapsodie (France)
- Jōyō (Japan)
- Kompakte Natriumgekühlte Kernreaktoranlage Karlsruhe (KNK-II, Germany)
- Dounreay Fast Reactor ((DFR), United Kingdom)
- Fast Breeder Test Reactor (FBTR)
- Prova Elementi Combustibile (PEC, Italy)

There are two fuel options considered for an LMR—oxide fuel and metal fuel. Reactors have been built and successfully operated using both fuel types. Currently in the United States the metal fuel is preferred because of inherent safety properties that limit consequences from beyond-design-basis accidents (BDBAs). However, there is limited operational experience with metal fuel outside of the EBR-I, EBR-II, Fermi 1, and DFR reactors. Oxide fuel was initially chosen for LMR power reactors because of experience gained with LWRs and its previous use in the United States and abroad.

Generally Type 316 stainless steel (SS) has been used for the structural materials and also for cladding because of its material properties and resistance to radiation damage. Newer alloys such as HT-9 have been developed for cladding; however, they have not yet been used in a commercial reactor.

Table 2.1 shows the configuration of several commercial power LMRs. Note that Phénix is listed as both a test reactor and a commercial reactor because testing was done at Phénix and it produced electrical power as well.

Table 2.1 Commercial LMR Power Reactors

Pool	Loop
Phénix (France)	BN-350 (Russia)
Prototype Fast Reactor (United Kingdom)	SNR-300 (Germany)
BN-600 (Russia)	Monju (Japan)
Super Phénix (France)	CRBRP (United States, never built)
Commercial Demonstration Fast Reactor (United Kingdom)	SNR-2 (Germany, never built)
BN-1600 (Russia, never built)	Fermi 1 (United States)

Table 2.2 shows typical materials used in an LMR along with the operational conditions.

Table 2.2 Typical Materials Used in an LMR Along with Operational Conditions

System	Materials/Operational Parameters
Fuel	Enriched UO_2, $PuO–UO_2$, or Pu/U-Zr metal alloys or actinides in either
Fertile blanket	$^{238}UO_2$
Clad	Stainless steel Type 316 or advanced alloys (e.g., HT-9)
Coolant	Sodium
Structure	Stainless steel Type 316
Control	B_4C enriched in ^{10}B
Core outlet temperature/pressure	500 to 550°C/~1 atm

2.3 Nuclear Safety, Technical, and Operational Issues Associated with LMRs

Two distinctive aspects of the LMR design have safety implications. The first is associated with the hardened neutron spectrum, in which structural materials and coolant have small absorption cross sections. The hardened spectrum results in a longer neutron mean free path, making neutron leakage reactivity effects much larger than in an LWR. Thus, LMR designs may not be in their optimal nuclear geometric configuration. A small change in the geometry of the core will affect the neutron leakage from the core, which can result in significant changes in reactivity. In addition, the harder neutron spectrum reduces the delayed neutron fraction to about one-half that found in an LWR. Thus, the reactor responds more rapidly to reactivity changes which have implications for the control and shutdown systems.

- The second aspect is associated with the sodium coolant.
- Sodium is reactive in the presence of air and water. Therefore, contact with these media must be prevented. Because the primary system operates at near atmospheric pressure and the steam system operates at high pressure, a significant pressure differential exists across the wall of the steam generator tubes. Because tube failure cannot be ruled out, an intermediate loop is provided between the primary radioactive sodium and the steam system. Generally, the intermediate loop contains sodium as the heat-transfer coolant, so impacts of sodium-water reactions associated with steam generator tube failures are not entirely eliminated; however, they do not impact the primary system boundary.
- Sodium becomes activated during operation, and this will affect maintenance and inspection activities.
- Sodium is opaque at the wavelengths of visible light; therefore, monitoring and in-service inspection require new and unique processes.

In the fast neutron spectra found in an LMR, sodium has a small but still significant absorption cross section. If the temperature of the sodium coolant increases because of an off-normal condition, the sodium density is reduced or, in the extreme, a void or bubble can form. A void or reduction in density of the sodium has three reactivity aspects: (1) the reduced density or voiding decreases the neutron capture in sodium, resulting in an increase in reactivity; (2) the reduction in density decreases moderation, further hardening the neutron spectrum, which increases reactivity; (3) a reduction in sodium density or voiding will also increase the neutron leakage from the core, resulting in a negative reactivity. In parts of the core (generally near the center), the reduction in sodium density will result in a positive reactivity feedback caused by reduced neutron capture and moderation; however, near the edge of the core, the leakage dominates and the reactivity feedback from reduction in sodium density or voiding is generally negative. The design of the core (diameter and axial length) can significantly affect the magnitude and direction of the sodium void reactivity both globally and locally. The impacts of sodium void must be analyzed in any type of accident that would increase the coolant temperature. Finally, sodium’s melting point is about 98 degrees C; therefore, trace heating must be available on all systems that contain sodium to prevent freezing.

Because of the high boiling point of sodium compared to the normal operating temperature of the core (883 degrees Celsius (C) vs. 550 degrees C), the LMR generally operates near atmospheric pressure. This results in much thinner pressure vessels and piping than in an LWR.

2.4 LMR Development in the United States

Several LMRs have been designed and operated in the United States. The first significant electrical power was produced by the EBR-I. The following is a discussion of the design of the various LMRs and their role in developing a commercial LMR fleet of reactors.

2.4.1 Experimental Breeder Reactor-I (EBR-I)

The EBR-I was built at Argonne West in Idaho and went critical in August 1951. It was operated at about 1.2 megawatts thermal (MWt) power and used highly enriched uranium/zirconium metal fuel alloy clad with Type 347 SS and was cooled by NaK (sodium potassium eutectic, liquid at room temperature). The NaK flow was gravity-driven. On December 20, 1951, the first significant electricity was generated by the reactor (it lit four light bulbs). It was the first reactor to demonstrate that breeding was possible. It also demonstrated that reactivity coefficients were important features of fast reactors (fuel, coolant, and structural materials as well as design layout all contributed to changes in reactivity). Four cores were designed and demonstrated in the EBR-I. The second core suffered a partial meltdown caused by a prompt positive power coefficient associated with a planned transient test with the main coolant flow stopped. The fourth core used metallic plutonium/aluminum instead of enriched uranium. Breeding ratios of 1.27 were demonstrated with this core. The metallic cores in EBR-I had limited burnups; as a result, many subsequent LMRs were designed to use oxide fuel based on experience with LWRs. EBR-I was decommissioned at the end of 1963. The reactor has been designated a national historical landmark by both the American Nuclear Society and the American Society of Mechanical Engineers.

2.4.2 Experimental Breeder Reactor-II (EBR-II)

The EBR-II was a demonstration reactor with an operating power of 62.5 MWt. It went critical in 1964 and was shut down in 1994. It typically operated at 19 megawatts electric (MWe), providing heat and power to the Idaho facility. The idea was to demonstrate a complete sodium-cooled breeder reactor power plant with onsite reprocessing of metallic fuel. This was successfully done from 1964 to 1969. The emphasis then shifted to testing materials and fuels (metal and ceramic oxides, carbides, and nitrides of uranium and plutonium) for larger fast reactors. Finally, it became the Integral Fast Reactor (IFR) prototype, using metallic alloy U–Pu–Zr fuels.

The EBR-II was important to the U.S. IFR program, which had the objective of developing a fully integrated system with pyro-reprocessing, fuel fabrication, and fast reactor in the same complex. The reactor could be operated as a breeder or not. IFR program goals were demonstrating inherent safety apart from engineered controls, improved management of high-level nuclear wastes by recycling all actinides, so that only fission products remain as high-level waste (HLW), and using the full energy potential of uranium rather than only about 1 percent of it. All these goals were demonstrated, though the program was aborted before the recycling of neptunium and americium was properly evaluated. IFR fuel first used in 1986 reached 19 percent burnup (compared with 3–4 percent for conventional reactors), and 22 percent was targeted. A further political goal was demonstrating a proliferation-resistant closed fuel cycle with plutonium being recycled with other actinides.

The demonstration of inherent safety was achieved by a series of tests conducted in 1986. These tests demonstrated the ability of the metal-fueled reactor to safely shut down and survive a total loss-of-flow event without scram system activation from full power without any damage to the fuel. A second test demonstrated a total loss of heat sink without scram from full power without any damage to the fuel. The key inherent mechanism for shutting down the reactor was thermal expansion of the core. The high thermal conductivity of metal fuel along with the thermal inertia of the pool design was the basis for safely removing the heat.

In 1994, Congress under the Clinton administration shut EBR-II down. EBR-II is now defueled. The EBR-II shutdown activity also included the treatment of its discharged spent fuel using an electrometallurgical fuel-treatment process in the Fuel Conditioning Facility located next to the EBR-II.

2.4.3 Southwest Experimental Fast Oxide Reactor (SEFOR) 1969–1972

SEFOR was operated by General Electric and funded by the U.S. government through Southwest Atomic Energy Associates, which consisted of 17 power companies and several European nuclear agencies. It operated from 1969 to1972.

SEFOR used mixed oxide fuel (MOX) with sodium coolant, was clad with stainless steel, and operated at 20 MWt. It did not produce electricity. The core consisted of 109 fuel assemblies about 3 centimeters (1 inch) in diameter and 91 centimeters (36 inches) long. The inlet temperature was 371 degrees C and outlet was 438 degrees C. It was constructed to test oxide-fueled sodium-cooled reactors, in particular the effect of core thermal expansion and reactivity feedback associated with oxide fuel heatup during accident conditions leading to stable core conditions. This was successfully demonstrated.

The deactivated facility (fuel and coolant removed) was acquired by the University of Arkansas in 1977. It was used to calibrate equipment and as a research tool for graduate students. SEFOR was designated a national nuclear historic landmark by the American Nuclear Society in 1986, the same year the university stopped using the facility.

In 2009 and 2010 characterization studies were conducted by the University of Arkansas for cleaning up the site.

2.4.4 FERMI 1

The world's first commercial liquid-metal fast breeder reactor (LMFBR), and the only one built in the United States, was the 94-MWe Enrico Fermi Nuclear Generating Station Unit 1. Designed in a joint effort between Dow Chemical and Detroit Edison as part of the Atomic Power Development Association consortium, construction started in 1956, and the plant went into operation in 1963. Fermi 1 used metal fuel with zirconium cladding 0.158 in. in diameter and 31 in. long. It was shut down on October 5, 1966, as a result of high temperatures caused by a loose piece of zirconium which was blocking the molten sodium coolant nozzles. Partial melting caused damage to six subassemblies within the core. The zirconium blockage was removed in April 1968, the core was changed to oxide, and the plant was ready to resume operation by May of 1970, but a sodium-coolant fire delayed its restart until July. It subsequently ran until August of 1972, when its application for renewing its operating license was denied.

2.4.5 Fast Flux Test Facility (FFTF)

The 400-MWt FFTF was in full operation from 1980 to 1993 at Hanford, WA, as a major national research reactor to test various aspects of commercial reactor design and operation, especially relating to breeder reactors. Principally, the FFTF was designed to test materials (cladding and structural) and fuels; it verified the CRBR fuel design. It also demonstrated operation of large-scale components such as pumps and heat exchangers. Its mission was extended to safety testing, especially to examine natural-circulation shutdown heat removal and passive power reduction during loss-of-flow conditions without scram. The safety tests were run at 50 percent power; the results showed that the oxide-fueled reactor provided inherent self-protection during loss-of-forced-flow conditions. The FFTF was not a breeder reactor itself but rather a sodium-cooled fast test reactor because it did not have a breeding blanket. It used mixed oxide fuel 0.23 in. in diameter and about 36 in. in length. The inlet temperature was 360°degrees C, and the outlet was 527 degrees C; the reactor operated only slightly above atmospheric pressure. It was closed down at the end of 1993 because of lack of future missions, and since 2001 it has been deactivated and kept in cold standby with the fuel removed. In May 2005 the core support basket was drilled to drain the remaining sodium coolant, which effectively made the reactor unusable. As the coolant was drained, the system was backfilled with high-purity argon gas to prevent corrosion. In 2006, it was designated as a national nuclear historical landmark by the American Nuclear Society (ANS).

2.4.6 Clinch River Breeder Reactor (CRBR) Project

The CRBR was a joint effort of the U.S. Atomic Energy Commission (AEC) (and its successor agencies, the U.S. Energy Research and Development Administration (ERDA) and DOE) and the U.S. electric power industry (principally the Tennessee Valley Authority (TVA) and Commonwealth Edison) to design and construct a sodium-cooled, fast-neutron nuclear breeder reactor. Three vendors were involved with the design. Westinghouse was the lead supported by General Electric (GE) and the Atomics International (AI) division of North American Aviation. The project was intended as a prototype and demonstration for building a class of such reactors, called liquid-metal fast breeder reactors (LMFBRs), in the United States. The project was first authorized in 1970. After initial appropriations were provided in 1972, work continued until the U.S. Congress terminated funding on October 26, 1983. Increasing costs and growing concerns about global proliferation were the basic reasons the project was terminated. The site for the CRBR was a 1,364-acre land parcel owned by the TVA adjacent to the Clinch River in Roane County, TN, inside the city limits of Oak Ridge, TN, but remote from the city's residential population. The reactor would have been rated at 1,000 MWt, with a net plant output of 375 MWe, and a breeding ratio significantly greater than 1. The reactor core was designed to contain 198 hexagonal fuel assemblies with two enrichment zones. The inner core would have contained 18 percent plutonium and would have consisted of 108 assemblies. It would have been surrounded by the outer zone, which would have consisted of 90 assemblies of 24 percent plutonium to promote more uniform heat generation. The fuel element design was essentially the same as that for the FFTF. The active fuel would have been surrounded by a radial blanket consisting of 150 assemblies of similar, but not identical, design containing depleted uranium oxide; outside the blanket would have been 324 radial stainless steel shield/reflector assemblies of the same overall hexagonal geometry. Construction started at the site when TVA was issued a Limited Work Authorization by the NRC. Later a Construction Permit was issued just before the project was terminated. This was the first and only license issued for a commercial LMFBR by the NRC. All other sodium-cooled reactors in the United States were certified by the AEC as test reactors except for Fermi 1, which the AEC certified as a commercial plant.

2.4.7 Integral Fast Reactor

The IFR was a generic reactor concept based on four technical features: (1) liquid sodium cooling, (2) pool-type reactor configuration, (3) metallic fuel, and (4) an integral fuel cycle, based on pyro-metallurgical processing and injection-cast fuel fabrication, with the fuel cycle facility co-located with the reactor. Much of the technology for the IFR was based on EBR-II, which was the first pool-type liquid-metal reactor. Metallic fuel was successfully developed as the driver fuel in EBR-II. From 1964 to 1969, about 35,000 fuel pins were reprocessed and refabricated in the EBR-II Fuel Cycle Facility, which was based on an early pyro-process with some characteristics similar to that proposed for the IFR. The IFR program was initiated by DOE in 1984 at Argonne National Laboratory (ANL) and Argonne National Laboratory–West (ANL–W) and was terminated with the closure of the EBR-II in 1994.

2.4.8 Advanced Liquid-Metal Reactor (ALMR—PRISM and SAFR)

Following the termination of the CRBR project, the DOE began studies to design a less costly reactor featuring passively or inherently safe features resulting in the ALMR program from 1989 to 1995. Two designs were developed, the PRISM by GE and the SAFR by AI's successor company Rocketdyne International (RI). Both were supported by DOE national laboratories, with ANL providing the principal support. Both designs were reviewed by the NRC and issued preliminary safety evaluation reports (SERs). DOE selected the PRISM design in 1988 for further development. The concept of this design was made up of nine identical reactor/steam generator modules grouped into three power blocks of three modules served by one turbine generator producing 465 MWe. The reactor and steam generator modules were to be factory-fabricated and delivered to a prepared site. The PRISM reactor was a metal-fueled pool design using EBR-II as a basis. It featured passive natural-convection air cooling around the reactor vessel, called the Reactor Vessel Auxiliary Cooling System (RVACS), supplemented by a second natural-convection air system called the Auxiliary Cooling System (ACS) around a helical coil steam generator to remove decay heat. The ALMR program was terminated by DOE in 1995 soon after the NRC review. GE privately continued with the design of the PRISM, renamed it S-PRISM (for Super-PRISM), improved the economics, and explored its use as an actinide waste burner reactor. PRISM is being marketed by GE Hitachi Nuclear Energy as an advanced SMR.

2.4.9 Super-Safe, Small and Simple (4S)

This reactor, designed by Toshiba Corporation, is a 30-MWt or 10-MWe sodium-cooled reactor (a 50-MWe design is available) located entirely below grade in a sealed reactor vessel 3.5 m in diameter and 25 m in length. It consists of 18 hexagonal fuel assemblies (U–Zr alloy fuel at 19.9 percent enrichment) that are 2.3 m in length. The outlet temperature is 510 degrees C. Primary system flow is maintained using electromagnetic pumps. It uses a natural-convection air reactor vessel cooling system similar to the PRISM’s RVACS and a natural-convection air Intermediate Heat Exchanger Auxiliary Cooling System (IRACS). The 4S has a 30-year refueling cycle. The entire sealed vessel is removed and replaced during refueling. The 4S is designed for remote applications and can be used for either electric generation or process heat or both.

3. CHARACTERIZATION OF LMR DOCUMENTS ENTERED INTO NRC'S KNOWLEDGE CENTER

A large volume of information exists within the United States on liquid-metal-cooled reactor (LMR) technology. Much of that information resides in the bibliographic databases maintained by the U.S. Department of Energy's (DOE's) Office of Scientific and Technical Information (OSTI). OSTI's database contains more than 5,000 conference reports and more than 15,000 technical reports on LMRs. In general, the conference reports are open literature publications having no access restrictions other than copyright-related limitations. However, most conference reports are overview or summary documents that do not provide detailed information that may be needed to support NRC KM objectives for potential licensing purposes. Many of the technical reports (as well as reports dealing with safety or design of Clinch River Breeder Reactor (CRBR), Power Reactor Innovative Small Module (PRISM), Sodium Advanced Fast Reactor (SAFR), or Advanced Liquid Metal Reactors (ALMRs)) and supporting technical information were considered for entry into the LMR portion of the U.S. Nuclear Regulatory Commission's (NRC's) Knowledge Center (KC). Those documents that were characterized and ultimately entered into the NRC KC are all publically available documents.

3.1 Document Sources

In terms of content, the focus was on the identification of documents that contained information on design parameters for LMRs, associated operating experience, regulatory/licensing/safety issues, and testing facilities. Information on international LMRs and experience was included as well when it was published in the open literature.

The following types of documents, as noted in bold, were surveyed to identify candidate documents, papers, and presentations to add to the NRC KC:

- **Technical Reports on U.S. Reactors and Facilities:** Included examining older information related to Experimental Breeder Reactor (EBR)-II, Fast Flux Test Facility (FFTF), CRBR, and Fermi 1.
- **Conference Proceedings:** Up until 1998, the ANS held an International Advanced (Fast) Reactor Safety Conference every 3 years where full papers were published. These papers have a copyright, which is held by the ANS and cannot be reproduced without their permission. Permission was sought and obtained to enter the most important papers from these conferences. Other conferences sponsored by Institute of Electrical and Electronics Engineers (IEEE), American Society for Testing and Materials (ASTM), American Society of Mechanical Engineers (ASME), and International Conference on Structural Mechanics in Reactor Technology (SMiRT) were considered as potential sources of LMR information as well. Some international conferences sponsored by the International Atomic Energy Agency (IAEA) have also published information (see IAEA below).
- **NUREG and NUREG/CR reports:** These documents tend to contain confirmatory information related to FFTF, CRBR, PRISM, and ALMR licensing activities and are most likely very applicable to the current reactor program. They are open literature publications.
- **Journal Articles:** Much information was distilled from technical reports and published in technical journals such as *Nuclear Science and Engineering*, *Nuclear Technology*,

Nuclear Safety Journal, *IEEE Spectrum*, and other IEEE journals and ASTM and ASME technology–specific publications. Generally this information is copyrighted and cannot be reproduced, except for the *Nuclear Safety Journal*, which was a government-funded journal and is not copyrighted. Again, as with the conference proceedings noted in the second bullet point, Oak Ridge National Laboratory (ORNL) sought and obtained permission to allow articles from these journals to be included on the NRC KC under LMRs.

- **Reactor Development and Technology (RDT) Standards:** Because of the absence of LMR-related consensus standards developed by professional societies and the American National Standards Institute (ANSI), DOE and especially its predecessor agencies (the U.S. Energy Research and Development Administration (ERDA) and the U.S. Atomic Energy Commission (AEC)) developed a set of best practices and processes that were called RDT (Reactor Development and Technology) Standards. RDT was the organization in AEC and later in ERDA that managed the LMR program. These standards have been renamed Nuclear Energy standards to reflect the currently responsible DOE organization. These standards were used in the design of EBR-II, FFTF, and CRBR. Most were written before 1983. They deal with specific issues such as testing of components, use of materials in certain applications, measurement techniques, handling of sodium, nondestructive examination (NDE), and welding techniques. Some were adopted by professional societies (ANS, ASME, ASTM, and IEEE) and incorporated in their standards. The program was administered by ORNL and was terminated by DOE in the early 1990s.
- **Operational Experience:** EBR-II and FFTF have extensive operational histories. In most cases these are well documented in progress reports, incident reports, and, in some cases, presentations at meetings (conferences). These data are usually not restricted but they are not widely published. Thus, interaction with subject matter experts may be the best way to find out about experience. Information also exists on international experience with Phénix, Superphénix, Jōyō, and Monju, usually in the form of presentations at international meetings.
- **IAEA Publications:** IAEA has a Knowledge Management Base with a large collection of information on LMRs. It contains information on databases that have been developed on operational experience, design information on LMRs, R&D information, and safety. Many of the documents are summary in nature, but some have a significant amount of detail. It has unrestricted access, and the documents can be easily downloaded. Given that the IAEA resources exist and are accessible directly by NRC staff, it was decided to highlight these as a resource but to use the limited resources for entering other documents into the LMR portion of the NRC KC that were not as easily accessible.

3.2 LMR Taxonomy for Categorizing LMR Documents in NRC KC

ORNL developed a taxonomy for indexing LMR information as an appropriate method for compiling, categorizing, and searching this information on the NRC KC. The objective was to develop a taxonomy that was similar in structure and general content as was done for an earlier project on HTGRs that was conducted by ORNL in which HTGR-related documents were entered into the NRC KC (Ref. 5). Obviously this LMR taxonomy includes design features and operational characteristics associated more specifically with LMRs. Table 3.1 presents the LMR taxonomy. The “main topics” as listed in the left-hand column represent the principal categories/framework by which the documents were entered into the NRC KC and represent the “Topics” as noted with the folder icons under the LMR section in the NRC KC. The right-hand column of Table 3.1 merely provides subtopics on the subject matter associated with the main topic and thereby serves to amplify the main topics, as well as aiding an NRC KC user searching for specific LMR subtopics. The main topics are directly searchable, while the subtopics are not except through free-form text searches. The numbers in the second column indicate how many documents were added to the NRC KC, with the “Main Topic” categorized as the principal topic. Note: Under reactor types, the number of documents has been indicated as “0”, because these documents have been included under either one of other more specific main topics in the table.

Table 3.1 LMR Taxonomy for the NRC KC

Main topic	Number of documents added	Subtopic
Fuel (oxide)	9	Fuel Testing/Qualification Basis Fuel/Coolant/Cladding Compatibility Fission Product Release Fuel Temperature Irradiation Behavior Physical Properties Fabrication
Fuel (metal)	4	Fuel Testing/Qualification Basis Fuel/Coolant/Cladding Compatibility Fission Product Release Fuel Temperature Irradiation Behavior Physical Properties Fabrication
Cladding	1	Physical Properties Fuel/Coolant Compatibility Irradiation Behavior
Coolant (Na/NaK/Li)	1	Physical Properties High-Temperature Creep Fuel/Cladding Compatibility Irradiation Behavior
Structural Materials	1	Intermediate Heat Exchanger (IHX) Piping/Pumps/Valves Steam Generator Reactor Internals (includes refueling machine) Vessel(s) Containment Materials Qualification Physical Properties
Reactor Types	0	Pool Loop
Reactor/Plant-Design/Analysis	6	Nuclear Thermal Hydraulics Balance of Plant Inert Atmosphere I&C Containment Residual Heat Removal Fission Product Behavior Analysis Codes/Simulation

Table 3.1 LMR Taxonomy for the NRC KM (continued)

Main topic	Number of documents added	Subtopic
Refueling/Onsite Storage/Transport (fresh/spent)	0	Issues Equipment Description Transfer/Storage
Accident Analysis	20	PRA Reactivity Feedback Plant Modeling Human Factors Severe Accidents Aerosol Transport (in containment) Fires/Spills Containment Integrity
Consequence Analysis	9	Fission Product/Aerosol Transport (outside containment) Environmental Impact Dose Emergency Preparedness
Structural Analysis	2	Containment Analysis Aging Core Catchers
Safety/Regulatory Framework	30	Issues from Previous Regulatory Reviews (CRBR, PRISM, SAFR, ALMR) National Standards (RDT and Consensus) International Regulatory Experience (IAEA, other) Classification of Safety Systems Acceptance Criteria Event Classification/Categories/Selection Inherent Safety Mechanisms Acceptance Criteria Defense in Depth In-Service Inspection (NDE) Quality Assurance/Testing/Inspection Equipment Qualification
Safeguards/Security	0	Issues with Materials Having High Actinide (Pu/Np/Cm/Am) Content
Small Test Reactors/Experimental Facilities	16	FFTF, TREAT, EBR, etc. (United States and foreign)
Operating Experience	10	Operational Events/Incidents/Data Maintenance/Testing/In-Service Inspection Reactor Startup/Shutdown/Transients

Appendix A lists the title and associated "main topic" for each of the 125 documents, papers, and presentations that were added to the LMR section of the NRC KC. While some of the categories were not used because a limited number of documents were added to the KC, this list of “main topics” should be flexible and detailed enough to support the addition of documents in the future when the primary subject of some documents considered for addition might fit appropriately into one of the topic areas.

3.3 Accessing and Retrieving LMR Documents in the NRC KC

Following from Figure 3.1, clicking on the LMRs link under the category of Reactors leads to the LMR section of the KC as presented in Figure 3.1. This screen describes the purpose of and membership guidelines for the LMR CoP. In the left-hand panel under the LMRs subcategory are listed the main topics or taxonomy for LMRs as just described in the preceding section. Any of these sub-topics can be selected to retrieve and view summary information on any given document, and then ultimately display the full document.

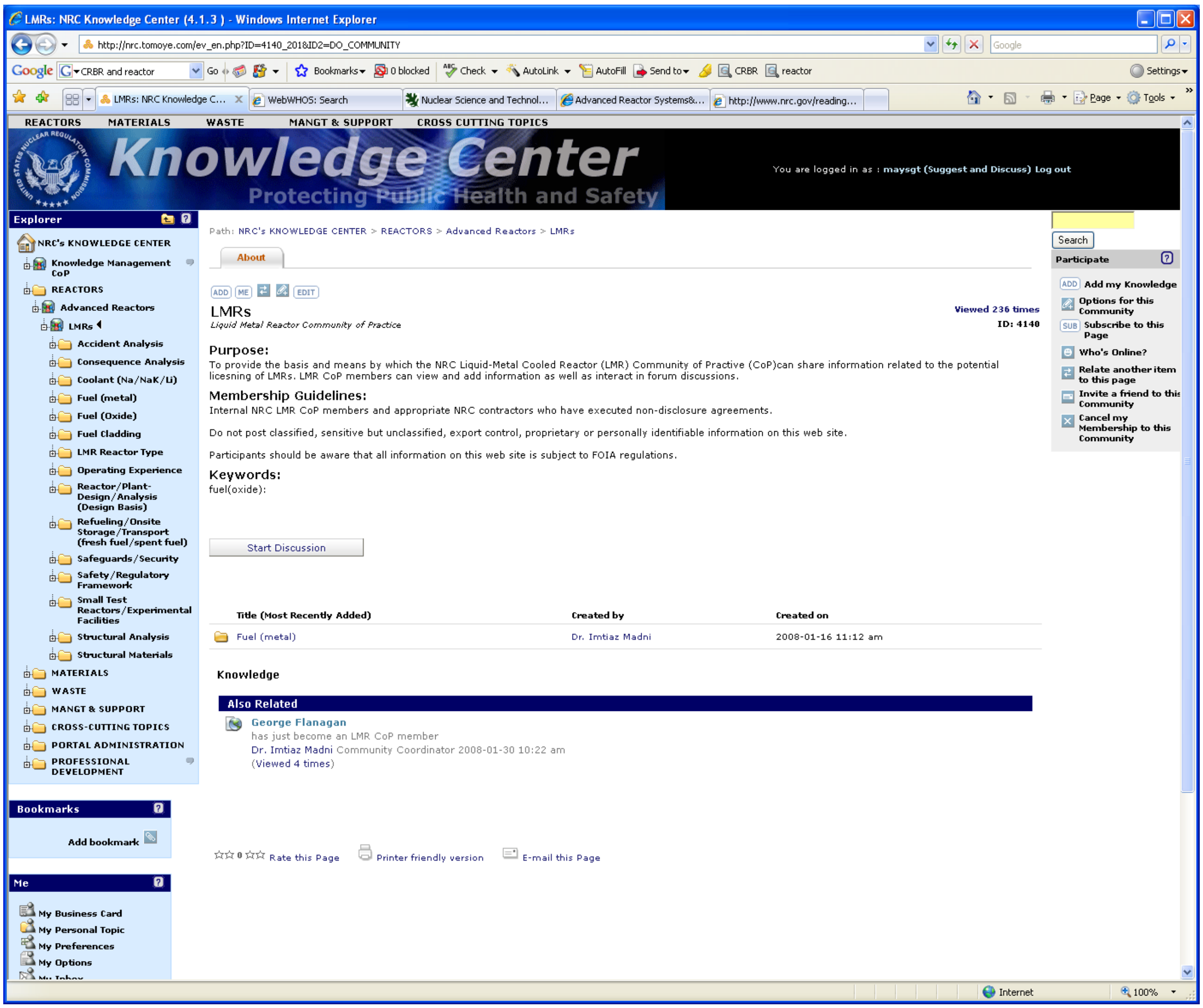

Figure 3.1 Screen for LMR CoP in KC

Once having selected a particular document, a “data sheet” then is displayed as presented in Figure 3.2 that provides pertinent details on the document—a brief abstract, a comment on the benefit and value of the documents, author(s), and keywords. The document can then be “opened” or “saved” by clicking on the filename as listed under the “File” category. Figure 3.2

presents, as an example entry, this information for a report entitled *Experimental and Design Experience with Passive Safety Features of Liquid Metal Reactors*.

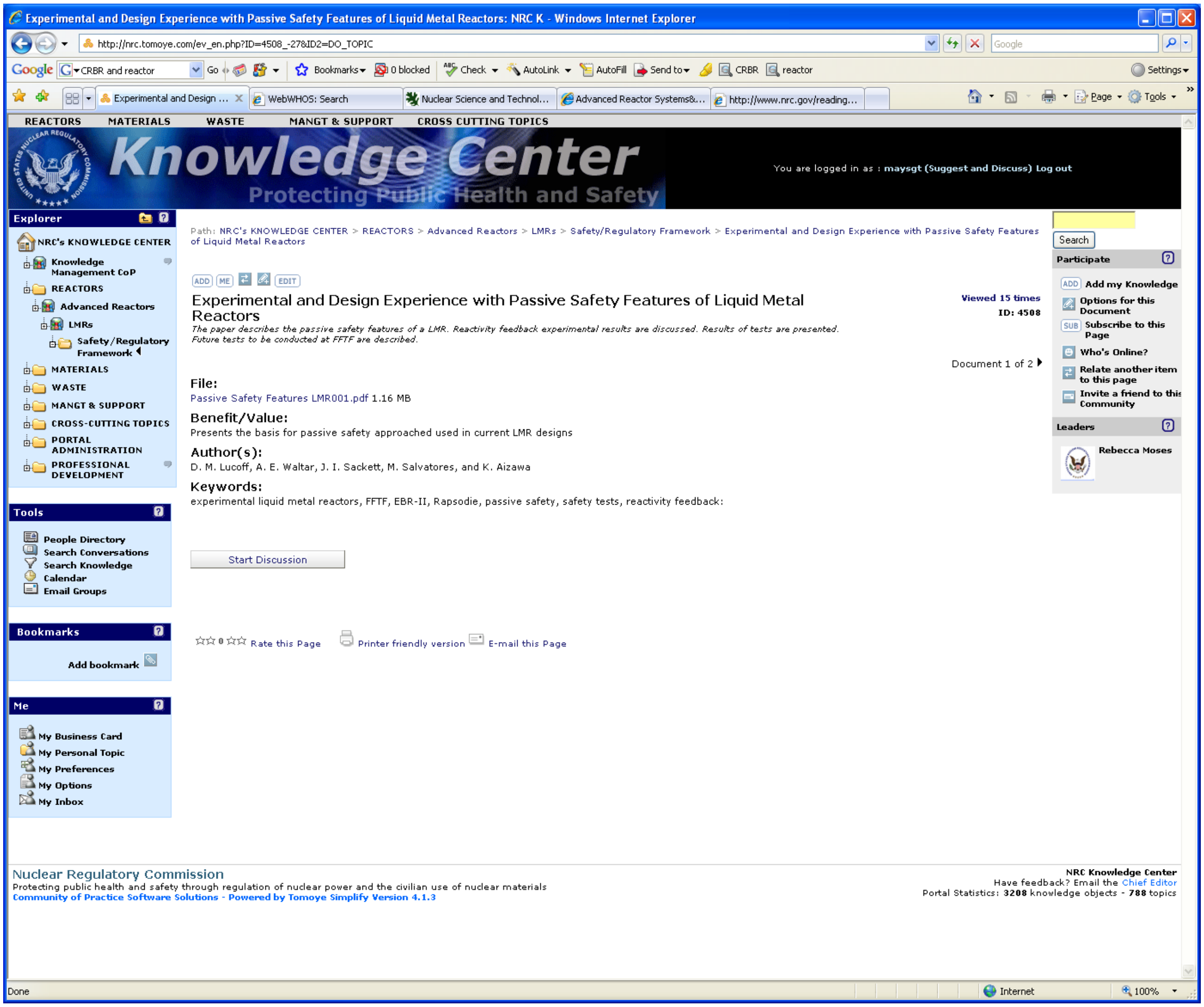

Figure 3.2 Example record for document entered into the "Safety/Regulatory Framework" topic for LMRs

4. LMR DESK REFERENCE

The objective in developing the liquid-metal-cooled reactor (LMR) desk reference (DR) was to prepare an electronic reference tool to provide quick access to background information on LMRs that might assist U.S. Nuclear Regulatory Commission (NRC) staff in becoming more familiar with LMRs in general, and not only compiling but also providing some search capability of the information within the DR. This DR contains summary design information for LMR commercial and demonstration reactors worldwide and information about safety- and licensing-related issues, operating experience, LMR subject matter experts, organizations with LMR experience and expertise, and links to other internet-based resources for additional LMR information.

This section of the report describes the content of the DR. The DR itself is a self-contained Portable Document Format (PDF) file that the user can navigate after opening the “Bookmarks” options for Windows PCs (or the “Table of Contents” option for computers running Apple Mac OS or its successors) to display the introductory page of the DR as presented in

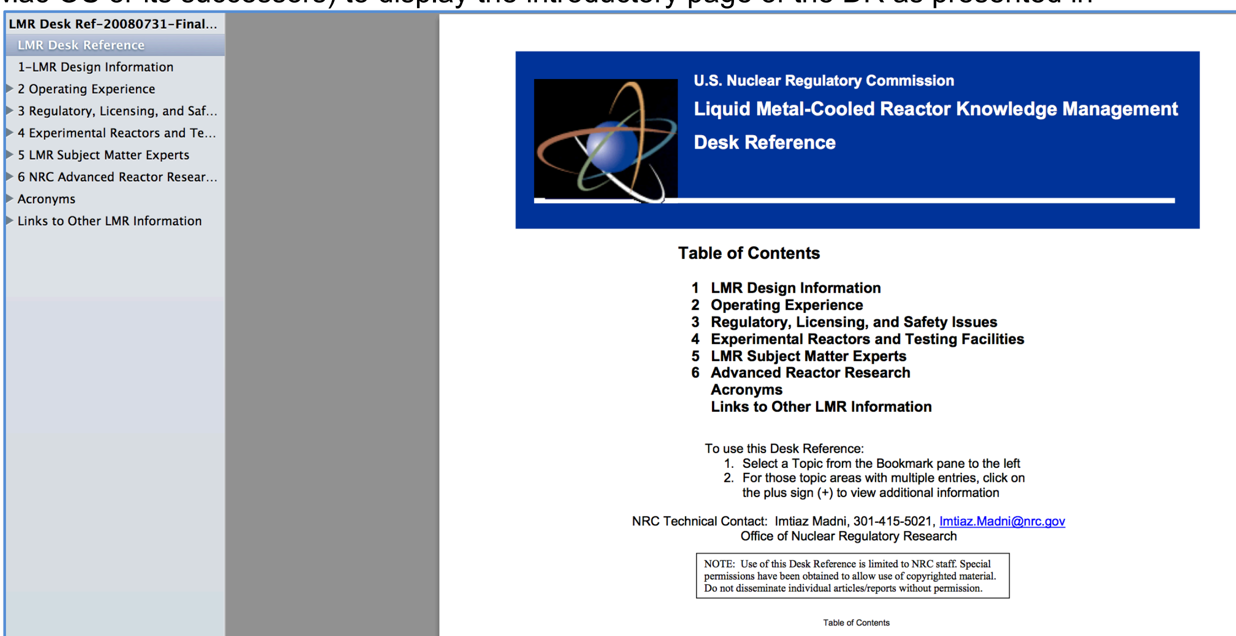

Figure 4.1. On the left-hand side are the bookmarks or table of contents illustrating the initial layer of organization as Chapters 1 (LMR Design Information) through 6 (Advanced Reactor Research) along with an acronym list and a list of links to other LMR information. Clicking on each “Chapter” then provides additional information on the content in that chapter in the form of papers, articles, tables, etc., that can be “clicked” to view. The user can also click on the Chapter Number Links on the actual page to access the initial page of each chapter.

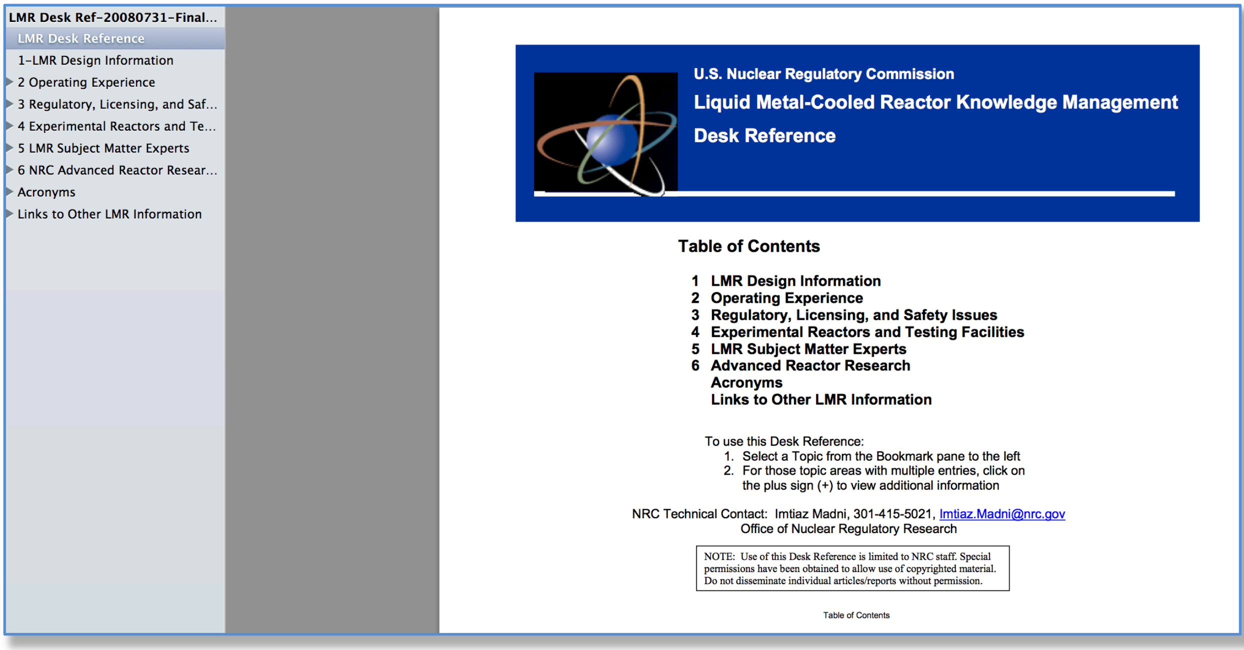

Figure 4.1 Introductory screen of LMR Desk Reference Tool

Figure 4.2 shows the Table of Contents or Bookmarks (left-hand portion of the screen) expanded after having clicked on both Chapter 3, “Regulatory, Licensing, and Safety Issues,” and Chapter 4, “Experimental Reactors and Testing Facilities” to present articles, papers, and tablular data as indicated. For example, the second entry under Chapter 3 is a paper entitled “Lessons Learned from the Licensing Process for the CRBRP” from a 1990 American Nuclear Society meeting on LMRs.

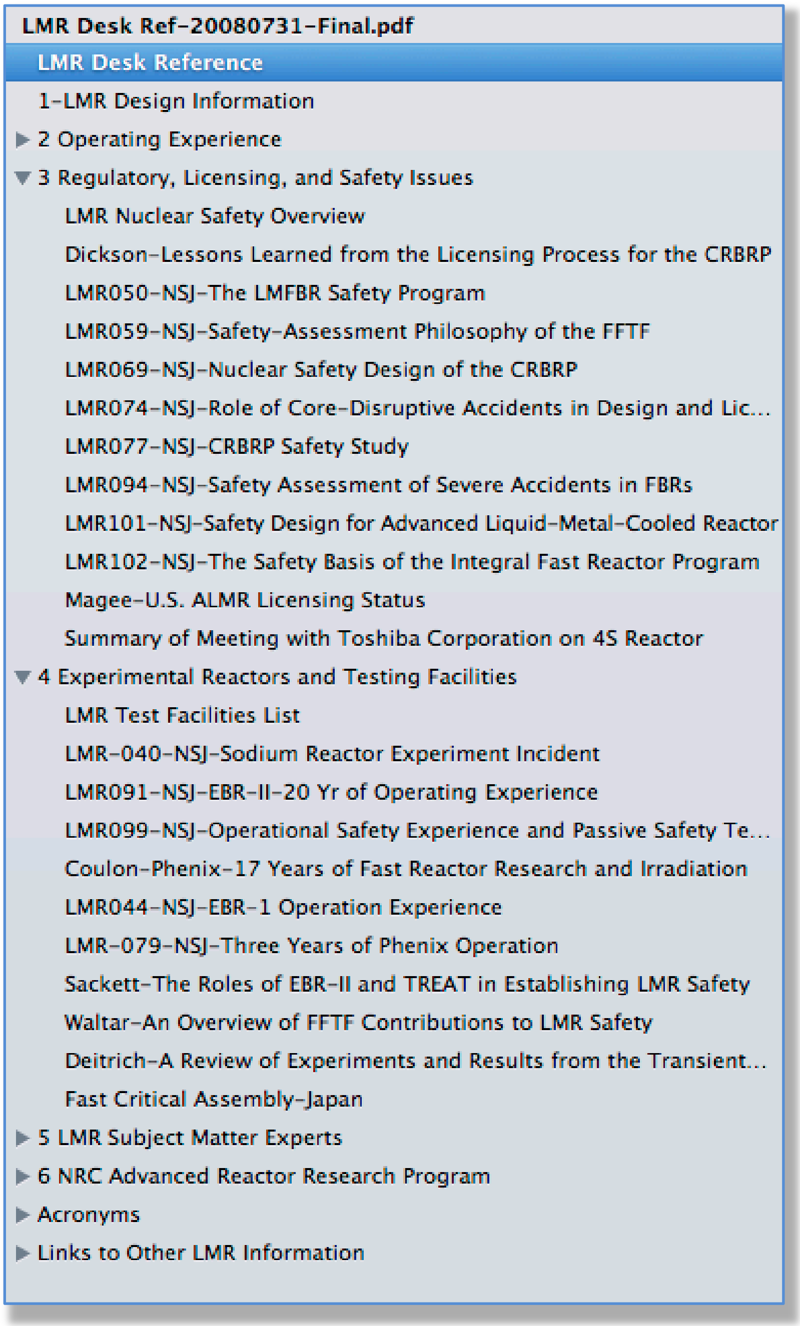

Figure 4.2 Expansion of Chapters 3 and 4 content of Desk Reference

Clicking on this entry then brings us the full paper that provides a summary of the licensing process and interactions with NRC on the CRBRP. Figure 4.3 presents the first page of the paper noting decisions on containment type, a ground acceleration value for a safe shutdown earthquake, early site preparation, limited work authorization, and hearings that involved both the Atomic Safety and Licensing Board and the Advisory Committee on Reactor Safeguards.

From LMR: A Decade of LMR Progress and Promise, ANS Winter Meeting, Washington, DC, November 11-15, 1990.

LESSONS LEARNED FROM THE LICENSING PROCESS FOR THE CLINCH RIVER BREEDER REACTOR PLANT

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ABSTRACT

This paper presents the experience of licensing a specific liquid-metal fast breeder reactor (LMFBR), the Clinch River Breeder Reactor Plant (CRBRP). It was a success story in that the licensing process was accomplished in a very short time span. The actions of the applicant and the actions of the U.S. Nuclear Regulatory Commission (NRC) in response are presented and discussed to provide guidance to future efforts to license unconventional reactors. The history is told from the perspective of the authors. As such, some of the reasons given for success or lack of success are subjective interpretations. Nevertheless, the authors' positions provided them an excellent viewpoint to make these judgments. During the second phase of the licensing process, they were the CRBRP Technical Director and the Licensing Manager, respectively, for the Westinghouse Electric Corporation, the prime contractor for the reactor plant.

1. INTRODUCTION

The Clinch River Breeder Reactor Plant was the only liquid-metal reactor plant to undergo the rigorous modern NRC licensing process. The licensing process on CRBRP was started in 1974 but was aborted in the spring of 1977, when President Carter directed the NRC to cease interactions with the CRBRP project. Although interactions with the NRC and the Advisory Committee on Reactor Safety (ACRS) were held during this first period, very little progress was made. One hindrance to progress was the belief at that time that CRBRP was to be the first of an entire new generation of reactors, because LMFBRs would be required in considerable quantities in the near future. Accordingly, the applicant, the U.S. Department of Energy (DOE)[a] assisted by Westinghouse Electric Corp., was concerned -- overly concerned in retrospect -- with setting precedent with any agreement. Hence the applicant was disinclined to concede a point that it didn't truly believe absolutely necessary for safety or licensing. It is also fair to note that licensing was not on the critical path for plant completion during the late 1970s, so delays to debate points were not considered deleterious to project cost or schedule. Two decisions of note were reached during this first phase interaction with the NRC:

1. a requirement was established for both a containment plus confinement, and
2. a horizontal acceleration of 0.25g zero period amplitude was selected for the safe shutdown earthquake.

The second and conclusive phase was started in September 1981. It was actually a new start as virtually entirely new staffs in both the NRC and the applicant were by then working on the process. This second phase achieved three major licensing actions:

1. exemption for early site preparation granted by the Commission in September 1982;
2. Limited Work Authorization (LWA) granted in May 1983 (preceded by public hearings in summer 1982, ACRS letter in July 1982, and ASLB findings in March 1983); and
3. positive ASLB conclusions on Construction Permit in January 1984 (preceded by ACRS letter in April 1983, and public hearings in summer 1983).

[a]DOE is used herein to refer to the U.S. Department of Energy and its predecessor agencies, the Energy Research and Development Agency and the Atomic Energy Commission (AEC). Co-applicants were the Tennessee Valley Authority and Project Management Corporation.

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3-Regulatory, Licensing, and Safety Issues

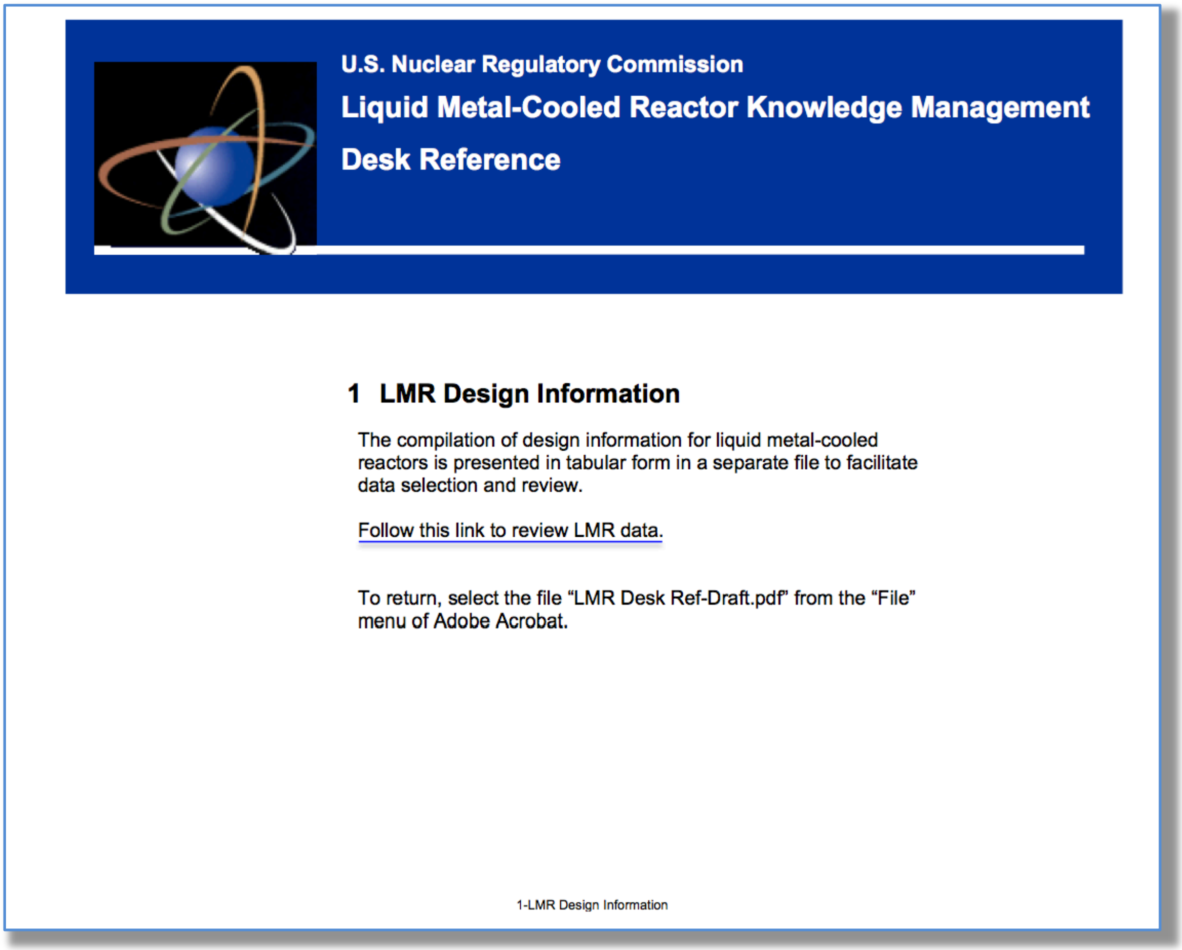
Figure 4.3 Example page from CRBRP licensing paper in Desk Reference

4.1 LMR Design Summary Information—Chapter 1

Chapter 1 of the desk reference (DR) contains extensive tabular data on reactor design parameters for some 33 LMRs—commercial reactors, demonstration/prototype fast reactors, and experimental fast reactors worldwide. The source for this information is an International Atomic Energy Agency (IAEA) publication, *Fast Reactor Database 2006 Update*, IAEA TECDOC Series No. 1531 (Ref. 6). The information in the DR includes design summary information on:

- reactor core and fuel design parameters
- control rods and drive mechanisms
- heat transport system—thermal hydraulics
- components—reactor vessel, pumps, heat exchangers, etc.
- shielding, containment, and safety systems
- safety and control systems
- refueling methods

Appendix B presents a limited number of design parameters for these LMR reactors for the purposes of including information in this report. The DR and IAEA TECDOC contain more detailed information. Clicking on the “Follow this link to review LMR data” hyperlink as presented in Figure 4.4 directs the DR user to a large spreadsheet contained in the DR for access to this design summary information.


U.S. Nuclear Regulatory Commission
Liquid Metal-Cooled Reactor Knowledge Management
Desk Reference

1 LMR Design Information

The compilation of design information for liquid metal-cooled reactors is presented in tabular form in a separate file to facilitate data selection and review.

Follow this link to review LMR data.

To return, select the file "LMR Desk Ref-Draft.pdf" from the "File" menu of Adobe Acrobat.

1-LMR Design Information

Figure 4.4 Chapter 1 of LMR Desk Reference—LMR Design Information

4.2 Operating Experience—Chapter 2

Chapter 2 of the DR contains a subset of those documents entered into the LMR section of the NRC Knowledge Center KC. Several of these documents provide a good introduction to some of the more significant operational experiences and events at LMRs. The documents in Chapter 2 include information on the following:

- high-level description of general design features of LMRs, noting differences between pool- and loop-type designs
- brief summary of the U.S. LMR operational experience
- summary of the Fermi 1 fuel melting incident
- design and operation of fast reactors in the former Soviet Union (USSR) including the following reactors:
  - BR-10
  - BOR-60
  - BN-350

- BN-600
- BN-800 (discussed from a design and development perspective at that time)
- BN-1600 (discussed from a design and development perspective at that time)

- summary of BN-600 LMR operational experience from 1982 through early 1997

Figure 4.5 presents black-and-white photographs of the fuel assemblies and fuel rods taken from the Fermi 1 article that is included in the DR.

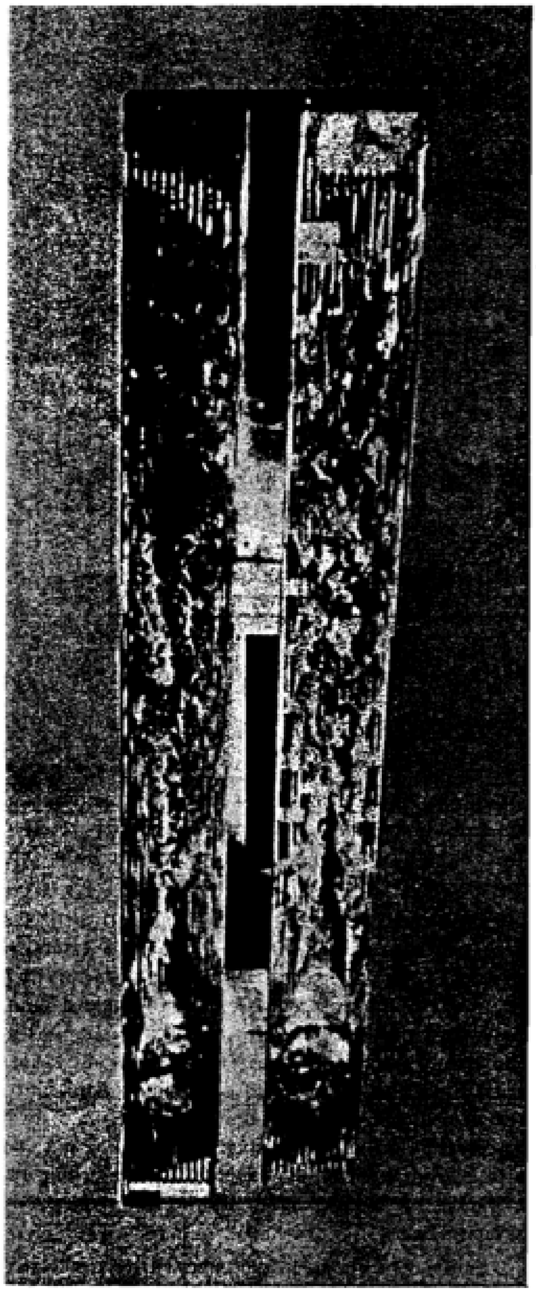


Fig. 6 Longitudinal section through core of subassembly M-217.

Fig. 7 Longitudinal section through top of core of subassembly M-098.

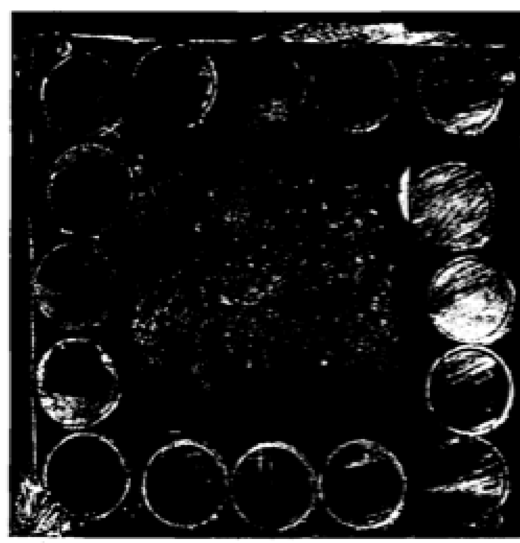


Fig. 8 Transverse section through lower axial blanket of subassembly M-098 just below core fuel casting.

Figure 4.5 Photographs from an article in the Desk Reference on the Fermi 1 fuel melting incident

4.3 Regulatory, Licensing, and Safety Issues—Chapter 3

Like Chapter 2, this chapter of the DR includes a subset of documents that were entered into the LMR section of the NRC KC. These documents provide information on regulatory, licensing, and safety issues on a variety of LMR reactors, LMR designs, and experimental facilities. Specifically, Chapter 3 includes information on:

- a brief overview of LMR nuclear safety issues in general
- lessons learned from the licensing process for the Clinch River Breeder Reactor Project (CRBRP) (see Figure 4.3)
- safety assessment philosophy of the Fast Flux Test Facility (FFTF)
- nuclear safety design of the CRBRP
- role of core-disruptive accidents in the design and licensing of liquid-metal fast breeder reactors
- the safety basis of the Integral Fast Reactor (IFR) program
- U.S. Advanced Liquid Metal Reactor licensing status
- a summary of a meeting between the Toshiba Corporation and the NRC on the 4S (Super Safe, Small, and Simple) reactor

4.4 Experimental Reactors and Testing Facilities—Chapter 4

This chapter of the DR contains 10 documents on LMR experimental reactors and testing facilities, plus a table providing summary information on selected LMR experimental reactors and facilities. The documents include some fairly comprehensive summaries on the operation of EBR-I, EBR-II, FFTF, and the Transient Reactor Test Facility (TREAT). Information and basic schematics of Japan's Fast Critical Assembly are presented. Of particular note is a paper entitled "The Roles of EBR-II and TREAT in Establishing Liquid Metal Reactor Safety."

Figure 4.6 is an illustration of the configuration of Japan's Fast Critical Assembly at Tokai designed for studying physics characteristics of fast breeder reactor cores. The facility offers considerable flexibility in simulating various fuel compositions and core geometries.

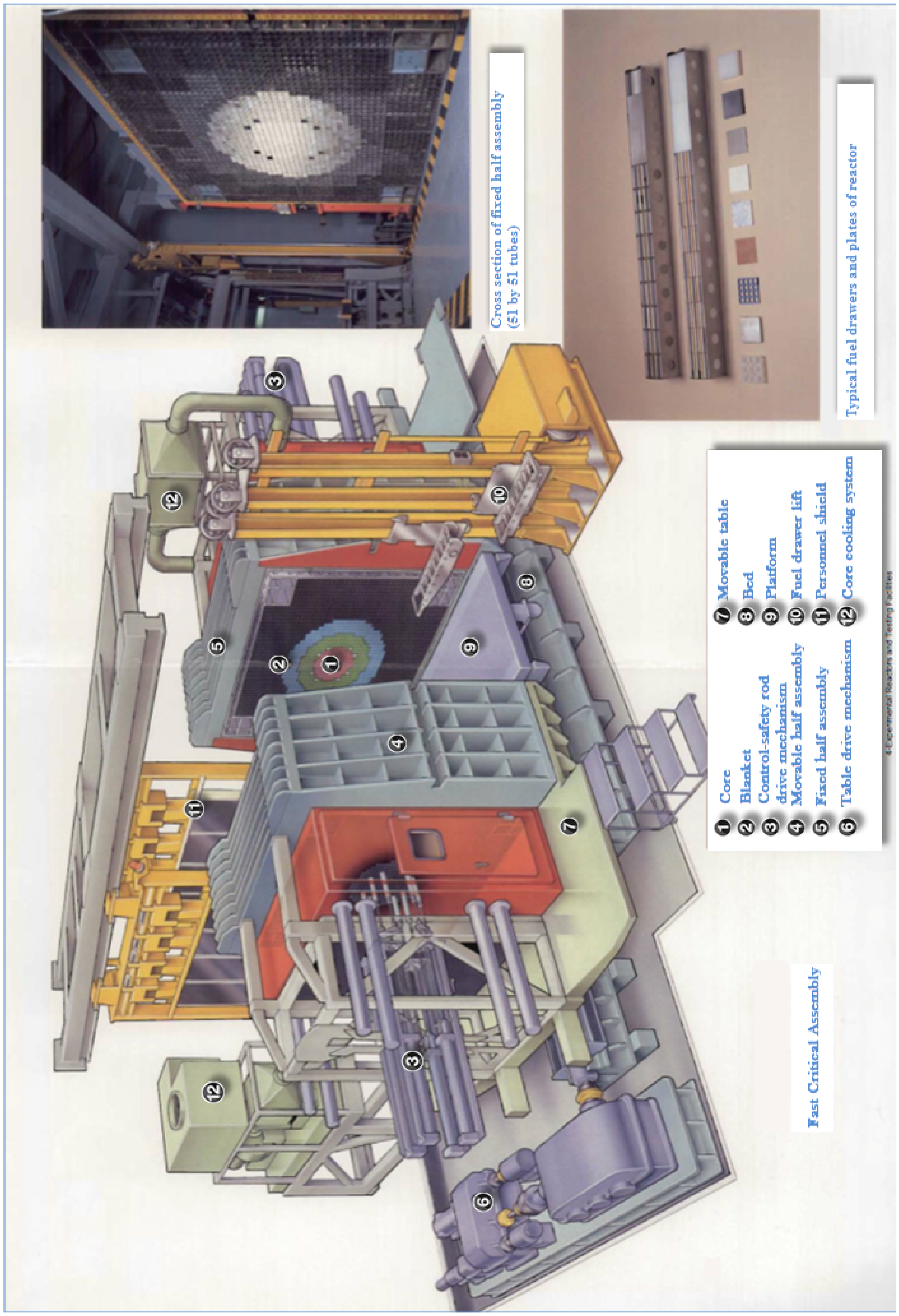


Figure 4.6 Japan's Fast Critical Assembly for evaluating physics parameters of fast breeder reactor cores

5. CAPTURING KNOWLEDGE AND EXPERIENCE FROM THREE LMR SUBJECT-MATTER EXPERTS

In addition to capturing and preserving textual information, part of knowledge management (KM) included one-on-one discussions with experts in the field who provided insights and addressed issues that may not be captured in the published literature. As part of the liquid-metal-cooled reactors (LMR) KM task, Oak Ridge National Laboratory (ORNL) was asked to provide a list of experts who would be willing to present a seminar at NRC based on their having prepared a white paper on a related LMR topic. Three were selected who have different areas of expertise.

Alan Waltar is a past president of the American Nuclear Society (ANS) and has been involved in the LMR safety area, especially in the area of safety analysis development to support the construction of the Fast Flux Test Facility (FFTF) and later the Clinch River Breeder Reactor Project (CRBRP). In addition, he is the author of two books on fast reactors. His paper and presentation were entitled “Key Aspects in Conducting Safety Analysis and Addressing Safety Issues Associated with the FFTF and CRBR.”

John Sackett was the Associate Laboratory Director of ANL–West and was extensively involved with the operation and safety of the EBR-II along with the other facilities at the site. Under John’s management, the EBR-II conducted two major tests as proof of the ability of a metal-fuel sodium-cooled reactor to withstand “loss of flow without scram” and “loss of heat sink without scram” beyond-design-basis accidents without any fuel damage. These tests were conducted at full power. Similar tests were conducted at FFTF for oxide fuel starting at 50 percent power. His paper and presentation were entitled “EBR-II Test and Operating Experience.”

Sterling Bailey worked for the General Electric Company in reactor design for CRBR and later on the 1,000 megawatts electric (MWe) Fast Breeder Reactor Program. His paper and presentation reflect a designer perspective on LMRs and is entitled “Industry Perspectives and Experiences in the Design of Liquid-Metal-Cooled Reactors.”

5.1 Synopsis of Dr. Waltar’s Paper and Presentation

In Dr. Alan Waltar’s paper and presentation, he discussed the basic designs of the FFTF and CRBRP. He described the safety approach for FFTF, focusing on four lines of assurance: (1) prevent accidents, (2) limit core damage, (3) maintain containment integrity, and (4) attenuate radiological consequences. He described the licensing process that was performed by the U.S. Atomic Energy Commission (AEC) (later the U.S. Energy Research and Development Administration (ERDA) and U.S. Department of Energy (DOE)) but supported by the U.S. Nuclear Regulatory Commission (NRC), especially the Advisory Committee on Reactor Safeguards (ACRS). He indicated that the focus of the reactor designer was on design-basis accidents (DBA), which addressed “line of assurance 2,” while beyond-design-basis accidents (BDBA) received significant attention from the regulators. He reviewed the results of the safety analysis and some of the issues that arose during the review of the FFTF’s preliminary safety analysis report (PSAR) and final safety analysis report (FSAR). He discussed the evolution of severe accident analysis from a very conservative Bethe-Tait analysis, resulting in very energetic accidents, to more mechanistic accident analyses, which in turn resulted in acceptance of 150 megawatt seconds (MW-s) energy by the NRC as an appropriate bounding case for containment studies. Reactor vessel analyses indicated that FFTF vessel failure would not occur below 350MW-s energy release. Lessons learned from the licensing of FFTF were as follows: (1) incorporate safety into the design through the lines of assurance approach; (2) use natural circulation, which was demonstrated to work for FFTF and, subsequently, all LMR designs to come along; (3) most emphasis should be on protected accidents (DBA), (4) large oxide-fueled sodium-cooled fast reactors are licensable, and (5) the many inherent safety features (e.g., low pressure, large margin to coolant boiling) provide an exceptionally favorable system with a large resiliency to thwart off-normal conditions.

He then describes the regulatory review history for the CRBR, including the issuance of a construction permit (CP) by the NRC. An operating license was never issued because of the termination of the project over mainly political and economic issues. The licensing leadership of the CRBR project indicated five general lessons learned:

(1) maintain a totally open approach,

(2) keep economics in mind,

(3) all legally allowed actions are possible,

(4) don't be afraid of being sued, and

(5) have the design nearly complete before starting the licensing process.

Two lessons learned were suggested for new, unique reactors: (1) provide tutorials for NRC staff and (2) categorize the General Design Criteria of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," into three groups:

(1) truly applicable,

(2) truly not applicable, and

(3) complied with in principle but not in the same way as for an LWR.

5.2 Synopsis of Dr. Sackett's Paper and Presentation

Dr. John Sackett presented a history of LMR worldwide operational experience along with some major conclusions, both positive and negative.

Positives

- Fast reactor fuel is reliable and safe, whether metal or oxide. Cladding failure does not lead to progressive fuel failure during normal or off-normal reactor operation.
- High burnup of fast reactor fuel is achievable, whether the fuel is metal or oxide. Acceptable conversion ratios (either as breeders or burners) are also achievable with either fuel type.
- Sodium is not corrosive to stainless steel or components immersed within it.
- Leakage in steam-generating systems with resultant sodium-water reactions does not lead to serious safety problems. Such reactions are not catastrophic, as previously believed, and can be detected, contained, and isolated.
- Leakage of high-temperature sodium coolant, leading to a sodium fire, is not catastrophic and can be contained, suppressed, and extinguished. There have been no injuries from sodium leakage and fire (operation at near atmospheric pressure is an advantage to safety).
- Fast-reactors can be self-protecting against anticipated transients without scram when fueled with metal fuel. Load-following is also straightforward.

- Passive transition to natural convective core-cooling and passive rejection of decay heat has been demonstrated.
- Reliable control and safety-system response has been demonstrated.
- Effective systems for purity control and cleanup of sodium have been demonstrated.
- Efficient reprocessing of metal fuel, including remote fabrication, has been demonstrated.
- Low radiation exposures are the norm for operating and plant maintenance personnel (less than 10 percent of that typical for LWRs).
- Emissions are quite low, in part because sodium reacts chemically with many fission products if fuel cladding is breached.
- Maintenance and repair techniques are well developed and straightforward.
- Electromagnetic pumps operate reliably.

Negatives

- Steam generators have not been reliable and are expensive to design and fabricate.
- Sodium heat-transport systems have experienced a significant number of leaks because of poor quality control and difficulty with welds. Also, because of sodium's high thermal conductivity, many designs did not adequately anticipate the potential for high thermal stress during transients.
- Many problems with handling fuel in sodium systems have occurred, primarily because of the inability to visually monitor operations.
- Failure of in-sodium components without adequate means for removal and repair has resulted in costly and time-consuming recovery.
- Sodium-cooled fast reactors have been more expensive than water-cooled reactor systems.

Dr. Sackett's paper describes the EBR-II reactor design and some of the distinctive features that it incorporated, as well as significant EBR-II milestones. EBR-II is the only United-States-operated metal-fueled LMR and the only US-operated pool design. Over its 30-year life, EBR-II carried out four missions: (1) LMR power plant operation with onsite fuel reprocessing, (2) irradiation facility for LMR fuels, materials, and plant dynamic testing, (3) inherent safety and operational reliability testing, and (4) IFR fuel development and plant testing. Dr. Sackett presented details covering EBR-II experience during each of these missions.

Dr. Sackett presented, to particular interest, about the passive safety tests in which an unprotected loss of flow from full power and an unprotected loss of heat sink from full power were performed, both without fuel damage. He described the inherent characteristics of metal-fueled sodium-cooled reactors and how these prevented any fuel damage.

Dr. Sackett discussed lessons learned from 30 years of EBR-II operating experience, with many of these being consistent with the experiences from worldwide operation of LMRs described above.

5.3 Synopsis of Dr. Bailey's Paper and Presentation

In this paper and presentation, Dr. Sterling Bailey discussed the basic issues related to the reactor physics design of a fast spectrum sodium-cooled breeder reactor. In particular, he pointed out how the decreasing capture cross section and increasing fission cross section along with the resonance capture in ^{238}U at higher neutron energies allow the reactor with a blanket to have a conversion ratio significantly greater than 1.0. He then described the AEC/ERDA/DOE/industry fast breeder base technology and design programs that existed in the 1960–1990 time frame, at times employing more than 3000 people with a cost of over $10B. He summarized the experimental/test reactors and the demonstration and prototype reactors that were built both in the United States and abroad. He also indicated that the designers developed over 300 design standards (reflecting good practices) as the designs began to take shape. These were called RDT Standards and most could be used today to design a new LMR. He summarized some of the major design issues facing LMR designers, such as (1) new reactor physics and shielding analytic requirements because of the fast spectrum, (2) new fuels, structural, and material issues, (3) thermal hydraulics issues related to sodium, (4) component issues for pumps, valves, steam generators, shutdown systems, and instrumentation and controls, (5) quality assurance, and (6) safety and licensing obstacles. He summarized some of the major test facilities that were used to verify the performance of the designs, including photos. Many of these facilities have been shut down and many have undergone decontamination and decommissioning.

The complete papers can be found in Appendix C.

6. SODIUM-COOLED FAST REACTOR TECHNOLOGY COURSE

One of the more significant liquid-metal-cooled reactor (LMR) knowledge management (KM) tasks completed was the development of a sodium-cooled fast reactor (SFR) technology course. The SFR is considered to be the most likely type of LMR deployed in the future for which NRC might have to review and evaluate a design certification document. The course is structured in 10 modules, which are described in Section 6.3 below, and is designed to be conducted over a nominal 3-day period.

Section 6.1 briefly discusses the objectives and assumptions of knowledge about SFRs by the NRC staff, while Section 6.2 provides a brief description of the overall course structure. Appendix D presents the SFR technology course agenda.

6.1 SFR Course Objectives and Assumptions

The overall objectives of the course are to (1) provide a fundamental understanding of SFR technology; (2) discuss design features and safety issues in general for LMRs; (3) inform course participants about past operation of LMR reactors, test reactors, and experimental facilities; and (4) summarize the design characteristics of Toshiba's 4S and General Electric's Power Reactor Innovative Small Module (PRISM) LMR designs. This course also compares and contrasts these designs with those of conventional pressurized-water reactors (PWRs) that are operating today. The principal intent is to provide background information to support NRC staff before their preparation for conducting safety assessments, reviews, regulation, and licensing of SFR systems.

These are the basic underlying assumptions regarding the content presented in this course:

- The U.S. Nuclear Regulatory Commission (NRC) staff has extensive capability for regulation of light-water reactors (LWRs) but would need some additional training on SFRs to be more knowledgeable about their design, technology, and safety issues before receiving applications for licensing review of SFR designs.
- The course will be taught over a 3-day schedule.
- The Toshiba 4S and GE PRISM SFRs will most likely be the reactor designs presented in the first applications that NRC might expect to see for licensing SFRs in the future.

6.2 General Organization of the Course

As noted previously, the course is structured in ten main modules. Each module represents an important area related to SFR technology, safety, and overall design features. There are specific learning objectives for each module.

A set of questions developed for each module can serve as review questions for discussion purposes or can be used as a brief quiz to reinforce what was presented by the course facilitators.

The course is planned so that the material and associated quizzes can be covered over three 7-hour days. It is assumed that each "hour" lasts 50 minutes, with 10-minute breaks between class sessions, and lunch is scheduled for 1 hour and 30 minutes each day.

6.3 SFR Technology Course Modules

These are the course topics and brief learning objectives for each of the ten modules:

- Module 1—SFR Introduction
  Objective: Present positive aspects as well as recognized safety issues of SFR systems. Summarize experience with SFRs in the past in various nations and highlight times of operation for experimental, demonstration, and commercial plants. Identify principal design differences between SFRs and LWRs and provide references for subsequent study.

- Module 2—Neutronics
  Objective: Provide an understanding of the neutronic behavior of sodium fast reactors and the fundamentals of fission capture and absorption cross sections to substantiate the need for fast neutron spectra. Show that fast neutrons can be used for breeding of additional fissile materials or burning of unwanted waste products. Show that the fast neutron spectra require greater enrichment of fissile material compared with LWRs and require different configuration of in-core fuel elements.

- Module 3—Coolants and Thermal Hydraulics
  Objective: Provide an understanding of the characteristics of alternative liquid-metal coolants and discuss the reasons for sodium as a coolant of choice for SFRs. Show that the need for fast neutron spectra precludes the use of high moderating coolants, such as water. Identify some other differences between SFRs and LWRs that arise from the differences in coolants.

- Module 4—Fuel Characteristics
  Objective: Develop an understanding of the characteristics of SFR fuel and the reasons for fuel configurations. Show examples of the designs of fuel that have been used in SFRs. Identify different fuel and cladding types and the technology supporting the choice of designs that have been used. Illustrate the fuel-related differences between SFRs and LWRs.

- Module 5—Systems and Components
  Objective: Identify the configurations of systems and components that have been used in SFRs in the past and the advantages and disadvantages of each type. Show the different types of vessels, pumps, heat exchangers, steam generators, and materials that have been used and the reasons for their use. Identify the major issues with each type of component. Show fuel-handling systems and the problems associated with them. Identify instrumentation types and the constraints imposed by the use of sodium coolants. Identify auxiliary systems needed for operation with sodium as a coolant.

- Module 6—Safety and Accident Analysis Module
  Objective: Identify events and accident sequences specific to SFRs and issues associated with the analysis and prediction of plant responses, particularly with respect to releases of fission products that could pose a hazard for the surrounding population and the environment. Show the differences between SFR accident sequences and those of LWRs. List protected events, unprotected events, and severe accidents. Identify and evaluate phenomena affecting the behavior of plants under accident conditions. Summarize the codes used for accident analysis. This module does not address security related events within the scope of DBAs and BDBAs such as intentional acts (i.e., conditional risk) and the resulting consequences.

- Module 7—Licensing Issues
  Objective: Provide important SFR safety analysis and licensing issues that are likely to arise when SFR designs come in for review.
- Module 8—Containment Systems Module
  Objective: Describe evolution of SFR containment systems as experience was gained over time. Present a comparison of SFR and PWR containments. Identify containment configurations that have been used in previous SFRs.
- Module 9—Selected Operational Experience
  Objective: Summarize operational experience with selected SFR commercial-scale plants that have generated significant electrical power and highlight operational and safety issues associated with these activities. Focus on experience with operation, maintenance, and issues affecting shutdowns and restarts.
- Module 10—Summary of PWR, 4S, and PRISM Design Characteristics
  Objective: Highlight detailed key factors of the Toshiba 4S and GE PRISM designs as compared with PWRs. 4S and PRISM are presented in somewhat more detail because these designs might be the most likely designs to be submitted to the NRC for regulatory approval at some point in the future. The course developers chose to compare them to PWRs because of the extensive experience already existing in the NRC for regulating PWRs.
- Module 11—Course Summary
  Objective: Summarize course content and verify whether course objectives were met.

These are some of the key references used in constructing the course (and noted in the course as suggested reading):

- Cochran, T., et al., "Fast Breeder Reactor Programs: History and Status," Research Report 8, International Panel on Fissile Materials, Princeton, NJ, February 2010.
- Graham, J., *Fast Reactor Safety,* Academic Press, New York, NY, 1971.
- International Atomic Energy Agency, "Fast Reactor Database, 2006 Update," IAEA-TECDOC-1531, Vienna, Austria, December 2006.
- International Atomic Energy Agency, "Liquid Metal Cooled Reactors: Experience in Design and Operation," IAEA-TECDOC-1569, Vienna, Austria, December 2007.
- Waltar, A.E., and A.B. Reynolds, *Fast Breeder Reactors*, Pergamon Press, Elmsford, NY, 1981.

7. SFR SAFETY ANALYSIS COMPUTER CODE COMPILATION

7.1 Background

This project, U.S. Nuclear Regulatory Commission (NRC) Job Control Number (JCN) N6975—Sodium-Cooled Fast Reactors Codes—was conducted independently of the liquid-metal-cooled reactor (LMR) knowledge management project (JCN N6472) described in this document, but it is complementary to all the activities in JCN N6472 and is included in this document to provide full coverage of all NRC LMR knowledge management activities. The rest of the material in this chapter provides a summary of and context for the work undertaken in this project. The complete results were documented in an informal report to NRC.

7.2 Context for and Objective of the SFR Computer Codes Characterization Project

Sodium-cooled fast reactors (SFRs) have been major programs in several industrially advanced nations, including the United States, France, Great Britain, Germany, the Soviet Union, and Japan, as well as (more recently) India, Korea, and China. These very large programs dated from the 1950s, with major efforts in the 1970s and 1980s focused on the safety of sodium-cooled fast reactors. One reason for this emphasis was that a sodium fast reactor was not in its most reactive configuration; that is, the reactor could become "prompt critical" and result in large energy excursions as a consequence of certain accident initiators. Also, although sodium has many positive characteristics as a reactor coolant, such as excellent heat transfer capability, ease of pumping, compatibility with most stainless steels, low pressure at operating conditions and high boiling point (thus, large margins to sodium boiling and voiding), it has serious shortcomings, such as energetic reactions with water or air, incompatibility with concrete and similar structural materials, and generation of noxious products, such as sodium oxides and hydroxides as well as incompatibility with normal fire-fighting agents, such as water. Nevertheless, sodium has been the coolant of choice for nearly all the designs for breeder reactors.

The safety of any nuclear reactor depends on the assurance of meeting the following major requirements:

- control of the heat generation process
- transport of heat from the fuel to the ultimate energy conversion system or heat sink
- control of radioactive material release and transport
- containment of any accidental releases
- prevention of or accommodation of severe accidents.

To provide the assurance that these requirements would be met, large expenditures were allocated to safety experiments in order to provide the data to understand the phenomena. In addition to the safety experiments, a large number of safety analysis codes were developed to analyze various aspects of the accident sequences that have been identified for sodium-cooled fast reactors.

SFRs share distinctive attributes associated with their design that may affect the safety of the reactors during certain accidents. In order to minimize the amount of sodium in the core, the fuel is arranged in a triangular array (as compared to a square array for most LWRs) with very tight spacing. The excellent heat transfer characteristics of sodium allow this configuration. Also, the fuel is grouped in hexagonal cans that form the fuel assemblies. This configuration is optimal for fuel use for power production in a fast spectrum reactor during normal operation, but it is not with respect to coolability in degraded states. The accident analysis capability needs to be able to model the neutronics attending fuel relocation in transient conditions because this effect could lead to a more reactive configuration in the accident state than was the case in the operating state. Also, rapid control rod (or shutdown scheme) operability needs to be analyzed to ensure that the reaction is shut down before significant damage occurs. For some sequences, the criticality of a significantly deformed core needs to be analyzed. The generation of heat from the decay of fission products is a key contributor to the heat loads that must be accommodated by the heat transport system. Thus, the decay heat prediction is extremely important. For many of these scenarios, the neutronics is closely coupled with the thermal hydraulics during the accident sequence.

The requirements of the code or code sets are summarized below, grouped in terms of:

- reactivity
- cladding integrity
- thermal hydraulics
- decay heat generation
- mechanical behavior
- chemical reactions
- sodium ejection and fires
- containment
- severe accidents
- plant dynamics

The report did not attempt to review all the codes, but only a number sufficient to analyze the spectrum of accident sequences. The report indicated what types of codes had been developed and examples of such codes. Detailed descriptions of the codes were also included. The sources for the information about the codes include a survey by Madni (Ref. 7) and its associated references, code user manuals, open literature, and summaries from the Oak Ridge Radiation Safety Information Computation Center (modified to meet NRC-specified requirements).

7.3 Summary of SFR Code Capabilities

Table 7.1 is taken from the report summarizing the capabilities of the SFR codes that were included in the review in terms of the phenomena that each code models. The objective was to characterize at least one code (principally U.S.-based) that addressed each phenomenon.

Table 7.1 Code Capability Matrix Showing the Capability of Part of the Code Set To Simulate the Indicated Phenomena

Code capability matrix	Code													
	SAS4A	SASSYS 1	SCALE	CONTAIN	SIMMER	SSC L	DIF3D	ANSYS	MELCOR	MELTSPRD	MAACS	ARGO	MC2	SE2
Phenomena														
Reactivity														
Reactivity feedback at high power		x					x		x				x	x
End-of-life prediction of reactivity feedback		x					x		x				x	x
Burnup control swing/control rod worth		x					x		x				x	x
Relative motion of core and control rods								x					x	x
Reactivity effects caused by gas-bubble entrainment	x	x			x				x				x	x
Core reactivity feedback	x	x			x		x		x				x	x
Core reactivity feedback—fuel motion and core restraint		x					x		x				x	x
Recriticality—potential for energetic events	x	x			x		x		x				x	x
Cladding Integrity														
Integrity of fuel with breached cladding		x												
Thermal Hydraulics														
Single-phase transient sodium flow		x				x			x			x		
Thermal inertia		x				x			x			x		
Pump coastdown profiles		x				x						x		
Sodium stratification		x				x			x			x		
Transition to natural convection core cooling		x				x			x			x		
Core flow distribution in transition to natural circulation		x							x			x		
Decay heat removal system phenomena		x				x			x			x		
Effect of subassembly flow distribution		x				x						x		
Coolant heating and margins to boiling		x				x			x			x		
Fuel dispersal and coolability	x			x	x				x	x				
Decay Heat Generation														
Decay heat generation	x	x	x		x				x					

Table 7.1 Code Capability Matrix Showing the Capability of Part of the Code Set To Simulate the Indicated Phenomena (continued)

Code capability matrix	Code													
	SAS4A	SASSYS 1	SCALE	CONTAIN	SIMMER	SSC L	DIF3D	ANSYS	MELCOR	MELTSPRD	MAACS	ARGO	MC2	SE2
Mechanical Behavior														
Mechanical changes in core structure		x						x						
Intact fuel expansion		x						x	x					
Relative motion of core and control rods		x						x						
Fuel cladding structural integrity at elevated temperatures		x						x	x					
Cooling system structural integrity at elevated temperatures		x						x						
Containment structural integrity								x	x					
Core restraint system performance		x						x						
Chemical Reactions														
Sodium-steam chemical reactions				x					x					
Pressure pulse impacts from chemical reactions				x					x					
Reaction product formation and deposition	x	x		x	x									
Sodium Ejection and Fires														
Sodium spray dynamics				x					x					
Sodium pool fire on inert substrate				x						x				
Aerosol dynamics				x					x					
Sodium/cavity liner interactions				x					x	x				
Sodium/concrete melt interactions				x					x					
Containment and Severe Accidents														
Containment structural integrity				x				x	x					
Radiation release and transport											x			
Plant Dynamics														
Plant dynamics						x			x					

The codes that are capable of addressing various phenomena listed above in Table 7.1 were summarized in the appendices of the report. Information on the following parameters was compiled, thus providing a preliminary characterization for each code:

1. Name of Program
2. Computer for Which Program is Designed and Other Machine Version Packages Available
3. Description of Problem Solved
4. Method of Solution
5. Restrictions on the Complexity of the Problem
6. Typical Running Time
7. Unusual Features of the Program
8. Related and Auxiliary Programs

9. Status and Availability to the NRC
10. Status of Verification and Validation
11. Strengths of Code
12. Weaknesses of Code
13. Other Codes Similar to This Code
14. Machine Requirements
15. Programming Language Used
16. Operating System
17. Other Programming or Operating Information of Restrictions
18. Name and Establishment of Author or Contributor
19. Materials Available
20. Sponsor
21. References

An example of a code description and the associated information compiled for the just-noted 21 parameters is presented in Appendix E of this report for the SAS4A code.

8. CURRENT ISSUES AND INITIATIVES FOCUSED ON LMRS

This chapter provides information on current issues and initiatives focused on liquid-metal-cooled reactors (LMRs) as of early 2013 to form a complete picture for U.S. Nuclear Regulatory Commission (NRC) staff to complement the historical information included in this document and provide a thread for tracking these current issues and initiatives at some point in the future.

8.1 U.S. Nuclear Regulatory Commission

8.1.1 Licensing Status—Pre-Application

No applications for a sodium-cooled fast reactor have been submitted to the NRC since the Clinch River Breeder Reactor (CRBR) license application in the 1970s. The NRC did prepare preapplication safety evaluation reports (PSERs) based on reviews of the Power Reactor Innovative Small Module (PRISM) and Sodium Advanced Fast Reactor (SAFR) pre-application design information descriptions, which were published as NUREG-1368 (Ref. 8) and NUREG-1369 (Ref. 9), respectively. The NRC also recently had preapplication presentations on the 4S concept by Toshiba and the PRISM concept by General Electric-Hitachi.

Presently, as noted on its Web site, the NRC is engaged in preapplication discussion with designers or vendors of three LMR designs:

- Toshiba's Super-Safe, Small, and Simple (4S) concept
  - Electrical output: 10 megawatts electric (MWe))
  - Reactor coolant: sodium
  - Outlet temperature: 510 degrees Celsius (C)
  - Refueling: 30 years (entire reactor module)
- General Electric-Hitachi Power Reactor Innovative Small Module (PRISM)
  - Electrical output: 311 MWe
  - Reactor coolant: sodium
  - Outlet temperature: 500 degrees C
  - Refueling: 12 to 24 months
- GEN4 Energy's Gen4 Module (G4M)
  - Electrical output: 25 MWe
  - Reactor coolant: lead-bismuth
  - Outlet temperature: 500 degrees C
  - Refueling: 10 years (entire reactor module)

Table 8.1 presents the status of these pre-application discussions with each of the three vendors. As indicated in the table, none of these three designers have indicated a firm date yet to NRC as to when they expect to submit an application for approval of a design certification (DC) or combined operating license application (COL).

Table 8.1 Licensing Status and Information for Potential LMRs Engaged in Pre-Application Interactions with NRC

Licensing	LMR designs		
	4S sodium	PRISM sodium	Gen4 lead module lead-bismuth
Letter of Intent	Updated 2/8/2012	Updated 4/20/2011	No information
Licensing Plan	Design Certification	COL prototype (long-term Manufacturing License)	COL (prototypical design) and/or Design Certification
Expected Submittal	Date not specified	Date not specified	Date not specified
Other Information		NRC staff conducted preapplication review in early 1990s that resulted in the publication of NUREG-1368 (Ref. 8).	
Source: NRC website (http://www.nrc.gov/reactors/advanced.html) on advanced reactors			

8.1.2 Advanced Reactor Licensing

In response to a U.S. Congressional request, the NRC prepared and submitted a report on advanced reactor licensing entitled *Report to Congress: Advanced Reactor Licensing* in August 2012 (Ref. 10). The report presents the NRC's strategy and approach for preparing to license advanced reactors. The NRC's anticipated time horizon for planning purposes for receipt of applications is for the next 10 to 20 years and beyond.

The principal activities supporting advanced reactors for licensing purposes, including sodium-cooled fast reactors (SFRs), are to:

- Identify and resolve significant policy, technical, and licensing issues.
- Develop the regulatory framework to support efficient and timely licensing reviews.
- Engage in research focused on key areas to support licensing reviews.
- Engage reactor designers, potential applicants, industry, and DOE in meaningful preapplication interactions and coordinate with internal and external stakeholders.
- Establish an advanced reactors training curriculum for the NRC staff.
- Remain cognizant of international developments and programs.

In the report's Executive Summary, one of the statements made regarding expected activities over the next 10 years, defined as "longer term" in the report, was as follows:

> "Within the longer term, the NRC anticipates continuation of the near-term activities and expanded activities pertaining to liquid-metal cooled reactor designs."

In the context of the report, "near term" is defined as a time frame within 5 years.

8.1.3 NRC Advanced Reactor Research Plan (ARRP)

Previously, I. K. Madni, one of the authors of this report, prepared input on LMRs for inclusion in the NRC's advanced reactor research plan (Ref. 11). That input addressed infrastructure needs for LMRs in the area of reactor systems analysis, which included T/H analysis, nuclear analysis, and severe-accident and source-term analysis. For accident analysis, those events that fall within the licensing basis (design-basis events) and beyond-design-basis events (severe accidents) were included. This input for LMRs is included as Appendix F to this report.

8.2 DOE

The U.S. Department of Energy (DOE) through its Office of Nuclear Energy (DOE-NE) is conducting research on LMRs specifically under its Advanced Reactor Concepts (ARC) program as well as LMR-related research under its advanced small modular reactor (aSMR) program. The following two sections briefly describe the key elements of research presently underway as funded by DOE.

8.2.1 Advanced Reactor Concepts (ARC) Program

The ARC program sponsors research, development, and deployment (RD&D) activities leading to further safety, technical, economic, and environmental advancements of innovative nuclear energy technologies. DOE-NE's objective is to pursue these advancements through RD&D activities at the DOE's national laboratories and U.S. universities, as well as through collaboration with nuclear industry and international partners. Program activities will focus on advancing scientific understanding of these technologies, establishing an international network of user facilities for nuclear RD&D, improving economic competitiveness, and reducing the technical and regulatory uncertainties for deploying new nuclear reactor technologies.

DOE's ARC Program is focusing on both fast spectrum and high-temperature reactors. Specific research and development (R&D) activities are aimed at LMRs. These include projects currently under way or planned that are associated with:

- experimental testing of LMR systems, subsystems, and components in liquid sodium to simulate their operation in a prototypic environment
- evaluation and updating of key LMR safety-analysis codes (e.g., SAS4A/SASSYS-1, CONTAIN-LMR, etc.)
- international collaborations with Japan and France on key safety issues
- advanced materials development examining fast reactor structural alloys and weldments—thermal aging, creep, sodium compatibility, etc.
- experimental work on LMR coolants focusing on thermal shock, liquid-metal freeze and thaw, and corrosion issues

Another ARC project involved forming a task force composed of over 40 researchers associated with SFR safety that prepared a report summarizing the major safety-related R&D activities that are needed to support the licensing of an SFR in the United States. The first recommendation was to preserve and document information from the Atomic Energy Commission/DOE LMR safety program, especially the safety information that resulted from the operation of the Liquid Metal Engineering Center, the Fast Flux Test Facility (FFTF) and (Experimental Breeder Reactor) EBR-II passive safety experiments, and the sodium experiments that were performed in the Transient Reactor Test Facility (TREAT) facility. All of these facilities are now either shut down or (in the case of TREAT) inactive. DOE, through the ARC program, has constructed a web-based interface that contains publicly available reports from these facilities. For those who can meet applied technology requirements, the actual data from the experiment will be available. This knowledge preservation system is expected to be available for public use in late 2013.

8.2.2 Advanced Small Modular Reactors (aSMRs)

The DOE aSMR Program objective is to support laboratory, university, and industry projects to conduct nuclear R&D on capabilities and technologies that are unique and support development of aSMR concepts for use in the mid to long term. SMR Advanced Concepts R&D activities are focusing on four key areas:

- developing assessment methods for evaluating aSMR technologies and characteristics
- developing and testing of materials, fuels, and fabrication techniques
- resolving key regulatory issues identified by NRC and industry
- developing advanced instrumentation and controls and human/machine interfaces

This program was initiated in late fiscal year (FY) 2012. Several of the R&D activities underway are of a cross-cutting nature so that an advanced liquid-metal SMR concept would benefit from their results, such as materials development, formulation of regulatory and licensing approaches for aSMRs, conducting economic analyses, performing experimental testing of passive safety features that are a characteristic of almost all SMR designs, and developing new sensors and measurement systems (given that these systems will likely be operating in more harsh environments given the compact designs for SMRs).

It is anticipated that future work will include the development of a preconceptual design for the liquid-metal SMR concept and an associated reactor technology development plan.

8.3 American Nuclear Society Standard 54.1 for LMRs

As a result of the experience with the CRBR, PRISM, and SAFR reviews, it became necessary to modify the wording of the General Design Criteria contained in Appendix A of 10 CFR 50, which are LWR-based, in order to accommodate the unique aspects of an SFR. These modifications were later captured in ANSI/ANS Standard 54.1, “General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant.” Because of the decline of the SFR program in the United States during the 1980s and 1990s, the ANS standard was withdrawn.

In anticipation of a possible application from Toshiba or General Electric-Hitachi, a revision of ANSI/ANS 54.1 has been initiated by the ANS, this one to be titled “Nuclear Safety Criteria and Design Process for Liquid-Metal-Cooled Nuclear Power Plants.” The revised standard will update the SFR general design criteria developed in the earlier version and will include a section on risk-informing the process used to select and classify the Licensing Basis Events into anticipated operational occurrences, DBAs, and BDBAs. Also included will be a risk-informed performance-based process for determination of the classification of systems, structures, and components (SSCs) and treatment of SSCs based on 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,” as well as a description of the role of probabilistic risk assessment (PRA) in the design process and in development of the defense-in-depth approach used by the design. It is anticipated that this standard will be balloted sometime in late 2013.

8.4 Generation IV International Forum

In addition to the activities associated with the revision of ANSI/ANS Standard 54.1 in the United States, the Generation IV International Forum (GIF) program under /OECD/NEA has drafted a set of safety design criteria for an SFR based on International Atomic Energy Agency (IAEA) Safety Standard SSR-2/1, “Safety of Nuclear Power Plants: Design” that was developed for

LWRs. Once approved by the GIF program, these criteria will be provided to the IAEA for use as a basis for a new Safety Standard for SFRs.

SFRs are one of six reactor technologies for which the GIF member countries are conducting collaborative research. Presently, five SFR “project arrangements” have been approved for work. These include:

- system integration and assessment
- safety and operation, focusing on experiments and modeling for passive systems and accident mitigation
- advanced fuels research looking at high-burnup minor actinide fuels and improved cladding
- component design and power conversion systems
- a demonstration project on minor actinide fuels, including irradiations, in Jōyō

8.5 International Atomic Energy Agency

The IAEA has a Knowledge Management Base that holds a large collection of information on LMRs. It contains information on databases that have been developed for LMRs on operational experience, design information, R&D information, and safety. Many of the documents are summary in nature, but some have a significant amount of detail. It has unrestricted access, and the documents can be easily downloaded. Much of the design and operational information is captured in IAEA-TECDOC-1569, “Liquid Metal Cooled Reactors: Experience in Design and Operation” (Ref. 12).

IAEA member states collaborate on LMR/SFR issues through participation in the IAEA Nuclear Energy Department’s Technical Working Group on Fast Reactors (TWG-FR). The IAEA periodically holds meetings on various technical issues for fast reactors. A meeting to be held in December 2013 is entitled “Technical Meeting on Status of IAEA Fast Reactor Knowledge Preservation Initiative.”

9. SUMMARY

This report documents the U.S. Nuclear Regulatory Commission's (NRC's) efforts and activities to develop and compile information on liquid-metal fast breeder reactors (LMRs), particularly sodium-cooled fast reactors (SFRs), as part of a concerted knowledge management program for LMRs. At the current time, the NRC is engaged in preapplication discussions with the vendors of two LMR designs. In anticipation of possibly receiving an application for a design certification (DC) from either of these two vendors or an application for a combined operating license (COL) from an applicant such as a utility sometime in the future, the NRC determined that it would be advisable to collect and organize key documentation related to design, operation, safety, and licensing into one place as a set of references to orient NRC staff who may not be as familiar with LMRs as they are with light-water reactors (LWRs). Thus, this report (1) documents the LMR knowledge management (KM) activities conducted under the sponsoring NRC Office of Nuclear Regulatory Research (RES) program and (2) integrates the results into one useful resource. In some instances, information will be directly included in this document; in other instances, this document will refer the reader to other resources and tools accessible to NRC staff. The bases for and background of the development of an NRC-wide KM program are also described and documented. While a number of review articles and documents on international LMR operating experience are included in the LMR section of the NRC KC and some references are made to international LMR experience in this report, the primary focus of this document is on U.S. LMR programs and activities.

Key accomplishments of these LMR KM activities include:

- developing an LMR taxonomy for categorizing and organizing LMR technical information entered in the NRC KC
- identifying, categorizing, and uploading some 125 full documents and technical reports to the NRC KC
- preparing an LMR "desk reference guide" on LMR design information, safety issues, operating experience, and LMR subject matter experts and organizations with LMR experience and expertise
- identifying three LMR experts and coordinating the development of three white papers and three corresponding presentations by these experts as part of the NRC RES seminar series (these presentations were video recorded as part of NRC RES archived information)
- developing an SFR technology course structured into nine modules (complete with module objectives, discussion questions, and annotated slides) that is available to be presented to NRC staff when it is deemed appropriate to do so

This document also presents relevant historical information on the various research and development (R&D) programs and their accomplishments for LMRs, starting with the U.S. Atomic Energy Commission (AEC) program that focused on commercial demonstration and development of an LMR from 1950–1989. Included is information on Experimental Breeder Reactor (EBR)-I, EBR-II, Fast Flux Test Facility (FFTF), Clinch River Breeder Reactor (CRBR), the advanced liquid metal reactor (ALMR) program, sodium advanced fast reactor (SAFR) design, and Power Reactor Innovative Small Module (PRISM) design. Information on Toshiba's 4S (Super Safe, Small, and Simple) design is also presented.

Much of the information compiled and collected for LMRs has been added to the NRC Knowledge Center, which is one of the NRC's key information technology applications for capturing and sharing knowledge.

To complement the "backward look" from a KM perspective, Chapter 8 represents a snapshot of current activities associated with LMRs underway at the NRC, U.S. Department of Energy, International Atomic Energy Agency, and standards organizations.

Thus, this report not only summarizes NRC LMR KM activities and points the reader to other resources with relevant LMR information on designs, operating experience, safety considerations, and licensing of LMRs, but should also be viewed as an LMR KM resource tool itself.

10. REFERENCES

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APPENDIX A LIST OF DOCUMENTS IN THE NRC KNOWLEDGE CENTER

Table A.1 List of Documents Entered into the U.S. Nuclear Regulatory Commission (NRC) Knowledge Center

Title of document	Category
Fast Reactor Meltdown Program	Accident Analysis
Equation of State for Fast Reactor Safety Studies: Part 1—Theoretical Relations	Accident Analysis
Fuel-Melting Incident at the Fermi Reactor on Oct. 5, 1966	Accident Analysis
Phenomenological Research in LMFBR Accident Analysis	Accident Analysis
Partial Blockages in LMFBR Fuel Assemblies	Accident Analysis
The Role of Core-Disruptive Accidents in Design and Licensing of LMFBRs	Accident Analysis
Effect of Engineered Safety Features on the Risk of Hypothetical LMFBR Accidents	Accident Analysis
Clinch River Breeder Reactor Plant Safety Study	Accident Analysis
The Mechanistic Analysis of LMFBR Accident Energetics	Accident Analysis
Simulations of Loss-of-Flow Accidents in an LMFBR with the Sodium Loop Safety Facility	Accident Analysis
In-Pile Experiments and Analysis of the Coolability of Uo_2 Debris in Sodium	Accident Analysis
Loss-of-Flow-Without-Scram Tests in Experimental Breeder Reactor-II and Comparison with Pretest Predictions	Accident Analysis
Methodology for Estimating Sodium Aerosol Concentrations During Breeder Reactor Fires	Accident Analysis
Assessment of Prism Response to Loss of Flow Events	Accident Analysis
Metal Fire Implications for Advanced Reactor	Accident Analysis
Thermodynamic Consequences of Sodium Spray Fires in Closed Containments, Part I–Experiments	Accident Analysis
Thermodynamic Consequences of Sodium Spray Fires in Closed Containments, Part II–Calculations with Pulsar	Accident Analysis
Sodium Aerosol Behavior in Liquid-Metal Fast Breeder Reactor Containments	Accident Analysis
Test and Code Development for Evaluation of Sodium Fire Accidents in the FBRs	Accident Analysis
Characteristics of the Aerosol Produced from Burning Sodium and Plutonium	Accident Analysis
Cladding Inner Surface Wastage for Mixed-Oxide LMR Fuel Pins	Cladding
Fission-Product Release and Transport in Liquid-Metal-Cooled Fast Breeder Reactors	Consequence Analysis
Aerodynamic Effects of the EBR-II Reactor Complex on Effluent Concentration	Consequence Analysis
Contamination Control of Sodium Releases from Liquid-Metal-Cooled Fast Breeder Reactors	Consequence Analysis
Consequences of an Accidental Release of Sodium to the Environment from an LMFBR	Consequence Analysis

Table A.1 List of Documents Entered into the NRC Knowledge Center (continued)

Title of document	Category
Estimated Doses and Risks Resulting from Routine Radionuclide Releases from Fast Breeder Reactor Fuel Cycle Facilities: A Summary	Consequence Analysis
LMFBR Source Term Experiments in the Fuel Aerosol Simulant Test (Fast) Facility	Consequence Analysis
Application of Aerosol Technology in LMFBR Design	Consequence Analysis
Radiological Source Terms or LMFR CDAs: A State-Of-The-Art Review	Consequence Analysis
S-Prism Fuel Cycle Study	Consequence Analysis
IAEA–IWGFR Meeting on Sodium Combustion and Its Extinguishment	Coolant
Actinide Recycle Potential in the Integral Fast Reactor (IFR) Fuel Cycle	Fuel (Metal)
Radionuclide Behavior During Normal Operation of Liquid-Metal-Cooled Fast Breeder Reactors, Part 1: Production	Fuel (Metal)
Radionuclide Behavior During Normal Operation of Liquid-Metal-Cooled Fast Breeder Reactors, Part 2: Transport	Fuel (Metal)
Pin-to-Pin Failure Propagation in Liquid-Metal-Cooled Fast Breeder Reactor Fuel Subassemblies	Fuel (Metal)
A Decade of Progress in Fast Reactor Fuel	Fuel (Oxide)
Phenix—17 Years of Fast Reactor Research and Irradiation	Fuel (Oxide)
Achievement of LMFBR Fuel Technology	Fuel (Oxide)
Performance of Fast Flux Test Facility Driver and Prototype Driver Fuels	Fuel (Oxide)
Fuel Pin Mechanical Behavior: Ten Years of LMR Experience	Fuel (Oxide)
Criticality Experiments with Fast Flux Test Facility Fuel Pins	Fuel (Oxide)
A Decade of RBCB Testing of LMR Oxide Fuel in EBR-II	Fuel (Oxide)
U.S. Studies on LMFBR Fuel Behavior Under Accident Conditions	Fuel (Oxide)
Fragmentation Modeling Relative to the Breakup of Molten UO_2 in Sodium	Fuel (Oxide)
Sodium Reactor Experiment Incident	Operating Experience
SRE Operating Experience	Operating Experience
EBR-I Operating Experience	Operating Experience
Hallam Nuclear Power Facility Operating Experience	Operating Experience
Three Years of Phenix Operation	Operating Experience
Review of Maintainability of Large LMFBR Designs	Operating Experience
Design and Operation of Fast Reactors in the USSR	Operating Experience
International Topical Meeting on LMFBR Safety and Related Design and Operational Aspects	Operating Experience
Operational Safety Experience and Passive Safety Testing at the Fast Flux Test Facility	Operating Experience

Table A.1 List of Documents Entered into the NRC Knowledge Center (continued)

Title of document	Category
Component Failure Rates Applicable to LMFBRs as Derived from LER Data	Operating Experience Accident Analysis
Development of LMR Reactor Physics Technology in the United States	Reactor/Plant-Design/Analysis
Advanced Liquid Metal Reactor Development at Argonne National Laboratory During the 1980s	Reactor/Plant-Design/Analysis
Subcriticality Measurement in an LMFBR	Reactor/Plant-Design/Analysis
Nuclear Safety Design of the Clinch River Breeder Reactor Plant	Reactor/Plant-Design/Analysis
Safety Instrumentation for the Sodium-Cooled Fast Reactor	Reactor/Plant-Design/Analysis
Equipment Cell Liners for Liquid-Metal-Cooled Fast Breeder Reactors	Reactor/Plant-Design/Analysis
In-Service Inspection Techniques for Liquid-Metal-Cooled Fast Breeder Reactors	Reactor/Plant-Design/Analysis
Optimization of the Man-Machine Interface for LMFBRs	Reactor/Plant-Design/Analysis
Safety Design for the Advanced Liquid-Metal-Cooled Reactor	Reactor/Plant-Design/Analysis
Advanced Burner Test Reactor Preconceptual Design Report	Reactor/Plant-Design/Analysis
The NACOM Code for Analysis of Postulated Sodium Spray Fires in LMFBRs	Reactor/Plant-Design/Analysis
A Review of Fast Reactor Program in Japan (April 2006–March 2007)	Reactor/Plant-Design/Analysis (?)
LMFBR Safety, I. Fission-Product Behavior in Sodium	Reactor/Plant-Design/Analysis (Design Basis)
Giant Electromagnetic Pump for Sodium Cooled Reactor Applications	Reactor/Plant-Design/Analysis (Design Basis)
Phase II Final Report of Feasibility Study on Commercialized Fast Reactor Cycle Systems	Reactor/Plant-Design/Analysis (Design Basis)
Risk-Oriented Analysis on the German Prototype Fast Breeder Reactor SNR-300	Reactor/Plant-Design/Analysis Accident Analysis
Application of Sodium-Concrete Reaction Data on Breeder-Reactor Safety Analysis	Reactor/Plant-Design/Analysis Accident Analysis
Upper Internals Structures and Their Effect on HCDA Energy Mitigation in Large LMFBRs	Reactor/Plant-Design/Analysis Accident Analysis
Seismic Design Technology for Breeder Reactor Structures: A Review	Reactor/Plant-Design/Analysis Accident Analysis
Safety Assessment of Severe Accidents in Fast Breeder Reactors	Reactor/Plant-Design/Analysis Accident Analysis
Special Topics Relative to the Development of an Optimized Inherently Safe Liquid-Metal Reactor	Reactor/Plant-Design/Analysis Accident Analysis
U.S. ALMR Licensing Status	Safety/Regulatory Framework
Lessons Learned from the Licensing Process for the Clinch River Breeder Reactor Plant	Safety/Regulatory Framework
A Perspective on Progress in Liquid Metal Reactor Safety	Safety/Regulatory Framework

Table A.1 List of Documents Entered into the NRC Knowledge Center (continued)

Title of document	Category
Safety Aspects of the U.S. Advanced Liquid-Metal-Cooled Reactor Program	Safety/Regulatory Framework
The Liquid-Metal Fast Breeder Reactor Safety Program	Safety/Regulatory Framework
Safety Assessment of Fast Sodium-Cooled Reactors in the United Kingdom	Safety/Regulatory Framework
An Appreciation of Fast Reactor Safety—1970, A UKAEA Report Compiled by the Safeguards Division of the Authority's Health and Safety Branch	Safety/Regulatory Framework
Safety Research Programs in the United States for Specific Nuclear Reactor Types	Safety/Regulatory Framework
Safety-Assessment Philosophy of the Fast Flux Test Facility	Safety/Regulatory Framework
1974 ANS Topical Meeting on Fast Reactor Safety	Safety/Regulatory Framework
Status of Research on Key LMR Safety Issues	Safety/Regulatory Framework
The Safety Basis of the Integral Fast Reactor Program	Safety/Regulatory Framework
Report on the Specialists' Meeting on Passive and Active Safety Features of Liquid-Metal Fast Breeder Reactors Organized by the International Atomic Energy Agency at Oarai Engineering Centre of Power Reactor and Nuclear Development Corporation, Japan, November 5–7, 1991	Safety/Regulatory Framework
Key Aspects in Conducting Safety Analysis and Addressing Safety Issues Associated with FFTF and CRBR (White Paper)	Safety/Regulatory Framework
Key Aspects in Conducting Safety Analysis and Addressing Safety Issues Associated with FFTF and CRBR (Presentation)	Safety/Regulatory Framework
A Perspective on Progress in Liquid Metal Reactor Safety	Safety/Regulatory Framework
Overview of the CRBRP Safety Study	Safety/Regulatory Framework
RDT Standard—Reactor Vessel for Liquid Metal Service (RDT E 2-3T), December 1973	Safety/Regulatory Framework
RDT Standard—Class 2 Valves for Liquid Metal Service (RDT E 1-19T), June 1974	Safety/Regulatory Framework
RDT Standard—Forced-Circulation Cold Trap Assembly for Removal of Sodium Impurities (RDT E 4-5T), January 1976	Safety/Regulatory Framework
RDT Standard—Instrument Tree for Sodium Cooled Reactors (Fabrication Only), (RDT E 6-18T), February 1973	Safety/Regulatory Framework
RDT Standard—Radial Reflector Assembly for Liquid Metal Fast Breeder Reactors (RDT E 6-19T) March 1978	Safety/Regulatory Framework
RDT Standard—Austenitic Stainless Steel Hexagonal Duct Tubes for Core Components and Assemblies (RDT E 6-20T), May 1976	Safety/Regulatory Framework
RDT Standard—Control Rod Absorber Pin for Liquid Metal Fast Reactors, (RDT E 6-25T), September 1976	Safety/Regulatory Framework

Table A.1 List of Documents Entered into the NRC Knowledge Center (continued)

Title of document	Category
RDT Standard—Class 2 Nuclear Components (Supplement to ASME Boiler and Pressure Vessel Code, Section III, Subsections NCA and NC), (RDT E 15-2NC-T), June 1978	Safety/Regulatory Framework
ANS Standard—General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant (ANSI/ANS-54.1-1989) May 1989	Safety/Regulatory Framework
Industry Perspectives and Experience in the Design of Liquid-Metal-Cooled Reactors (Presentation)	Safety/Regulatory Framework
Industry Perspectives and Experience in the Design of Liquid-Metal-Cooled Fast Reactors (White Paper)	Safety/Regulatory Framework
Feedback from Practical Experience with Large Sodium Fire Accidents	Safety/Regulatory Framework
Review of Fast Reactor Operational Experience Gained in the Russian Federation; Approaches to the Coordinated Research Project	Safety/Regulatory Framework
The Sodium Loop Safety Facility	Small Test Reactor/Experimental Facilities
The Roles of EBR-II and TREAT in Establishing Liquid Metal Reactor Safety	Small Test Reactors/Experimental Facilities
An Overview of Fast Flux Test Facility Contributions to Liquid Metal Reactor Safety	Small Test Reactors/Experimental Facilities
Reactor Physics Results from Fast Flux Test Facility Operation	Small Test Reactors/Experimental Facilities
FFTF Operational Results: Startup to 100 MWd/kg	Small Test Reactors/Experimental Facilities
Experimental Breeder Reactor-II: 20 Years of Operational Experience	Small Test Reactors/Experimental Facilities
EBR-II Test and Operating Experience	Small Test Reactors/Experimental Reactors
EBR-II Contributions to Sodium-Cooled Reactor Technology	Small Test Reactors/Experimental Reactors
Application of the Gem Shutdown Device to the FFTF Reactor	Small Test Reactors/Experimental Reactors
Safety Analysis of FFTF Loss of Flow Without Scram Tests	Small Test Reactors/Experimental Reactors
Results of the 1986 FFTF Inherent Safety Tests	Small Test Reactors/Experimental Reactors
Posttest Analysis of the FFTF Inherent Safety Tests	Small Test Reactors/Experimental Reactors
Comparison of the SASSYS/SASA4A Radial Core Expansion Reactivity Feedback Model and the Empirical Correlation for the FFTF	Small Test Reactors/Experimental Reactors
Fast Flux Test Facility Passive Safety Reactivity Feedback Measurements	Small Test Reactors/Experimental Reactors
Fast Flux Test Facility (FFTF) Feedback Reactivity Components	Small Test Reactors/Experimental Reactors
Fast Flux Test Facility Passive Safety Flow Transient Test	Small Test Reactors/Experimental Reactors
Comparison of Reactivity Feedback Models for the FFTF Passive Safety Tests	Small Test Reactors/Experimental Reactors
Design and Evaluation of FFTF Containment	Structural Analysis
Fast Flux Test Facility Core Systems	Structural Materials
Design Options for the Core Support Structures for Liquid-Metal-Cooled Reactor Plants	Structural Materials

APPENDIX B LIST OF EXPERIMENTAL LIQUID-METAL-COOLED REACTORS

Table B.1 Liquid-Metal-Cooled Reactor Technology Experimental Reactors and Testing Facilities*

Facility name	Facility type	Other names	Location	Country	Capabilities	Status	Years operated
BOR-60	Test reactor		Dimitrovgrad	Russian Federation		Active	1968–present
Experimental Breeder Reactor-II	Test reactor	EBR-II	Idaho National Laboratory, Idaho Falls, ID	United States		Shutdown	1961–1998
Fast Critical Assembly	Test reactor	FCA		Japan	Large split-table critical facility capable of testing mockups of full-size reactor cores	Active	
Fast Flux Test Facility	Test reactor	FFTF	Richland, WA	United States		Shutdown	1980–1996
High Temperature Liquid Sodium Test Loop	Out-of-pile	HTLSTL	Yokohama	Japan	High temperature liquid-sodium test loop; maximum temperature 600°C; flow rate currently 1 m^3/min with plans to expand flow rate to 10 m^3/min	Active	
Hot Fuel Examination Facilities	Out-of-pile		Oak Ridge National Laboratory, Oak Ridge, TN, and Idaho National Laboratory, Idaho Falls, ID	United States	Hot cell capable of handling irradiated fuel for post-irradiation examination (PIE); a wide range of instrumentation is available	Operable at ORNL and INL	

Table B.1 Liquid-Metal-Cooled Reactor Technology Experimental Reactors and Technology Facilities (continued)

Facility name	Facility type	Other names	Location	Country	Capabilities	Status	Years operated
Jōyō	Test reactor		Ibaraki	Japan		Active	2003–present
Liquid Metal Engineering Center	Out-of-pile	Santa Susanna Test Facility; Energy Technology Engineering Center	Canoga Park, CA	United States	Capable of testing large components at flow rates of 380 m^3/min (100,000 gpm) (80-MW heaters); Large pumps, both EM and mechanical (Sodium Pump Test Facility (SPTF)); Steam generators/heat exchangers (Sodium Components Test Lab (SCTL))	Closed and decommissioned	1966–1998
Phénix	Test reactor			France	Irradiation test facility	Decommissioned	1973–2004
Southwest Experimental Fast Oxide Reactor	Test reactor	SEFOR	Fayetteville, AR	United States	20-MWt reactor for testing oxide LMR fuel	Decommissioned	1969–1972
Thermal Hydraulic Out-of-Reactor Safety Facility	Out-of-pile	THORS	Oak Ridge National Laboratory, Oak Ridge, TN	United States	Sodium loop containing electrically heated LMR fuel pins and bundles and capable of testing up to three 61-pin bundles. THORS was capable of static, flow, and transient tests, leading to large-scale boiling of sodium coolant	Decommissioned	1972–1981

Table B.1 Liquid-Metal-Cooled Reactor Technology Experimental Reactors and Technology Facilities (continued)

Facility name	Facility type	Other names	Location	Country	Capabilities	Status	Years operated
Transient Reactor Test Facility	Test reactor	TREAT	Idaho National Laboratory, Idaho Falls, ID	United States	A reactor capable of providing transient power capabilities to specimens (fuels). Power levels are capable of reaching melting points of most fuels. TREAT has high-speed instrumentation capabilities to monitor the experiment during the transient.	Cold shutdown	
Whole Pin Furnace Test Facility	Out-of-pile	WPF		Japan	LMR fuel pin testing with exterior furnace heating in a hot cell	Active	
Zero Power Physics Reactor	Test reactor	ZPPR	Idaho National Laboratory, Idaho Falls, ID	United States	Large split-table critical facility capable of testing mockups of full-size reactor cores up to 14 ft. in diameter	Shutdown	

* Note: For information on additional LMR experimental reactors and facilities, please contact Imtiaz Madni of NRC at Imtiaz.Madni@nrc.gov regarding information compiled by the OECD-NEA's Task Group on Advanced Reactor Experimental Facilities (TAREF). An extensive list of experimental facilities in the United States and internationally is provided in the TAREF report for SFRs [Ref. 13].

APPENDIX C WHITE PAPERS

APPENDIX C.1

KEY ASPECTS IN CONDUCTING SAFETY ANALYSIS AND ADDRESSING SAFETY ISSUES ASSOCIATED WITH FAST FLUX TEST FACILITY AND CLINCH RIVER BREEDER REACTOR

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Prepared for the U.S. Nuclear Regulatory Commission
Under Arrangement with the Oak Ridge National Laboratory

Comments in this paper will be addressed to both the Fast Flux Test Facility (FFTF), located at the Hanford Reservation in Southeastern Washington State, and the Clinch River Breeder Reactor Project (CRBR), once envisioned for construction on the banks of the Clinch River near Oak Ridge, Tennessee. However, the bulk of the focus will be directed to the FFTF, given the author's more intimate knowledge of the safety issues addressed during the regulatory process for that facility.

For the FFTF, we shall first address the regulatory history and then deal with design-basis accidents (DBAs) and beyond-design-basis accidents (BDBA). We then shift to the key safety questions that remained open following the preliminary safety analysis report (PSAR) phase and then discuss how these safety issues were appropriately addressed during the final safety analysis report (FSAR) phase. Finally, we deal with the major safety test programs conducted in the FFTF after operations began, along with a brief summary of the key lessons learned during the overall regulatory review and subsequent operations phases.

The CRBR safety experience was necessarily confined to preliminary safety studies and regulatory review, because the project was stopped before substantial construction was begun. Some lessons learned during the licensing process will be briefly reviewed.

FFTF: The Fast Flux Test Facility at Hanford, Washington

I. Regulatory Review History

The FFTF was conceived in the 1960s as a 400-MWt sodium-cooled fast spectrum test reactor, designed to test fuels and materials that would be needed for the expected rapid use of fast breeder reactors for commercial power generation. The FFTF, itself, was not a breeder reactor. Rather, it was configured with radial and axial reflectors to enhance the neutron flux needed for rapid testing of new fuels and materials. The internal core conversion factor was about 0.6, far below the value needed for actual breeding. Because the primary mission of the facility was to provide a prototypic environment for materials testing (complete with extensive internal instrumentation and special test loops), the system did not contain any capacity for the generation of electricity. Rather, heat from the secondary sodium system was transferred to air-dump heat exchangers. Figure C.1 shows the overall heat transfer system for this “loop-type” reactor, and Figure C.2 is a cross section of the primary vessel and the reactor itself. Table C.1 provides FFTF design parameters.

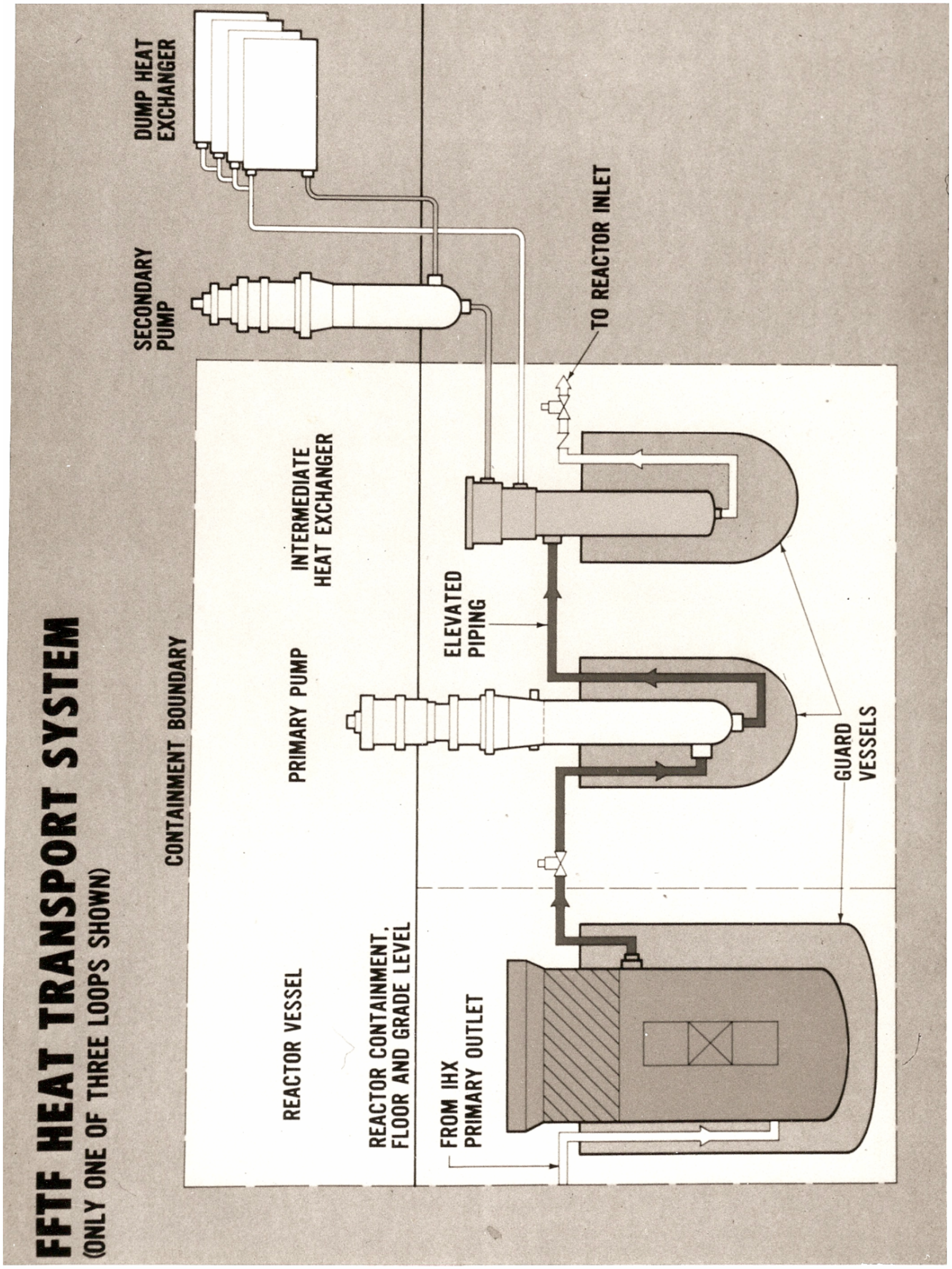


Figure C.1 Heat transfer system for the FFTF

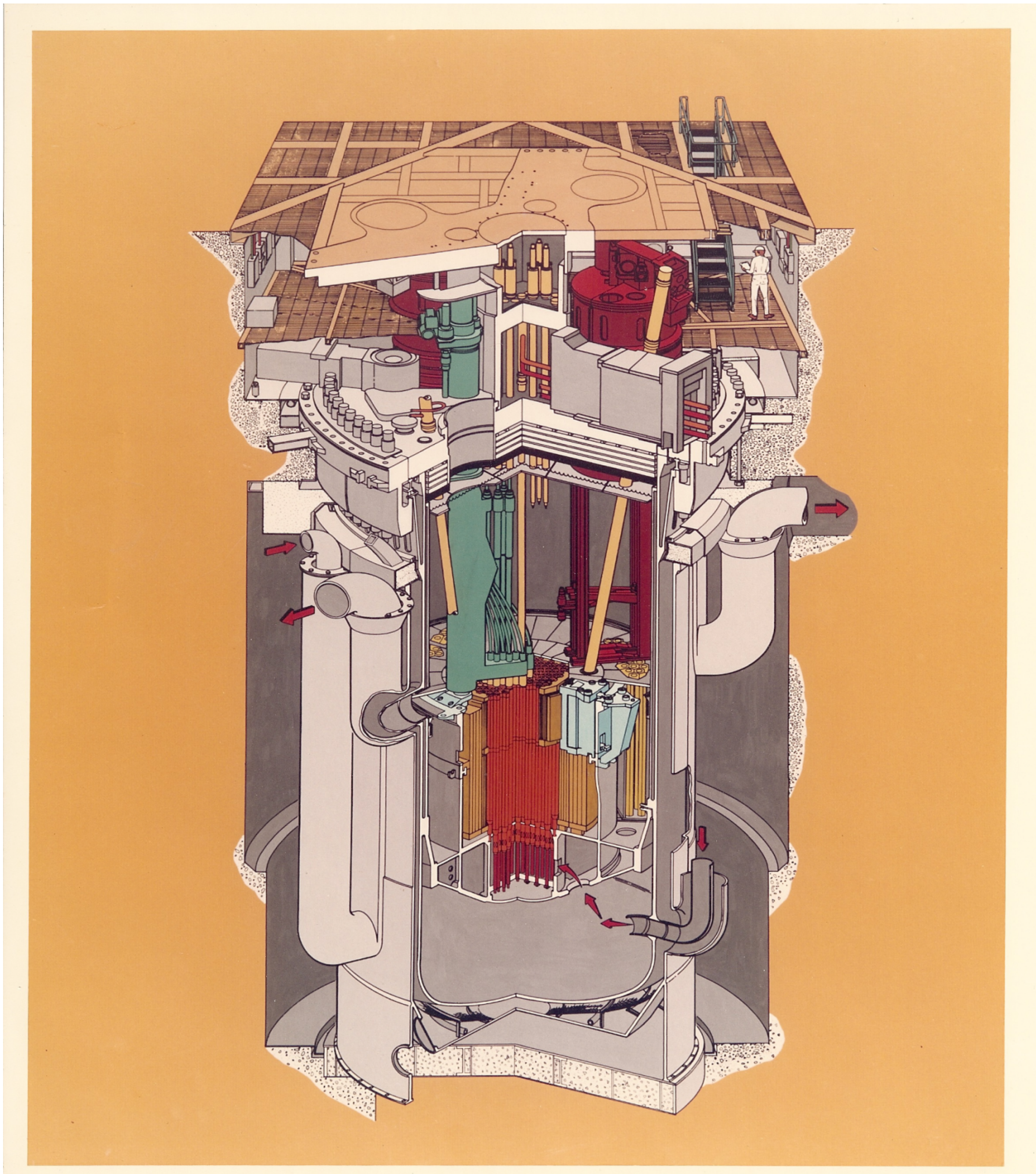


Figure C.2 Cross section of the FFTF reactor vessel

Table C.1 FFTF Design Parameters

FFTF DESIGN PARAMETERS

REACTOR CONTAINMENT VESSEL	
Material	Carbon steel
Diameter	135 ft 0 in.
Overall Height	186 ft 8 in.
Depth Below Operating Floor	77 ft 10-1/2 in.
Wall Thickness Above Grade	1 to 1-3/8 in.
Internal Design Pressure	10 psig
REACTOR VESSEL	
Material	304 stainless steel
Internal Diameter	243 in. (about 20 ft)
Height	517 in. (about 43 ft)
Wall Thickness	2-3/8 to 2-3/4 in.
Vessel Liner Internal Diameter	19 ft 10-1/2 in.
REACTOR GUARD VESSEL	
Material	304 stainless steel
Wall Thickness	1 to 2-1/2 in.
Annulus Between Reactor Vessel and Reactor Guard Vessel	8-in.
REACTOR VESSEL HEAD	
Material	Low carbon alloy steel
Diameter	25 ft
Thickness	about 22 in. (plus shielding plates)
Weight	214 tons
GENERAL	
Power, Excluding Closed Loops	400 MW
Closed Loops Capability	4 at 2.3 MW each
NUCLEAR CONTROLS	
Type of Absorber	B_4C Pellets
Boron-10 per Assembly	1.16 to 1.28 kg
Pins per Assembly	61
Pin Cladding - OD	0.474 in.
- Thickness	0.051 in.
Pellet Diameter	0.362 in.
Duct - Wall Thickness	0.120 in.
- Length	12 ft
- Material	316 SS 20% CW
Cladding Material	316 SS 20% CW
Wire Wrap Material	316 SS 17% CW
HEAT REMOVAL - MAIN HEAT TRANSPORT SYSTEM	
Circuits	3
Rating per Circuit	133 MW
DHX Modules	12
Rating per DHX Module	33 MW
Sodium Flow Rate	43,500 gal/min
Reactor Vessel - Inlet Temperature	680°F nominal
- Outlet Temperature	980°F nominal
- Inlet Pressure	133 psig nominal

Table C.1 FFTF Design Parameters (continued)

FFTF DESIGN PARAMETERS

CORE CONFIGURATION		
Core Positions	- Rows 1-6 (Fueled Zone)	91
	- Rows 7-9 (Reflector Zone)	108
	- Total Positions	199
Core Assembly	- Shape	Hexagonal
	- Dimension Across Load Pads	4.715 in.
	- Length (for Most)	12 ft
Active Core	- Fueled Height	36 in.
	- Equivalent Diameter	47.2 in.
	- Volume	1034 liters
Positions for Driver Fuel Assemblies		74
Primary Control Rods (Boron Carbide)		3
Secondary Control Rods (Boron Carbide)		6
Fixed Peripheral Absorber Assemblies		0-15
Positions for 40-ft Independently Instrumented Test Assemblies		8
CORE PHYSICS		
Total Fissile Mass		563 kg
Neutron Flux (Peak)		7×10^{15} n/cm^2-s
0.1 MeV		60 to 65%
Conversion Ratio		0.43
Fuel Cycle, Nominal		100 Full Power Days
Average Discharge Burnup		45 MWd/kg
Limiting Peak Burnup		80 MWd/kg
Power Density		0.39 MW/liter
Doppler Coefficient		-0.005
Delayed Neutron Fraction		0.003
DRIVER FUEL		
Fuel Assembly Power		3 to 7 MW
Assembly Width at Load Pads		4.715 in.
Assembly Length		12 ft
Fuel Pin Linear Average Power		7.3 kW/ft
Driver Fuel Assembly Average Coolant Flow		18.7×10^4 lb/hr
Initial Core Average Coolant Velocity		21 ft/s
Fuel Type		PuO_2-UO_2
Pu % (of Pu and U) - Rows 1-4		about 22.4%
Pu % (of Pu and U) - Rows 5-6		about 27.4%
Pu Fissile Content ($^{239}Pu + ^{241}Pu$)		88 wt%
Uranium Type		Natural
Duct Wall Thickness		0.120 in.
Duct Material		316 SS 20% CW
Fuel Pins per Fuel Assembly		217
Fuel Pin Cladding	- OD	0.23 in.
	- Thickness	0.015 in.
	- Material	316 SS 20% CW
Fuel Pin Wire Wrap	- Diameter	0.056 in.
	- Material	316 SS 17% CW
Fuel Pellets per Pin		about 150
Fuel Pin	- Length	about 94 in.
	- Fueled Length	36 in.

The PSAR was developed in the late 1960s, then under the auspices of the Atomic Energy Commission (with Battelle Northwest serving as the principal federal contractor). Westinghouse Hanford Company acquired the federal contract for this project on July 1, 1970, and the PSAR was submitted in September 1970. The original Atomic Energy Commission (AEC) later became the Energy Research and Development Administration (ERDA) when the decision was made to split off the regulatory arm of the AEC to become the U.S. Nuclear Regulatory Commission (NRC). ERDA later became the U.S. Department of Energy (DOE), and the

Westinghouse Hanford Company was awarded the contract for management and operation of the Hanford Engineering Development Laboratory.

Whereas there was no requirement for the NRC (and its predecessor within the AEC) to formally license an AEC (ERDA/DOE) facility, it was the policy at the time of initiating the FFTF to subject the project to a full-scale regulatory review. This was done for two reasons: (1) to make sure that this facility would meet the strictest, independent regulatory review, and (2) to bring the NRC up to speed for licensing sodium-cooled fast reactors because the expectation at that time was that many fast breeder reactors would be needed to meet increasing national needs for electricity and maintain a long-term supply of nuclear fuel. We now know that uranium supplies are substantially more plentiful (therefore delaying the need for breeder reactors for a few more decades), but that knowledge was not available in the 1960 time frame.

Although FFTF, which used sodium as a coolant, posed a new challenge to prevailing regulatory procedures, the regulatory approach adopted for FFTF used existing NRC guidelines as closely as possible. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," was used when it was issued. The Regulatory Guides were used for seismic testing and analysis of structures, systems, and components (SSCs). The Regulatory Guides for nuclear-safety-related SSCs were also used to develop functional criteria such as separation, redundancy, performance goals, etc.

The fundamental safety approach for designing and evaluating the performance of the FFTF was based on Lines of Assurance (LOAs). This approach recognizes that any accident sequence could (at least theoretically) progress though either natural or designed barriers. It provides a balance between the probability of a particular consequence and the severity of that consequence. Whereas a robust safety system was designed to stop any accident sequence from progressing to a point of core damage, the Lines of Assurance philosophy recognized that such barriers might fail—and the logical approach is then to assess the associated consequences of such failure. Thus, all conceivable accident sequences were followed in a mechanistic manner to provide the required answers.

The four levels of the Lines of Assurance were as follows:

LOA I—Prevent Accidents (Build sufficient robustness into the basic design to minimize the initiation of any kind of accident. Design Safety Criteria constituted a major part of every system in conformance with 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities");

LOA II—Limit Core Damage (Establish failure limits in the fuel and coolant system and be assured with a high degree of confidence that the safety systems designed into the plant would arrest any accident sequence in such a way that plant operation could continue);

LOA III—Maintain Containment Integrity (Minimize the probability that any serious accident—often referred to as a Beyond-Design-Basis Accident—would progress to the point of containment breach); and

LOA IV—Attenuate Radiological Consequences (Minimize the radiological consequences of any remote accident sequence that might penetrate containment).

The review of the PSAR leading to construction authorization took 31 months (Ref. 1), including 23 substantive meetings with the NRC and the Advisory Committee on Reactor Safety (ACRS). As noted above, the PSAR was submitted in September 1970 and an Interim Construction Permit was authorized in February 1972. Full construction was authorized in May 1973, after the receipt of an ACRS letter expressing confidence for taking this major next step.

As a prelude to approaching the next section of this paper, we note that Design-Basis Accidents (DBAs) are essentially those accidents considered to test Line of Assurance Level II; namely, consider credible accident initiating mechanisms, minimize the frequency of such off-normal events, and then ensure that adequate safety margins exist—all with the objective of verifying that the reactor design is fundamentally safe. Beyond-Design-Basis Accidents (BDBAs) are hypothesized to test LOA Levels III and IV; namely, characterize containment margins and then calculate the possible radioactive releases should the containment barrier be compromised.

It might be noted that considerable attention was paid to the BDBAs in the regulatory process for FFTF, probably relatively more emphasis than justified. The main reasons for such emphasis was likely twofold: (1) The FFTF regulatory review was deemed a test case for licensing potentially large sodium-cooled reactors (and, therefore, given substantial scrutiny to determine end-of-spectrum consequences of such reactors), and (2) the types of analytical and experimental programs necessitated by such exploration are academically stimulating, thereby inherently attracting a good deal of attention from professionals who enjoy large challenges. However, it should be clearly noted that despite the relatively large expenditure of resources dedicated to BDBAs in the FFTF regulatory review, real safety comes from design measures built into the plant…and then appropriately tested for plant robustness in the first two Lines of Assurance.

II. Design-Basis Accidents (DBAs)

The two generic accident categories of accidents in an operating nuclear reactor are reactivity insertion events and loss-of-cooling events. In either case, the concern is overheating the core—leading to core damage if not controlled by inherent features of the plant or by appropriate engineered safeguards. All nuclear reactors are equipped with plant protection systems to arrest either type of generic accident.

In addition to these two classes of accidents, the FFTF was evaluated to assess the following:

- natural circulation cooling
- assurance of piping integrity
- emergency power
- seismic design
- core thermal design (including hot channel factors)
- instrument and control design
- quality assurance
- radiation protection
- waste management
- sodium spills
- fuel handling
- external events (including fire, flood, tornado, and earthquake protection)

As we will point out later, the two items from the above listing that took the longest to resolve were the adequacy of natural circulation cooling and the assurance of piping integrity.

With regard to reactivity insertion events, a number of events were postulated without regard to their credibility in order to envelop all such initiating events. Such items included the continuous withdrawal of a control rod, the meltdown of a single control rod, the loss of hydraulic hold-down of the fuel assemblies, the movement of the radial core restraint system, and a cold sodium insertion—even though design features essentially negate their possibility.

There are six control rods in the FFTF that can be individually moved for reactivity control (all enriched in boron carbide), along with three safety rods that are always fully inserted during shutdown but are fully withdrawn before reactor operation. The maximum worth of any single control rod is about 4 dollars. In assessing the potential consequences of uncontrolled control rod movement, reactivity ramp rates from about 5 cents/sec up to 50 cents/sec were evaluated for reactor system response.

The instrumentation in this reactor is configured in a 2-out-of-3 logic, meaning that if any two sensors indicate an abnormal rise in coolant temperature (or a myriad of other indications of off-normal conditions), the Plant Protection System (PPS) will spring into action, wherein all the of control rods—as well as the three safety rods (worth a total of about 24 dollars)—are dropped into the reactor to shut down the nuclear reaction. For such a postulated accident, the upset conditions were shown to be far below any failure thresholds.

The meltdown of a single control rod was conservatively evaluated to introduce a ramp rate on the order of 10 cents/sec (for a total reactivity insertion of a few dollars). Again, the PPS was shown to more than adequately terminate the excursion with no core damage.

Whereas the fuel elements in a reactor such as FFTF are very heavy, the upward flow of sodium coolant does introduce a drag force. Hence, the lower core structure was designed to allow a small bypass flow to enter a low-pressure plenum to offset the upward hydraulic force on each fuel element. The potential loss of this hydraulic hold-down was postulated as another test of the PPS and, as predicted, the PPS was more than adequate to ameliorate any damage to the core during such an event. It is worth noting that the inlet channels into the fuel assemblies were carefully designed to negate any inlet flow blockage of the type that occurred in the Fermi 1 reactor.

The core is held together at the base by a mechanical fuel socket arrangement in the lower core support structure. But, if left unrestrained, the fuel elements could “flower out” in and above the active core region. Hence, a radial core restraint system was designed to keep the core tight in a radial direction. Special duct pads were included on all fuel element assemblies to take the radial load induced by the radial core restraint system. Here again, it was assumed that for some reason this radial restraint system would fail in a manner to allow the core to move outward. Whereas this would normally result in a negative reactivity, several possibilities were considered that might introduce a small positive reactivity. Again, the PPS was shown to deal with any such possibilities.

Finally, it was postulated that overcooling might occur in the secondary system and cold sodium would be introduced into the core. Because the overall sodium void coefficient in the FFTF is negative (though positive in the central core region), and the Doppler coefficient is strongly negative, such a situation would result in core cooling—thereby introducing a positive reactivity insertion. The maximum reactivity insertion under such a condition was determined to be less than the conditions analyzed above and the PPS was determined to adequately handle such a situation.

It should be noted that a classic question for any reactor system is how long operations can be safely continued in accommodating random fuel pin failures. There was an early concern for fast reactors that one pin failure might release fission gas at the failure site, thus temporarily starving coolant for surrounding pins and causing pin-to-pin failure propagation. However, a testing program in EBR-II (the “run beyond failure” program) clearly showed that this was not a safety problem for liquid-metal cooled systems (Ref. 2).

For loss-of-flow events, two levels of escalating concerns were evaluated. First, it was assumed that all offsite electrical power was lost. Under such conditions, emergency power required to drive the coolant pumps at low speed (using “pony motors”) would automatically come on, although there would be a slight time delay (a few seconds) for this to happen. The primary coolant pumps were specifically designed to have considerable inertia, such that the drop from full flow to 10 percent flow would take about 50 seconds. The resulting action of the PPS was shown to provide ample protection of the core for this case.

A more severe case would be the loss of all offsite power AND loss of emergency power (supplied by standby diesel generators), resulting in loss of all forced coolant flow. Again, however, PPS action was shown to drop the power level in the core fast enough to prevent any core damage. Natural circulation of the sodium coolant would provide effective cooling of the core. This is discussed further in a later section of this paper.

The loss of flow by any one of the three primary pumps would, of course, provide only a small test of the PPS (given that no core damage would be inflicted by the loss of all power to all three primary pumps).

The loss of functioning by the flow controllers could potentially result in a continuous flow reduction. This potential was evaluated, again with the result that the PPS would recognize the power-to-flow imbalance and respond accordingly.

A pump seizure event was also analyzed. The concern was that pump seizure might result in a more abrupt reduction in flow because the rotating inertia from that pump would become immaterial. Again, however, the PPS responded appropriately.

As another test of the PPS, it was assumed that air flow was restricted to the air-dump heat exchangers (despite redundancies included in the design to prevent such restrictions)—resulting in the loss of the ultimate heat sink. This would, of course, cause the primary coolant temperature to rise. Again, the PPS was shown to adequately respond in such a way that no core damage would be inflicted.

Other potential accident sequences were performed to determine bounds for any conceivable type of reactivity insertion or loss-of-flow event and the analyses performed by both the applicant and the NRC concluded that proper action by the PPS in the FFTF would adequately protect the core under any credible situation.

III. Beyond-Design-Basis Accidents (BDBAs)

Whereas the analyses performed for the Design-Basis Accidents were relatively routine (very important, of course, but relatively easy to perform because no material failures are incurred), this is not the case when one postulates that the PPS completely fails. Early concerns for fast reactors were that if the PPS should become completely inoperable, an accident might proceed all the way to a core meltdown and subsequent disassembly. This potential outcome emanated from the fact that fast reactors are very compact machines wherein more than a single critical mass is contained in the enriched fuel—should this fuel all be compacted into a single clump. Hence, if one were to postulate overpower or loss-of-cooling transients with no protection from the PPS, it is conceivable that a collapsed core could go critical with a very high reactivity insertion rate—with the accident ultimately terminated through core disassembly.

Such was the “state of the art” at the time the Experimental Breeder Reactor-I (EBR-I), Experimental Breeder Reactor-II (EBR-II), Dounreay Fast Reactor (DFR), Rapsodie, and Fermi 1 reactors were built. The model initially used to assess the consequences of a postulated energetic core-disassembly accident was the so-called Bethe-Tait model (Ref. 3), named after the two reactor physicists that developed a simplified disassembly model that provided a closed-form analytic solution. One of the first exposures of this author to this approach was a meeting with Professor Hans Bethe (in his office at Cornell University) along with members of the Power Reactor Development Corporation (owners of the Fermi 1 project for Detroit Edison) who had analyzed the Fermi 1 fast spectrum reactor that was built near Detroit. The amazing and somewhat disturbing result of that encounter was to learn that the analysts of the Fermi 1 reactor safety analysis team had spent so much time conducting Bethe-Tait analysis that they were “mesmerized” into believing that this MUST be the way fast reactors behave under unprotected conditions! Fortunately, Professor Bethe recognized that his earlier work was intended only to be an “order of magnitude” type of analysis, which likely was adequate for providing upper bound results for the early, small reactors—because the energetic release for such reactors could be readily contained with reasonably sized containment structures even for very conservative estimates. However, he recognized that with the advent of more powerful analytical techniques, made possible by larger computer systems, a more mechanistic approach would provide a considerably better basis for evaluating the consequences of unprotected accidents in fast-spectrum systems.

Accordingly, one of the major contributions of the FFTF regulatory review process was to wean the profession away from the ultra-conservative Bethe-Tait model and focus on more mechanistic methods to determine potential consequences of unprotected events. At the time of the initial FFTF studies, however, one complication was postulated that could make matters worse than determined by classic Bethe-Tait analyses; namely, the original Bethe-Tait model assumed the equation-of-state (i.e., the relationship between core temperatures and the pressures building up to cause the disassembly) to be the vapor pressures of the fuel (mixed oxide in the case of FFTF). However, it was noted by Hicks and Menzies (Ref. 4) that the molten fuel would transfer heat energy to the surrounding sodium, and if done instantly, the resulting sodium vaporization could result in even higher levels of energetic release (i.e., more damage to the containment system).

For FFTF, we developed a coupled neutronics, multi-channel thermal-hydraulics code (the MELT family of codes) (Ref. 5) to follow either transient overpower or transient undercooling accidents in order to better assess the core conditions just before a disassembly phase. (Remembering, of course, that any such accident sequences are truly BEYOND the design basis; as such, they have often been referred to as Hypothetical Core Disruptive Accidents.) An Argonne National Laboratory (ANL)-developed hydrodynamic code, VENUS (Ref. 6), was then coupled to the MELT code to determine the energetic release associated with these postulated accidents (Ref. 7). Additional work was done to assess the transfer of heat from the molten and largely vaporized fuel to the surrounding sodium (for cases in which sodium could be credibly argued to be available for such an interaction), and the energy expansion (determined by the code SOCOOL) (Ref. 8) was transferred to the mechanical deformation code ASPRIN (Ref. 9) and later to the more sophisticated code REXCO (Ref. 10) to determine the damage to the reactor vessel.

For the PSAR, this approach (Ref. 11) was used to determine a bounding case for both unprotected transient overpower (UTOP) accidents and unprotected transient undercooling (UTUC) accidents in the FFTF.

For the UTOP, it was arbitrarily assumed that the maximum worth control rod was withdrawn at the maximum rate physically possible, which translated into a reactivity ramp rate of about 50 cents/s. Because there was little experimental data available at that time to determine how fuel pins might fail under such circumstances, it was conservatively assumed that they would fail at the axial midplane—wherein molten fuel might flow within the pins toward the break at the core centerline (resulting in a substantial positive reactivity). Further, it was conservatively assumed that the molten fuel being ejected through the cladding rupture would instantly transfer its heat to the surrounding sodium, causing the sodium to flash into vapor and be ejected from the core (further exasperating the situation caused by the positive sodium void reactivity in the mid-core region). The bottom line under these assumptions resulted in the initiating ramp rate of 50 cents/s being escalated up to around 200 dollars/s at the onset of core disassembly.

Whereas this appeared to be an alarming result (later shown from in-pile test results to be unrealistically conservative), the actual energy release as determined by the VENUS code was relatively small because most of the sodium was still in the core and, thereby, presented a "hard" or "heated, confined liquid" equation-of-state—causing very rapid disassembly with a relatively modest energy release, calculated to be about 150 megawatt seconds (MW-s). This energy release corresponds to approximately 34 kilograms (75 pounds) of TNT (based on work energy conversion fractions determined from the SL-1 accident), although with a pressure response considerably less destructive than a TNT explosion. The vessel was shown to be more than adequate to accommodate such a bounding accident (Ref. 12).

Several other initiating conditions were analyzed, including the possibility of a large sodium bubble passing through the core. In some cases the initiating ramp rate was larger than 50 cents/sec, but given the extreme levels of conservatism included in the assumed transfer of molten fuel energy (at the time of permanent nuclear shutdown) into workable energy, the 150 MW-s work energy was deemed to provide a suitable upper bound for containment response purposes.

The unexpected loss of flow (ULOF) accident was then analyzed with the same code system. Without PPS protection, the coolant was calculated to begin boiling in about 5 seconds, followed shortly by cladding melting and subsequent fuel slumping. Because sodium boiling began near the top of the core, the overall reactivity consequences of reactor voiding provided a negative reactivity to prevent core disassembly (despite encountering some positive reactivity spikes when the central regions of the core were voided). If subsequent core melting was postulated to result in a condition of recriticality, the energy release was determined to be relatively small because of the much lower reactivity ramp rate at the time of criticality—despite the "softer" (fuel vapor) equation-of-state. Hence, the 150 MW-s energy release calculated for the UTOP was judged to bound all of the ULOF accident scenarios. Later analyses, conducted with considerably more sophisticated models (Ref. 13), provided further assurance of this conclusion. Analyses conducted for the unprotected loss of heat sink (ULOHS) accident produced results similar to those for the ULOF.

It should be noted that when the sodium boiling analyses were first conducted, there was some speculation that considerable superheating might occur before boiling—thereby causing relatively instant boiling of the core central regions where the sodium void coefficient was positive. However, several experiments were conducted that demonstrated sufficient nucleation sites in an operating environment would be available to reduce superheating to essentially zero.

For this 150 MW-sec energy release, accepted by the NRC as an appropriate bounding case for containment studies, the resulting vessel strains were considerably below the actual yield strengths. The results are noted in Figure C.3. Mechanical deformation calculations carried out for the primary vessel indicated that vessel failure would not occur below an energetic release of about 350 MW-s (Ref. 14).

During the FSAR phase, several experiments were carried out in the Transient Reactor Test Facility (TREAT) at Idaho Falls using prototypic pins in a near-prototypic environment. Because TREAT is a thermal reactor, the flux spectrum could not be modeled as well as desired, although cadmium shielding was used around the test loop to screen out much of the thermal neutron spectrum to better match the spectrum that would be expected under actual fast reactor accident conditions.

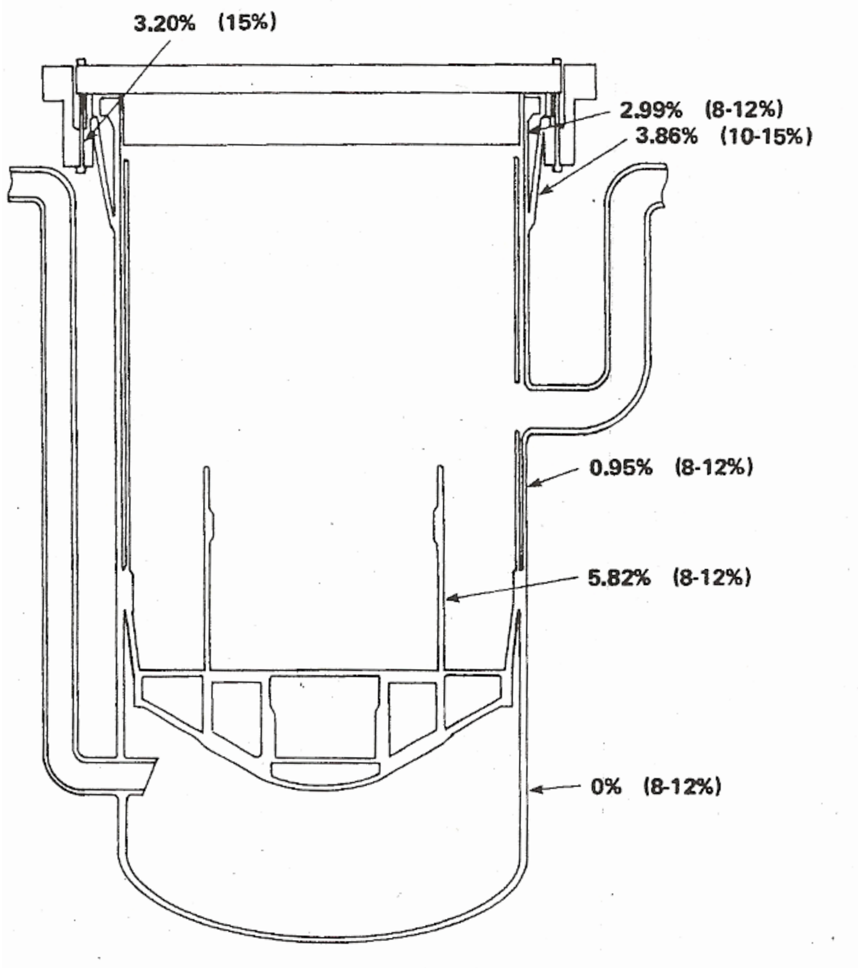


Figure C.3 FFTF vessel strains calculated for the BDBA—compared to allowable strains to failure

The UTOP test series in the TREAT reactor demonstrated that pins would actually fail near the top of the core where the cladding is weakest because of high temperature (Ref. 15). Hence, any molten fuel flowing inside the pins would move in a strongly negative reactivity direction. Further, once the molten fuel entered the coolant channel, it would be literally washed out by the high-velocity sodium coolant (recall that in the UTOP, it is assumed that the pumps are still energized; it is a reactivity excursion wherein the PPS is hypothesized to completely fail). Hence, these experimental results provided strong evidence that the results of a UTOP would be well below the 150 MW-sec bounding basis. Certainly substantial core damage could be done, but the energetic release would be minimal (Ref. 16).

For the ULOF and ULOHS situations, the TREAT experiments showed that sodium boiling and subsequent cladding melting would occur but would very likely not lead to recriticality (Ref. 17). Because of the difficulty in modeling this behavior, the SAS4A code was developed at ANL, which contained models for both sodium boiling and cladding melting and slumping (Ref. 18). Further, a more mechanistic hydrodynamic disassembly code, SIMMER (Ref. 19), was developed at the Los Alamos National Laboratory (LANL) and later analyses for the ULOF and ULOHS accident sequences were conducted with these two code systems. In the meantime, phenomenological arguments were developed (Ref. 20)to provide upper-bound estimates for the energy release during the so-called “Transition Phase,”—the condition of the core after the loss of core geometry but before either fuel or sodium vapor building to the point of causing hydrodynamic disassembly pressures to become effective. The arguments were based on the natural dispersion tendencies (including the release of fission product gases) of a core internally heated by radioactive decay, thus ruling out the possibility of a recriticality.

Once the reactor excursion was shut down from a reactivity point of view (i.e., no additional energy release resulting from neutronic considerations), the CACECO code (Ref. 21)was used to calculate temperature and pressure transients within the containment. In addition, the data derived from a fairly comprehensive set of sodium/concrete interaction tests (Ref. 22) were used to confirm that the core debris could be adequately cooled to bring the entire accident sequence into a long-term quiescent state (Ref. 23).

Combining the new modeling capabilities with the experiments conducted within the FSAR phase, both the applicant and the regulator agreed that the 150 MW-s energetic release determined for the BDBA was adequate for assessing containment response.

IV. Key Open Safety Issues Unresolved from PSAR

The key safety issues that remained open after the regulatory processing of the PSAR were as follows:

- Natural Circulation & Cooling and Emergency Power
- Piping Integrity (the basic design features and in-service inspection measures)
- Beyond-Design-Basis Accidents
- Design Fallback Features (such as whether the head compartment should be sealed and whether an ex-vessel core catcher was needed)

All of these issues were addressed during the preparation, submittal, and review of the FSAR.

The natural circulation and piping integrity issues will be addressed separately in subsequent sections of this report. Emergency power was agreed to be sufficient with the installment of two diesel generators (to complement an independent power supply from the Bonneville electrical grid). The principal BDBA issues were addressed above, although some aspects of the overall containment margin considerations continued throughout the regulatory process.

In Supplement #1 to the NRC PSAR evaluation (Ref. 24), issued in December 1974, there was agreement that the BDBA consequences were likely manageable. It was further agreed that the head compartment above the core need not be sealed and inerted. This allowed for an air atmosphere, which proved very helpful during operations because operators could directly access and service the moving machinery located in that region during operation.

In Supplement #2 to the NRC PSAR evaluation (Ref. 25), issued in March 1975, it was agreed that an ex-vessel core catcher was not needed. This was a major step forward in the regulatory review of sodium-cooled fast reactors. It may be recalled that it was a cooling fin, originally attached to a core-spreading device located inside the primary vessel below the core of the Fermi 1 fast reactor (located near Detroit, Michigan), that came loose during operation and was swept up into the core, blocking coolant flow through a cluster of assemblies and causing partial fuel melting. This ironic situation, wherein a device specifically installed for safety reasons actually ***caused*** a severe safety problem, led both designers and regulators to openly question whether systems installed for "hypothetical accidents" were really warranted. During the design of the FFTF, a special cavity below the reactor vessel was specified and actually installed. However, in attempting to design a "core catcher," both the applicant and the regulator agreed that it would be very difficult to provide guaranteed cooling needed for such a device, and that the extra accommodations might prove counterproductive. Hence, a mutual decision was made to fill the "core-catcher" room with concrete and eliminate the device entirely. The ACRS concurred with these decisions (Ref. 26).

The FSAR was submitted in March 1976. During the review of the FSAR, the applicant and the regulator reached agreement on all safety issues except for the following:

- natural circulation cooling
- piping integrity
- control room habitability
- containment margins

Three formal rounds of questions (including 28 separate sets of submittals totaling 766 items) took place during the FSAR review process. The FSAR was formally approved by the NRC in August 1978 (Ref. 27). A supplement to the FSAR was released in May 1979 with advice to attach a sand and gravel venting system to the containment to ensure that any vapors generated during a BDBA would be scrubbed before being released into the environment. More will be said on this issue later.

V. Resolution of Key Open Safety Issues

Natural Circulation For any reactor system designed for the coolant to flow upward through the core, there is a desire to have adequate margin in the overall heat transport system to allow for core cooling during a normal shutdown if forced flow is not available (i.e., should the primary pumps fail to perform). FFTF was specifically designed for this situation. The elevation differences between the major components shown in Figure C.1 illustrate the thermal buoyancy head expected to be available to allow natural circulation to perform the required core heat removal without the benefit of the primary pumps during normal shutdown conditions. Further, the inertia built into the primary pumps was specifically designed to ensure an extended coastdown time in order to allow a transition to natural circulation cooling to be effective.

Substantial analysis was conducted to determine whether coolant temperatures could be kept sufficiently below the boiling point to ensure safe shutdown under such conditions. The NRC accepted the analytical results as providing a high level of confidence, but they stipulated that the applicant should demonstrate the performance of natural circulation cooling during the startup phase of actual operations.

Hence, during acceptance testing for the FFTF, a series of tests was performed to confirm and demonstrate the effective transition to natural circulation for decay heat removal. All tests were initiated from a complete loss of electrical power to the primary pumps (both the large main motors and the small pony motors).

Figure C.4 illustrates the transient response of the core during the final test—a scram from full power to natural circulation. The results clearly indicated that the core could be cooled by natural circulation without power to the primary pumps. The plant remained on natural circulation for approximately 2 and 1/2 days to demonstrate the effectiveness of long-term natural circulation decay heat removal. This test series closed this open safety question.

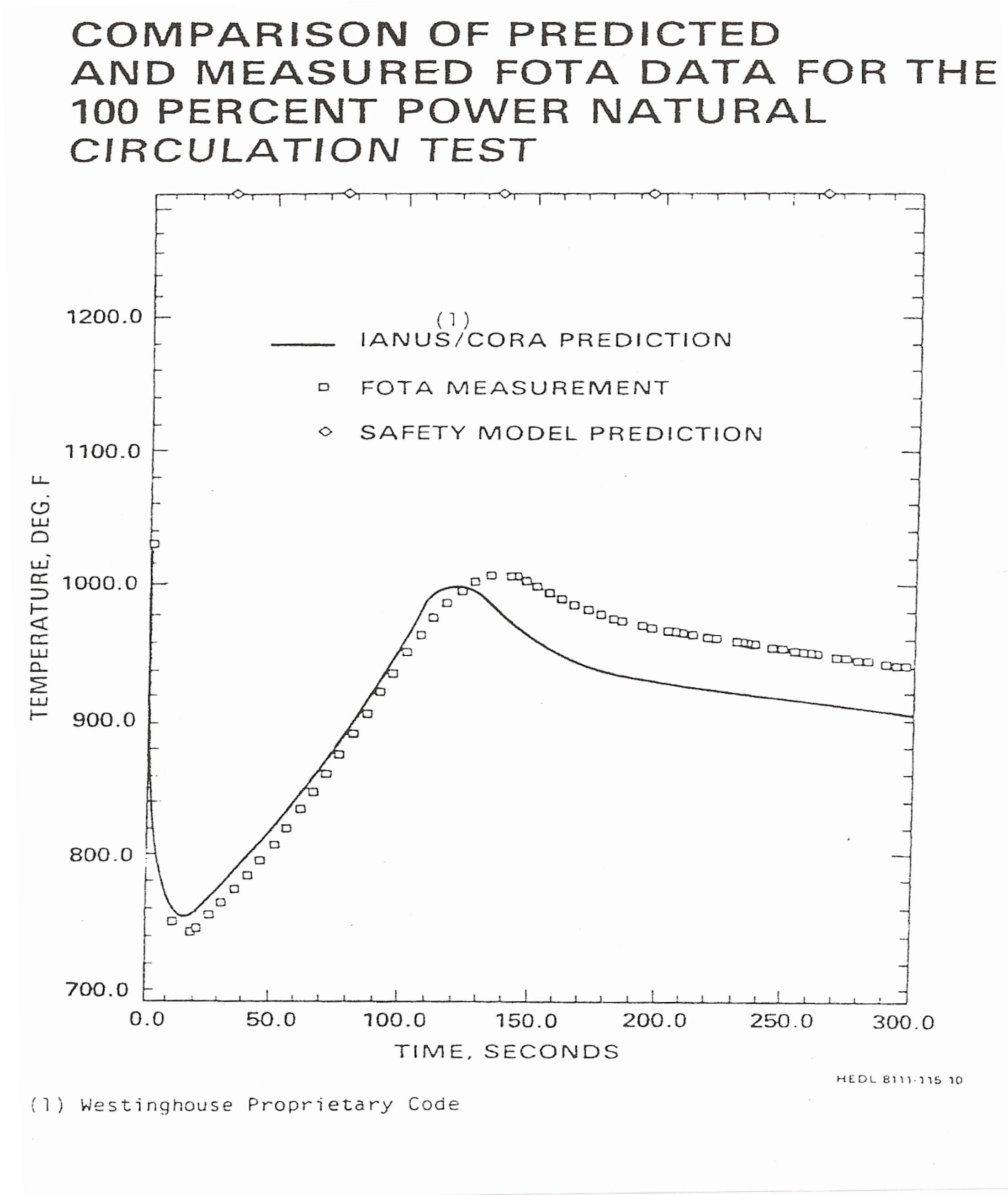


Figure C.4 Comparison of calculated and measured peak coolant temperature in the FFTF for the 100 percent power natural circulation test

<u>Piping Integrity</u> The technical concern about piping integrity was whether a double-ended simultaneous pipe rupture could occur and, if so, what would be the consequences? It was recognized early on that such a possibility would be remote in a sodium-cooled system, primarily because the pressures are so low. Compared to LWR systems, where the pressures are of the order of 100 atmospheres, sodium-cooled systems operate at near ambient pressure—with peak pressures only high enough to ensure proper flow through the system.

Furthermore, investigations on the ductile materials used for primary and secondary piping provided considerable evidence that small leaks would occur well before the possibility of major breaks in the piping.

Nevertheless, NRC remained uncomfortable. Was there a possibility that piping degradation could commence and be undetected? Could a break occur during a seismic event and remain undetected? For the latter possibility, a “leak before break” situation would not apply. Should a seismic event occur, the reactor would automatically scram, but a pipe break might prevent natural circulation from removing decay heat.

In anticipation of such a concern, guard vessels were included in the original design and they were installed around the reactor inlet piping during construction (along with guard vessels surrounding the reactor vessel itself, the primary pumps, and the intermediate heat exchangers) so that any break in that crucial section of the inlet piping would be contained and provide some back-pressure. This would more than likely allow natural circulation to provide the necessary heat removal capability. However, the NRC insisted that a sodium aerosol leak detection system be added. This system was designed, built, and installed—resolving this open safety issue (Ref. 28).

Control Room Habitability During the review process, the question of control room habitability was raised. The concern was the ability of the operators to properly function in the event of a major accident. One possibility was to build a separate and remote control room (with capability to arrest accident conditions). The other option was to modify the original design of the control room to shield the operators from any unacceptable levels of sodium aerosol or radiation that might occur during a major accident. The project selected the latter option and, with the concurrence of the NRC, made provisions to seal (isolate) the control room under major accident conditions—including the possibility of a tornado. The locations of air intakes, complete with isolation dampers, were also upgraded.

Containment Margins As noted earlier, the NRC and ACRS agreed that the containment system for FFTF did not have to meet the energy release levels that would be calculated by the ultra-conservative Bethe-Tait approach. The 150 MW-s energy release, complete with the calculated structural consequences, was judged to be adequate. However, it was mutually agreed to conduct a series of core melt-through tests, consisting of sodium/concrete interactions and hydrogen interactions. By folding the results of these tests into the BDBA analyses, the containment was shown to be adequate. As a final precautionary measure, however, the regulator requested that a gravel bed filter system be installed to ameliorate any containment release of hazardous substances. Such a system was built and installed.

A 10-Year Anniversary of LMFBR [liquid-metal fast breeder reactor] Progress was held at the 1990 Annual Meeting of the American Nuclear Society (ANS), wherein several papers summarized recent progress in the advancement of fast reactor technology (Ref. 29). One of the papers (Ref. 30) provided an overall summary of the regulatory experience gained regarding the BDBA analyses conducted in support of the FFTF.

VI. Major Safety Test Programs

As noted earlier, the principal purpose of the FFTF project was to test fuels and materials projected to be needed for a successful fast breeder reactor program. Accordingly, an aggressive testing program was conducted to evaluate a series of different fuel types and cladding systems that could be safely used for new liquid-metal-cooled fast reactors. The fuel types were oxide, metal, carbide, and nitride. The principal cladding types were Type 316 20 percent cold-worked stainless steel, D9 (an advanced austenitic stainless steel), and HT-9 (a ferritic steel).

In addition, an innovative passive safety testing program was conducted in 1986. Static tests were conducted (to better separate the inherent reactivity feedback coefficients in FFTF) and a unique set of transient tests were conducted, first for low-flow conditions and then with a new invention called Gas Expansion Modules (GEMs). A summary of these testing advances, all relevant to reactor safety, is included below.

A) Fuel Systems

The principal fuel system tested for a wide variety of compositions, configurations, and burnup was mixed oxide fuel. The plutonium content for the inner zone of fuel was typically 22.4 percent and that of the outer zone about 27.4 percent. The outer zone was of a higher fissile content to help flatten the radial power curve. Over 48,000 full-length (3-ft) driver pins were irradiated in FFTF as well as over 16,000 full-length test pins (Ref. 31). This clearly led to statistically significant numbers for fuel evaluation purposes.

Figures C.5 through C.9, respectively, illustrate the FFTF fuel system, the driver pins, the driver fuel assemblies, the control assemblies, and an overall core map.

A special Core Demonstration Experiment (CDE) program was conducted using 23 fuel assemblies consisting of 169 pins per assembly of annular fuel and HT9 cladding (based on the CRBR design). The core map for this case is illustrated in Figure C.10. The purpose of this test program was to demonstrate the acceptable performance of MOX in a heterogeneous core configuration. CDE consisted of ten fuel and six blanket assemblies located at the center of the FFTF, lead fuel test assemblies operated under one- and two-sigma conditions, and fuel assemblies located at the edge of the core at low power conditions. Fuel from the CDE program was successfully irradiated to very high burnups, with some 500 pins reaching levels beyond 220 MWd/kg (Ref. 32). In addition to the successful steady-state irradiation program, pins from this core configuration were transient-tested in the TREAT reactor with results even surpassing the robustness of the base fuel pins. This author is of the opinion that mixed oxide fuel (with the compositions used in the FFTF program) represents a proven, licensable fuel system for sodium-cooled fast spectrum systems.

Although the performance of metal fuels in EBR-II was encouraging, there were still reservations about how well the fuel would perform in full-size pins in a more prototypical fast reactor environment. Accordingly, pins were manufactured and irradiated in the FFTF. Early results (Ref. 33) to 10 atom percent burnup were quite encouraging and post-irradiation examination (PIE) showed behavior consistent with the existing data base from the shorter, metallic fuel pins irradiated in EBR-II. Other metal fuel pins were irradiated to nearly 150 MWd/kg at very high pin power (i.e., nominal peak of 56 kW/m or 17 kW/ft) and were reported (Ref. 34)to have performed quite well, although fuel column length increases of 7 percent were surmised from thermal data collected during irradiation. These length increases saturated at about 1.5 atom percent burnup and had no apparent degradation of performance. This is something that must be accounted for in the design and operation of a fast reactor using these metal fuels. Before the FFTF was shut down, more than 1,000 metal fuel pins (U-Pu-Zr) were irradiated with no pin breaches being observed. This lends credence to the selection of metal fuel for an advanced reactor.

It was concluded, based on both the EBR-II data and that obtained in FFTF, that a full core of metal fuel could be successfully loaded into the FFTF for full power operation (Ref. 35). That step was never taken, however, because of the early termination of the FFTF operational program.

Both carbide and nitride fuel systems have been considered for fast reactors and some experience has been obtained from foreign reactors. Accordingly, a few pins of both types were tested in FFTF, but the numbers are quite small.

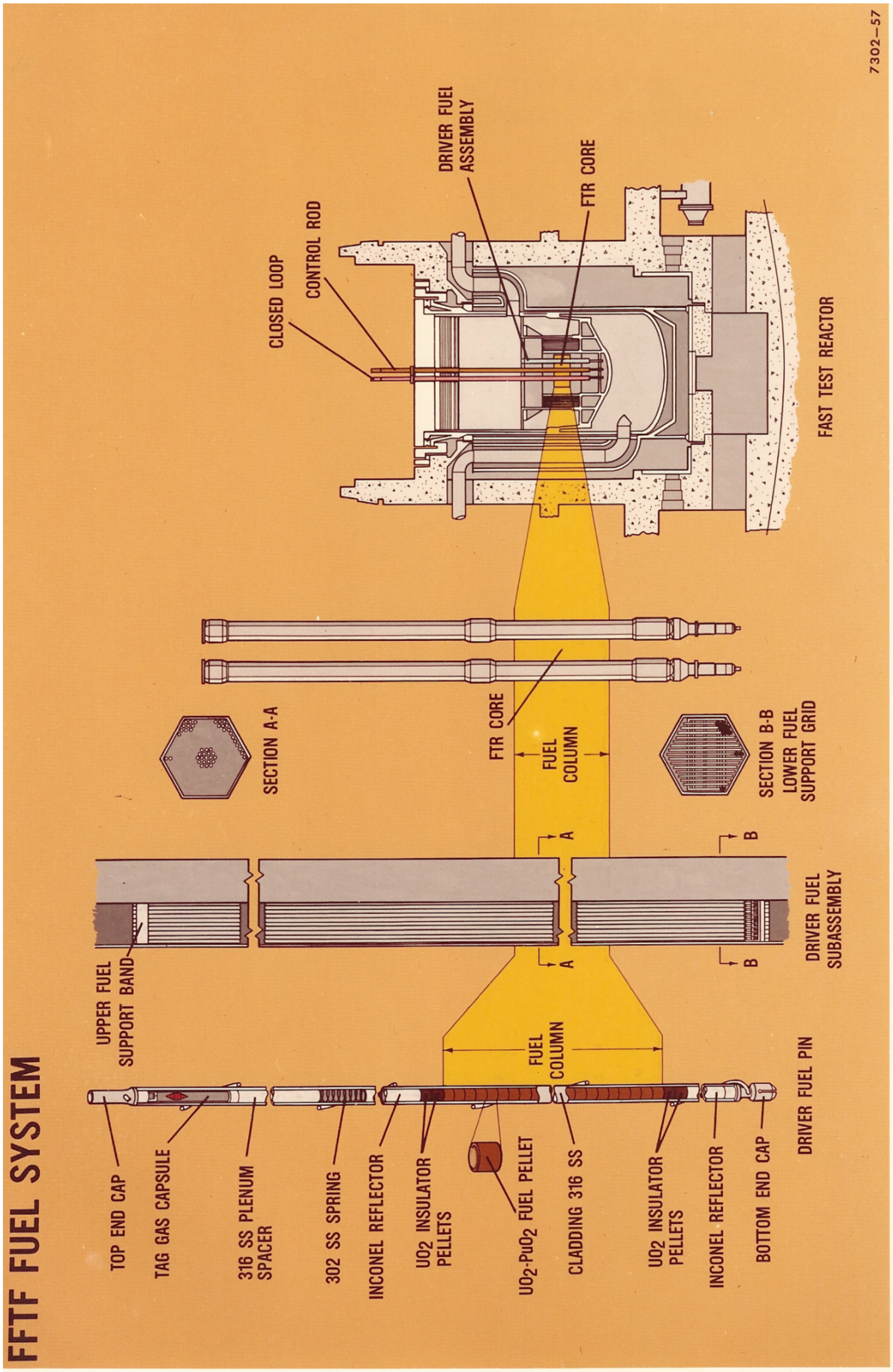


Figure C.5 FFTF Fuel System

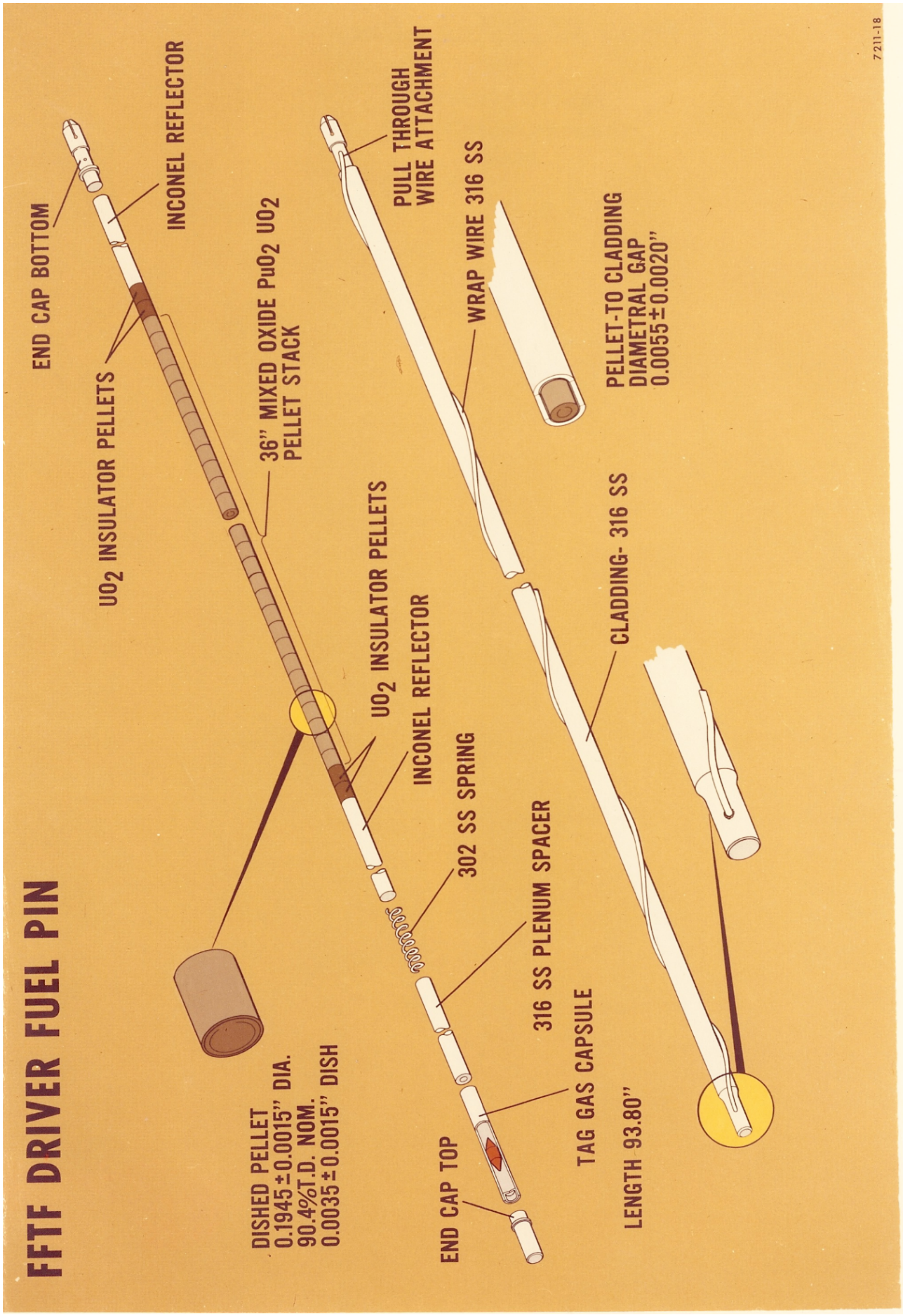


Figure C.6 FFTF Driver Pins

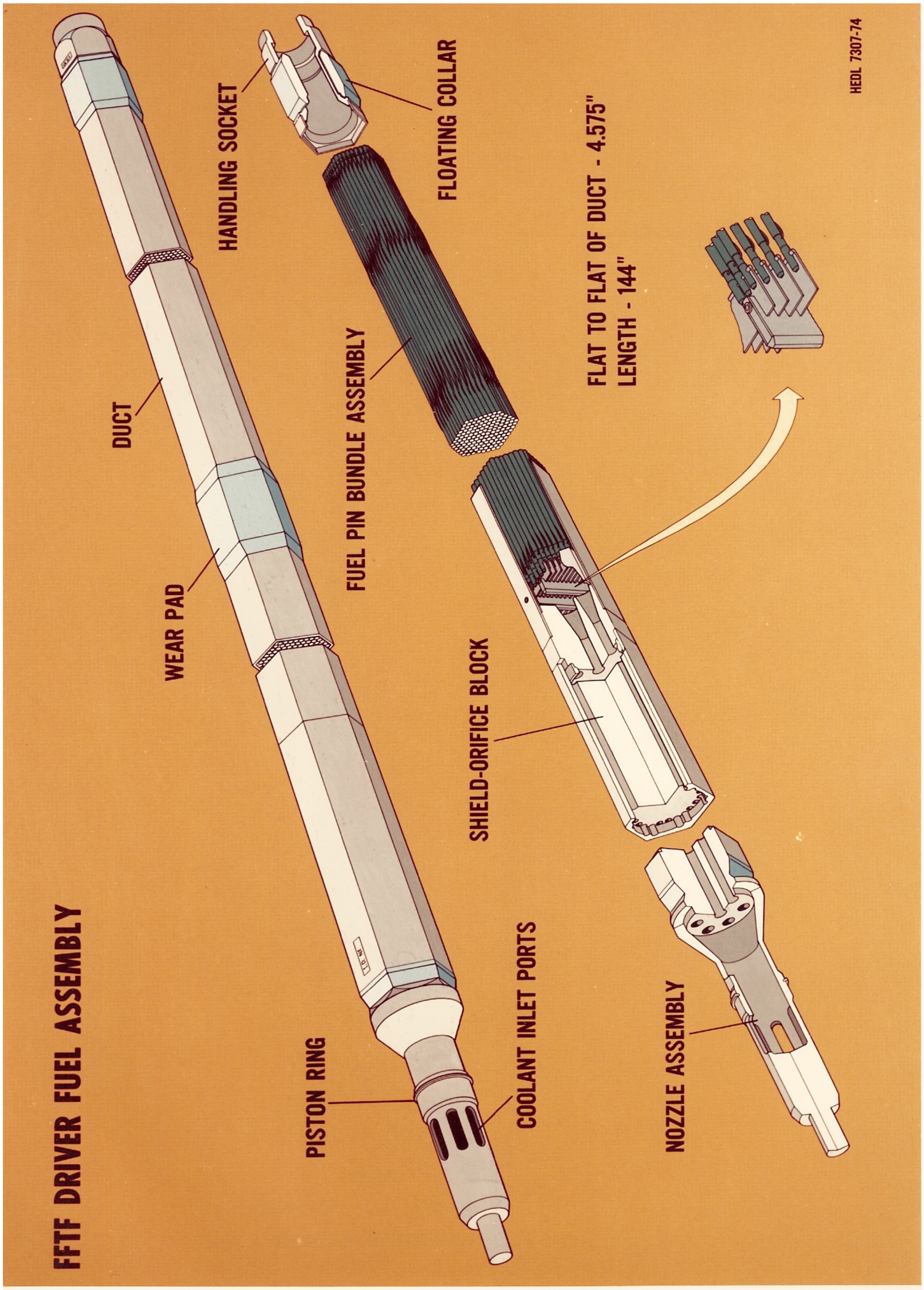


Figure C.7 FFTF Driver Fuel Assemblies

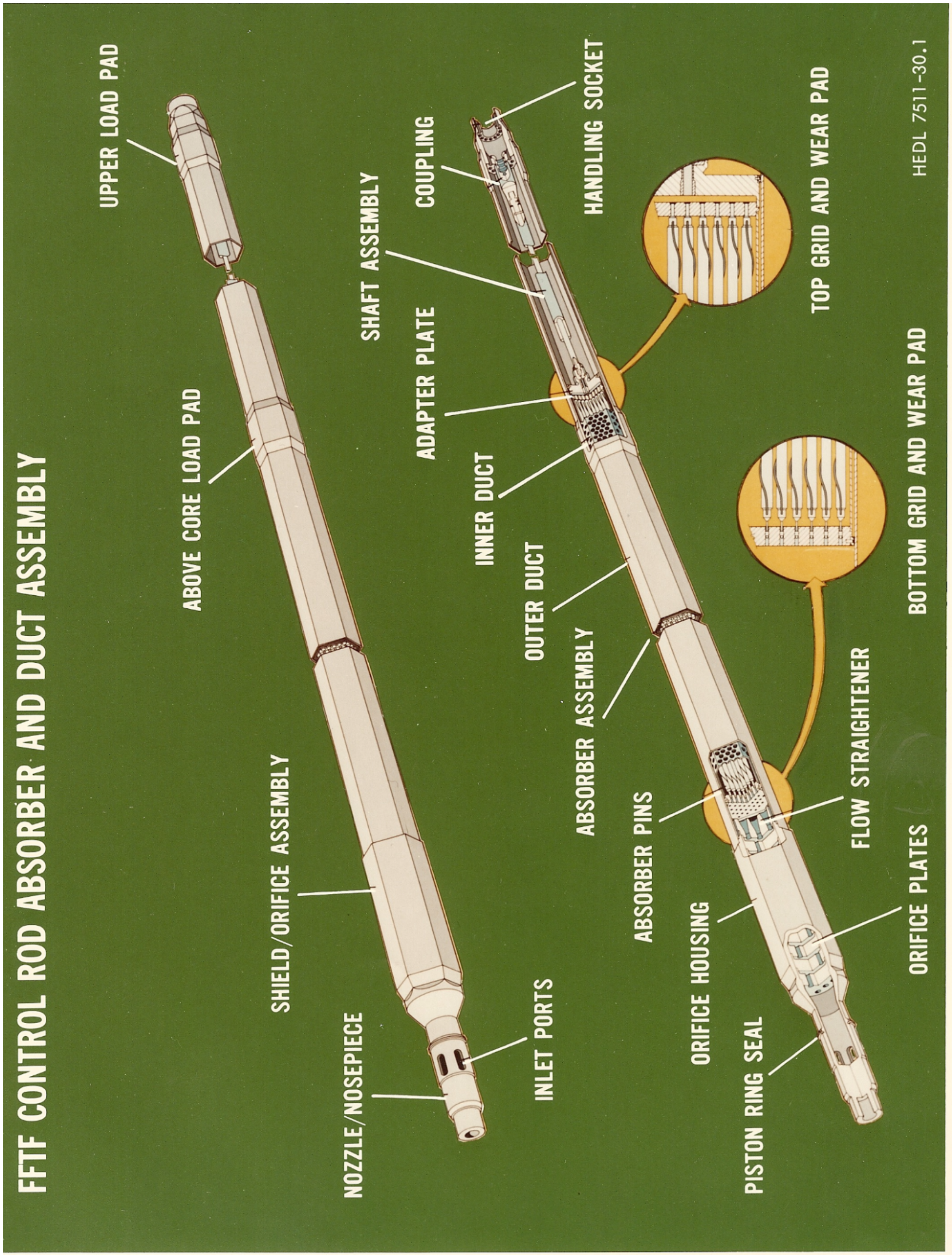


Figure C.8 FFTF Control Assemblies

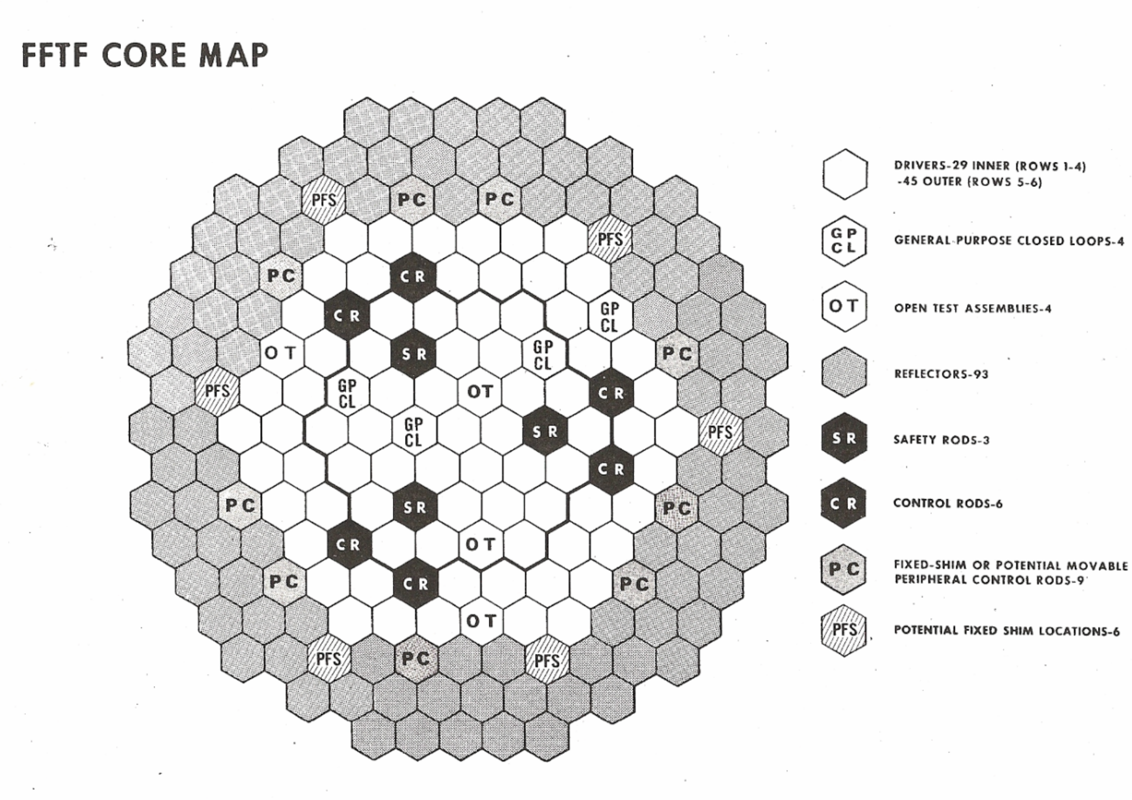


Figure C.9 FFTF Core Map

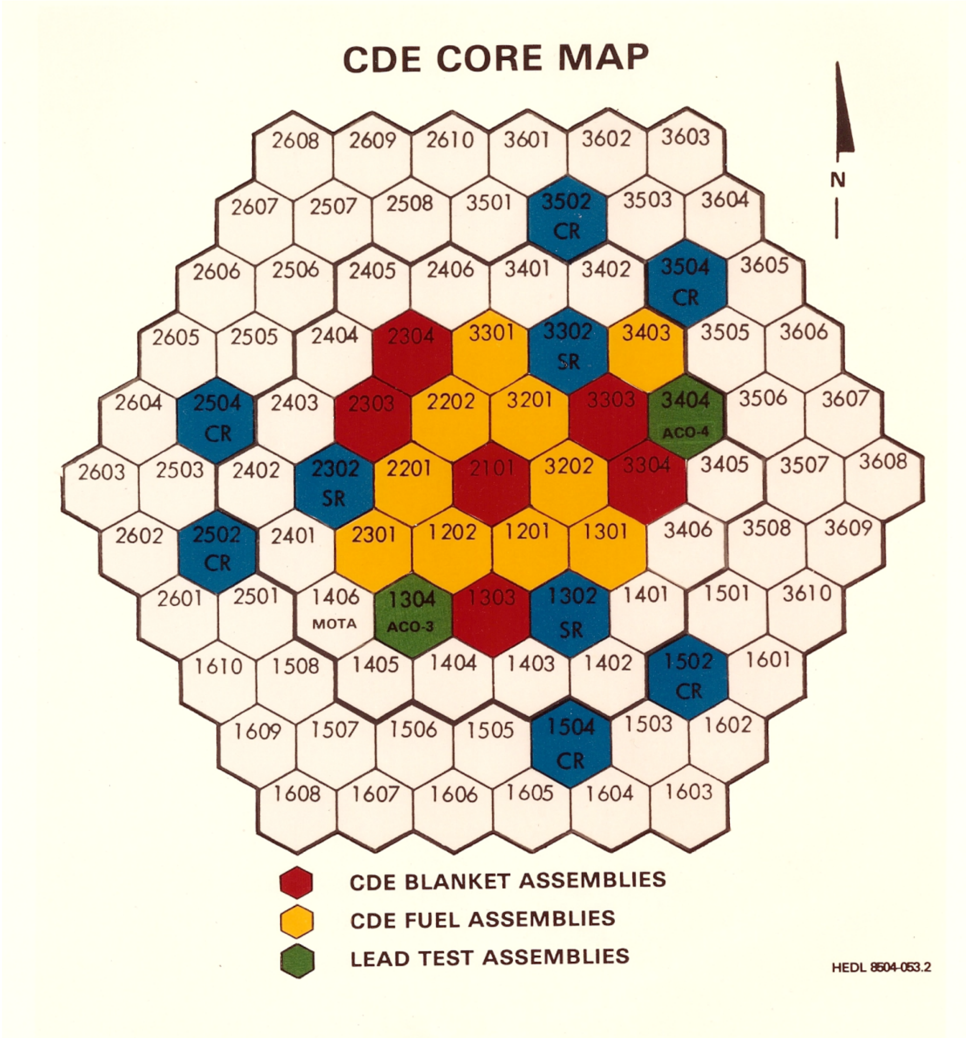
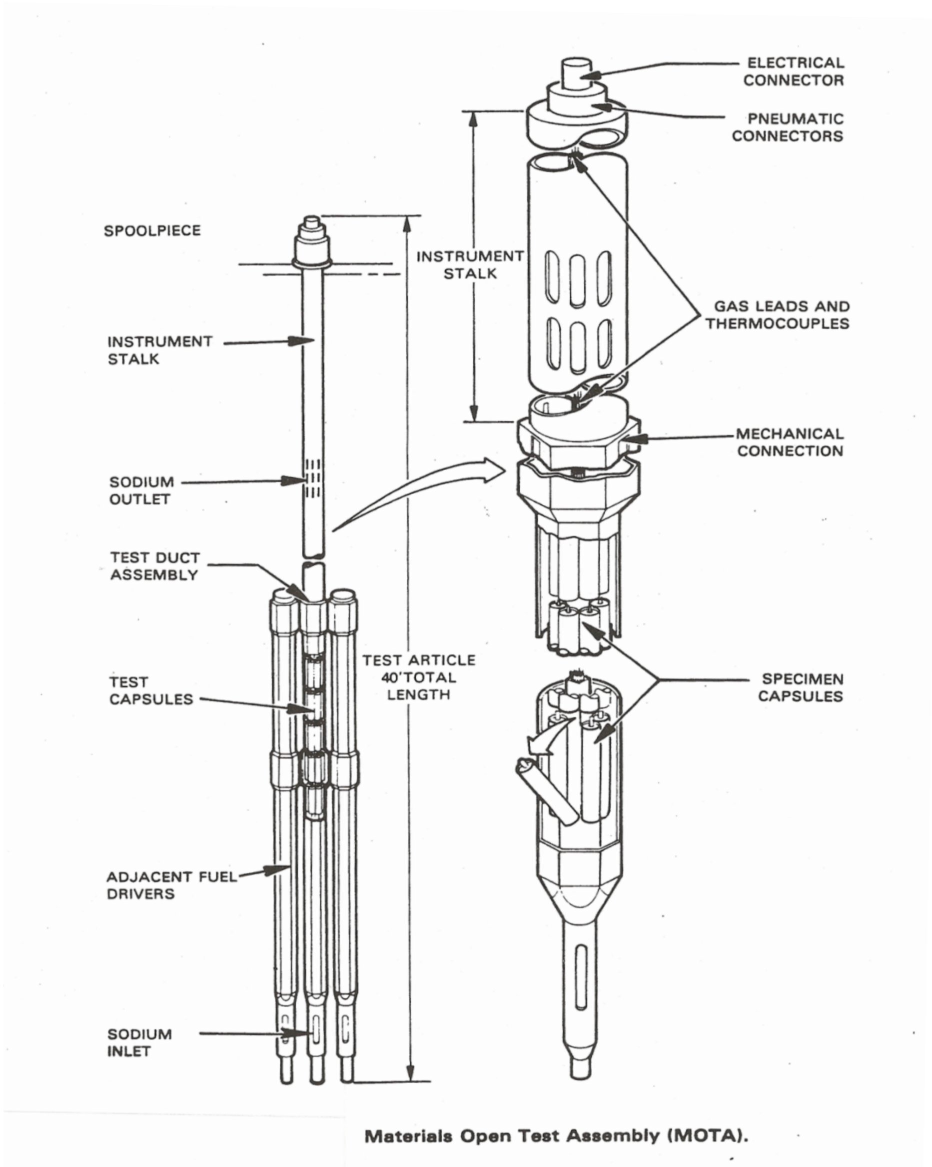


Figure C.10 The FFTF CDE Core Map

One of the safety concerns for the carbide fuel pins is that the very high thermal conductivity of the carbide system causes it to satisfy the spontaneous nucleation criterion (Ref. 36), which could lead to a sodium-vapor explosion under severe accident conditions. A similar concern might be expressed for the metal fuel system with its even higher thermal conductivity, but several tests at ANL confirmed that molten metal uranium fuel would result in a froth when it encounters sodium. Similar tests were not conducted for carbide fuel, at least to the knowledge of this author. In any event, only about 18 full-length sodium-bonded and 200 helium-bonded carbide pins were irradiated in FFTF.

Approximately 54 short nitride pins were irradiated in FFTF—mainly of direct interest to the reactor space program. The initial evaluation of these nitride pins indicated favorable safety characteristics (Ref. 37). A potential concern of nitride systems is disassociation of the fuel at very high temperatures. However, the conditions tested in FFTF were at temperatures far below this safety concern.

One of the standout features of the FFTF is the Materials Open Test Assembly (MOTA). This assembly (illustrated in Figure C.11) is very heavily instrumented and has the capability of accepting various special gas mixtures in differing axial locations to allow a fairly wide range of operating temperatures (the introduction of varying mixtures and rates of gas allows more or less cooling capability). Accordingly, tens to hundreds of small samples of differing materials can simultaneously be tested and carefully monitored in this distinctive test assembly. During the 10-year operating life of FFTF, on the order of a thousand material samples were irradiated. This allowed a very rapid way to screen new materials for eventual testing under full-scale conditions.



Materials Open Test Assembly (MOTA).

Figure C.11 Materials Open Test Assembly (MOTA)

Substantial testing of full-length pins with various cladding materials was successfully carried out. The original driver pins were clad with 20 percent cold-worked 316 stainless steel. The original subassembly ducts were likewise made from this material. Whereas such testing was considered successful, the burnup was limited to about 120 MWd/kg (fluence of

1.6 x 10^{23} N^0/cm^2) because of excessive duct swelling. Pioneers in the fast reactor program will recall that it was void swelling in the cladding and subassembly duct material, discovered in the 1960s in the Dounreay reactor in the U.K. that inflicted a huge "damper" on the fast reactor program. After that early discovery, metallurgists worked overtime to determine the cause of such void swelling and came up with a variety of materials for testing to see whether that problem could be overcome. Without such new materials, the burnup levels for fast spectrum reactors would be greatly limited, and high burnups are required to justify the cost of the fissile enrichments needed for such systems.

Accordingly, a new austenitic stainless steel called D9 was extensively tested in FFTF with favorable results. This material allowed burnups up to about 160 MWd/kg (fluence of 2.5 x 10^{23} N^0/cm^2) before duct swelling became the limiting factor. This was followed by using HT9, which is a ferritic material. This cladding and duct material allowed burnups to reach well over 200 MWd/kg (3.0 x 10^{23} N^0/cm^2), which greatly exceeded the original goal burnup of FFTF (about 80 MWd/kg) and should satisfy the economic conditions needed for commercial deployment (Ref. 38). The only disadvantage of HT9 is that the acceptable operating temperature is less than that of the austenitic steels. Hence, some work was started with dispersion-strengthened materials (successfully tested in small quantities in MOTA but not converted to full-length fuel testing).

It should be noted that the life-limiting structure for fuel burnup in FFTF was generally the duct, rather than the cladding for the fuel pins. Only a limited amount of void swelling of the ducts could be tolerated before a concern would arise regarding the ability to withdraw burned fuel assembles from the core without undue friction.

B) Passive Safety Tests

Whereas the overall reactivity feedback can be readily determined in an operating fast reactor system by forcing the reactor into various controlled transient situations, it is often difficult to separate the various feedback mechanisms. Hence, a testing program was set up to place the FFTF into a variety of steady-state conditions and carefully analyze its response in an effort to isolate the key reactivity feedback mechanisms.

For instance, by keeping the fuel temperature constant while altering the power and flow levels, the Doppler feedback could be nullified while changing cladding and duct temperatures, thus measuring axial and radial feedback. Likewise, temperature variations could be induced while keeping cladding and duct temperatures constant—thereby isolating the Doppler effect. Through a careful planning and test execution process, 198 different static conditions of the reactor provided considerable insight in separating the key reactivity feedback mechanisms operating in FFTF (Ref. 39). One of the key determinations of the testing program was that the grid plate radial expansion was about 40 percent more effective than previously thought to be the case. Having lacked such knowledge during the regulatory processes that preceded reactor operation, only Doppler feedback was relied on for calculating the consequences of off-normal conditions. Given a better understanding of these feedback coefficients, the general conclusion is that the calculated consequences would be even less severe than assumed for the bounding cases used for containment margin analyses.

A set of safety transient tests was then conducted in three basic steps. Step 1 was to test the effectiveness of natural circulation cooling starting from core conditions in which a thermal head did not exist before initiating the transient. Recall that the natural circulation tests during the original startup testing program were conducted by initiating scram from full power (and shutting off power to the primary and pony motor pumps). For such conditions, the initial outlet coolant temperature was high, which would thereby provide a thermal buoyancy driving head to promote natural circulation. The latter tests started at low power, so that the coolant temperatures had to build up to drive natural circulation. This test series also included a test in which the reactor was operated at low steady-state power levels with only natural circulation flow for cooling. These tests were successful in that natural circulation was indeed initiated and the reactor transient proceeded to a stable and cooled configuration (Ref. 40).

Step 2 in the safety transient testing program was to conduct a small “controlled loss-of-flow” transient. The primary purpose of this test was to provide additional calibration material in better assessing the reactor feedback model used in conducting transient analyses for the FFTF (Ref. 41).

Step 3 in the safety transient testing program was the most spectacular. An ultimate (usually unattainable) safety goal of any reactor system would be to design the reactor so that it would automatically shut itself down under any conceivable situation—including unprotected transient overpower (UTOP) and unprotected loss of flow (ULOF) accidents. Both the applicant and the regulator agreed that the FFTF could survive the UTOP with little core damage (even though several subassemblies would need to be replaced under the most severe situations). However, for an unprotected loss of flow accident in the FFTF, it was clear that substantial coolant boiling would occur, followed by cladding melting and subsequent fuel melting. This would result in substantial damage to the core—likely requiring a full core replacement.

Accordingly, a Gas Expansion Module (GEM) was cleverly devised by Jim Waldo to take advantage of neutron leakage from the core during a postulated ULOF. The GEM itself is a very simple device. It consists of a subassembly duct that has been capped at an appropriate distance above the active core region and then inserted at the core radial periphery. As shown schematically in Figure C.12, when the pumps are energized and running at normal speed, sodium flows into the GEMs and becomes static as it pressures the inert gas into the top region of the GEM. This liquid sodium at the core periphery causes neutrons to scatter back into the core and contribute to the neutron balance required to maintain criticality. However, if power is lost to the pumps, they coast down—relieving pressure at the core inlet and the compressed inert gas in the GEMs forces the sodium once residing in the GEM down below the active core region. This automatically provides an escape path for neutrons and they leave the core—resulting in a negative reactivity to shut down the chain reaction.

The reactivity worth of each GEM is determined by its location. For the FFTF, most of the GEMs placed at the core periphery were worth about 17 cents. Hence, nine GEMs were loaded into the core and equally spaced around the periphery of the core, providing a combined negative reactivity worth of about 1.5 dollars on loss of flow. The reactor was then raised to 10 percent power (40 MW) and the PPS was modified to eliminate the automatic reactor scram when all power was cut off to the primary pumps. This process was continued, with the last test being conducted from 50 percent full power (200 MW) and, in all cases, the reactor shut itself down with no intervention from the PPS or the operators. As noted from the results of the most extreme transient, shown in Figure C.13, none of the temperatures in the core reached safety limits before a successful shutdown (Ref. 42).

As a parenthetical note, this author arrived late at the FFTF for the final (most extreme) test at 200 MW and was just entering the gate when the chief test engineer was doing the site-wide PA system countdown leading to the termination of power to the primary pumps. Ten…nine…eight….. As the count wound down, I had instant flashbacks of doing the calculations for such a situation (without GEMs, of course) during the earlier years of the regulatory review. I envisioned the horrendous mess of the core that our computer modeling had predicted….and I thought “We’ve come a long way, baby!”

GEMs introduced -$1.50 into the core on unprotected loss of flow

- **Reactor shut down with no damage**

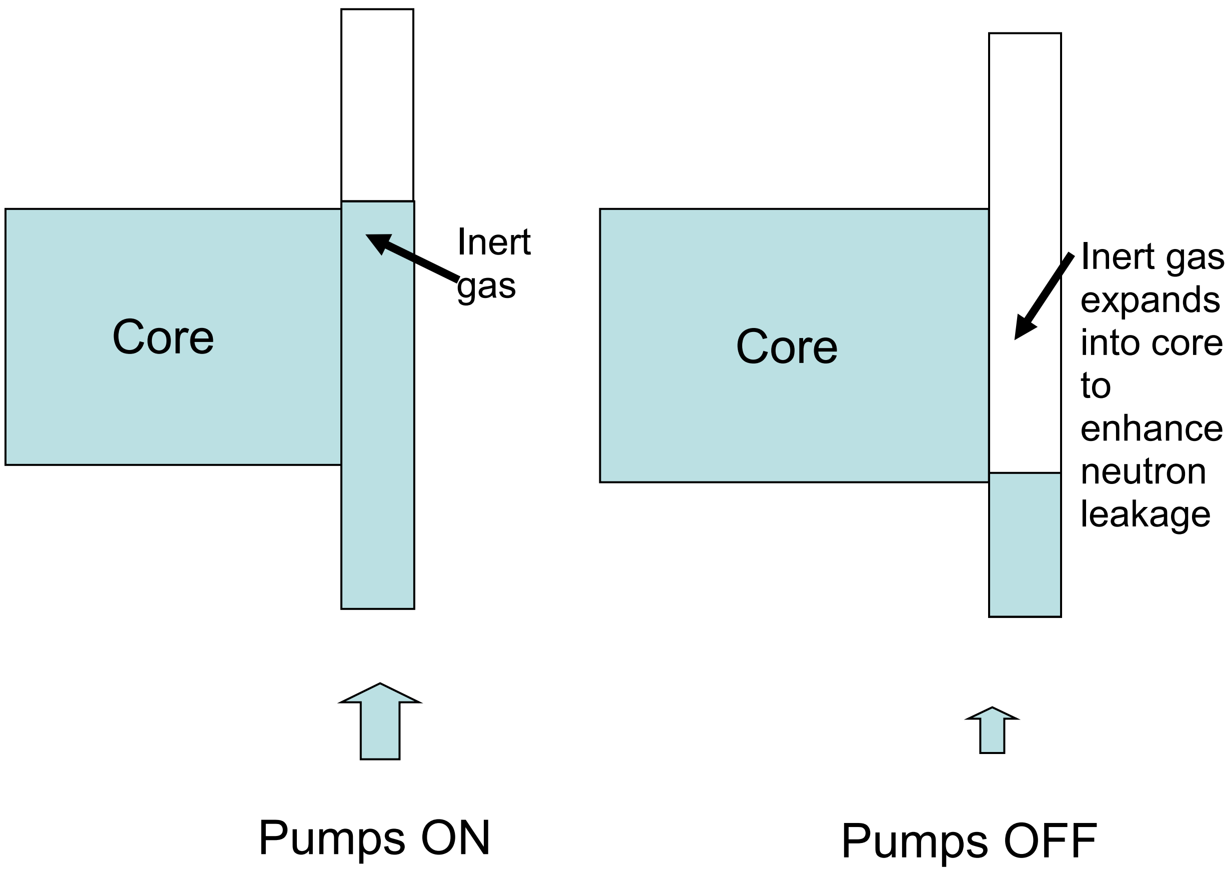
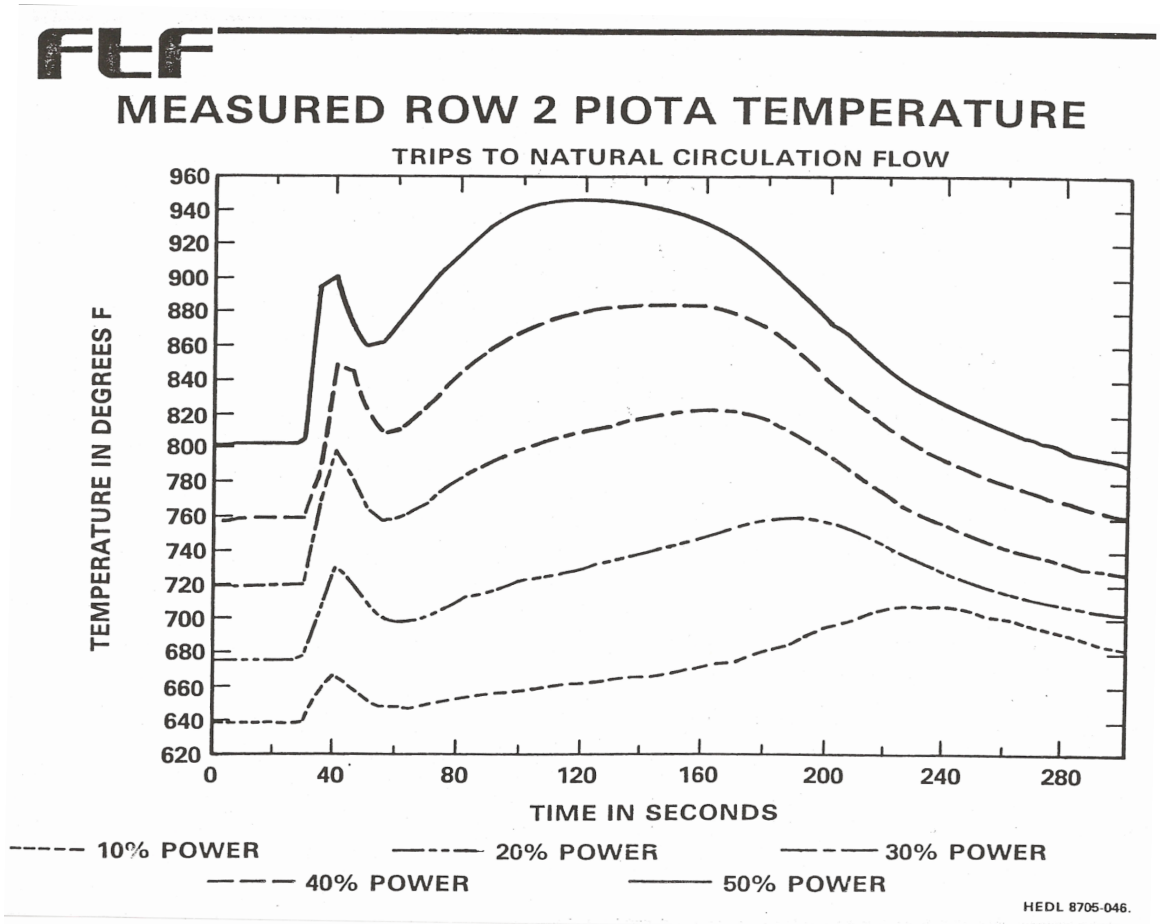


Figure C.12 A sketch of the FFTF Gas Expansion Modules (GEMs)



GEM Tests with Unprotected Loss of Flow

Figure C.13 FFTF transient tests for ULOF from 50 percent full power (200 MW) with GEMs

VII. Key Lessons Learned

Though perhaps oversimplified, I would list four major lessons learned from the regulatory process carried out for FFTF.

1. It is absolutely necessary to **incorporate safety into the design**. Safety is not something to be "added on" as fixes. Certainly we learned that lesson from the Fermi 1 reactor, where cooling fins were added to a core dispersal device below the core well after the design and much of the construction had proceeded. This resulted in an inadequate design and the lack of a good operational analysis. Because of this "band aid," added in haste to satisfy a safety concern that arose during the licensing process, a cooling fin became dislodged during operation and led to significant fuel melting. Indeed, this incident prevented a successful legacy for this reactor. In the case of FFTF, employing the Lines of Assurance approach worked very successfully.

2. **Natural circulation was demonstrated to work** for FFTF and, by analogy, can be shown to work for liquid metal-cooled systems designed with a sufficient thermal head in the primary system. This is a powerful safety feature.

3. **Most of the safety emphasis should be addressed to Protected Accidents**. Indeed, conducting BDBA analyses is fascinating, but an overdue emphasis on accidents that can never occur, or are of extremely low probability, can become a misuse of precious resources. Providing a comfortable margin against unforeseen circumstances is clearly laudable and must be done, but such efforts and expenditures of resources must be kept in perspective. A major advance during the FFTF regulatory review was to "put to bed" the ultra-conservative Bethe-Tait approach to determining BDBA consequences. Another advance was the agreement from the regulator that a core catcher was not needed.

4. The stiff regulatory process conducted at FFTF clearly indicates that **a large oxide-fueled, sodium-cooled fast reactor is licensable.** The many inherent safety features (e.g., low pressure, large margin to coolant boiling, etc.) provide an exceptionally favorable system with a large resiliency to thwart off-normal conditions. The same statement can likely be made for metal-fueled sodium-cooled fast reactor systems, although a full core of metal fuel was not tested because of the early termination of the FFTF program.

CRBR: The Clinch River Breeder Reactor Project

I. Regulator Review History

Because the CRBR was proposed as a fully commercial liquid metal-cooled fast reactor, it was clear that a full-scale NRC review would be necessary. The unusual part of the CRBR licensing process was that it was carried out in two distinct time frames.

Phase I of the licensing process started in 1974 and it was terminated in the spring of 1977 when President Jimmy Carter ordered a work stoppage. He was concerned that the building of fast reactors using plutonium (which was being separated in a pure form using the PUREX process developed by the weapons program) might lead to nuclear proliferation. By stopping commercial reprocessing in the United States, he believed that this example would lead to the termination of fuel recycling in nuclear programs worldwide. History has shown that his action had precisely the opposite effect; namely, other nations, such as France and Britain, seized on the opportunity and proceeded to develop a worldwide oligopoly in the reprocessing business.

In any event, President Ronald Reagan reversed the reprocessing decision and the CRBR resumed the licensing process in September 1981. This second phase was terminated, however, when the U.S. Congress stopped the process in November 1983.

The major accomplishments reported by the project (Ref. 43)during Phase I included an agreement by both the applicant and the regulator that both containment and confinement would be employed at CRBR. Also, the seismic criterion was set to be a 0.25 horizontal ground acceleration.

The major accomplishment noted for Phase II was gaining an exemption to permit early site preparation. A Limited Work Authorization (LWA) was granted and a positive conclusion was reached by the Atomic Safety and Licensing Board (ASLB) to grant a construction permit.

Figure C.14 provides an illustration of the proposed CRBR plant and Figure C.15 shows a sketch of the core map. Figure C.16 is a layout of the heat transport system. Table C.2 contains a listing of key design parameters.

CRBRP

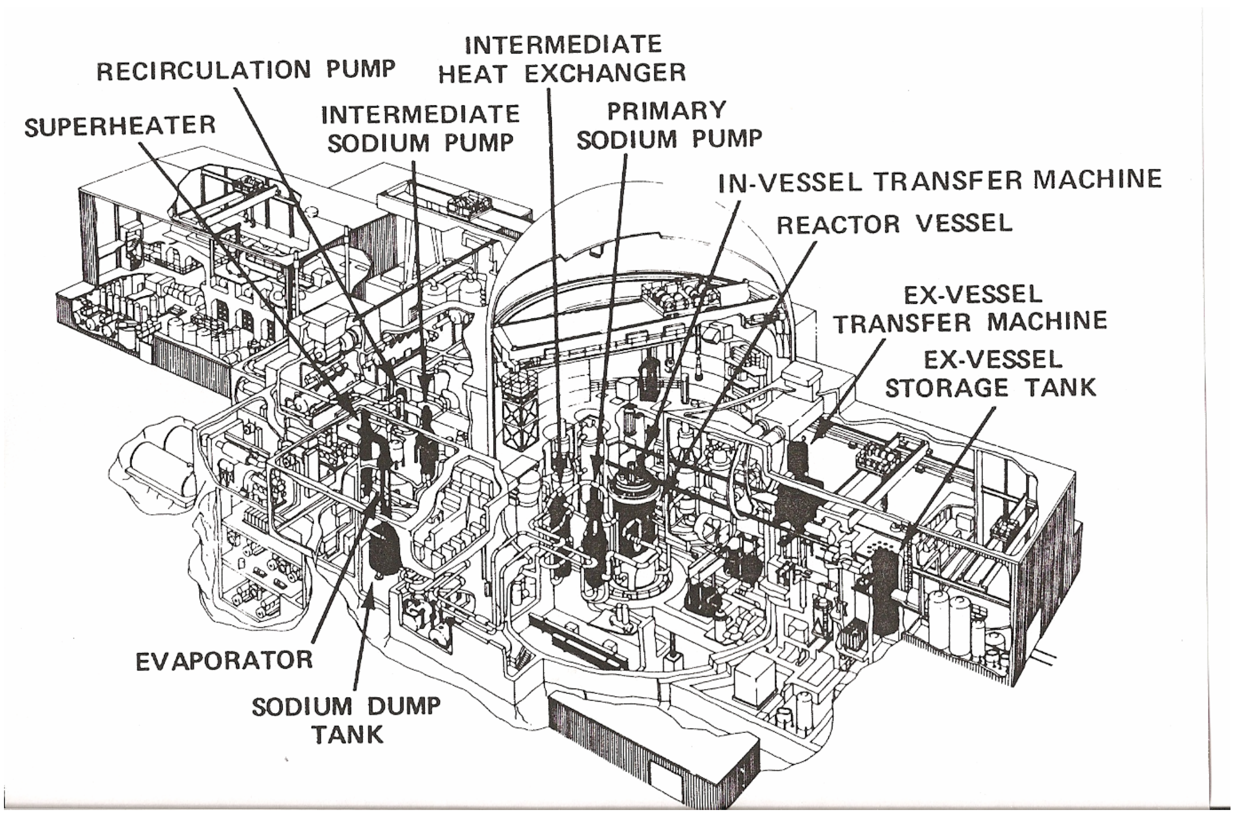


Figure C.14 The proposed CRBR plant

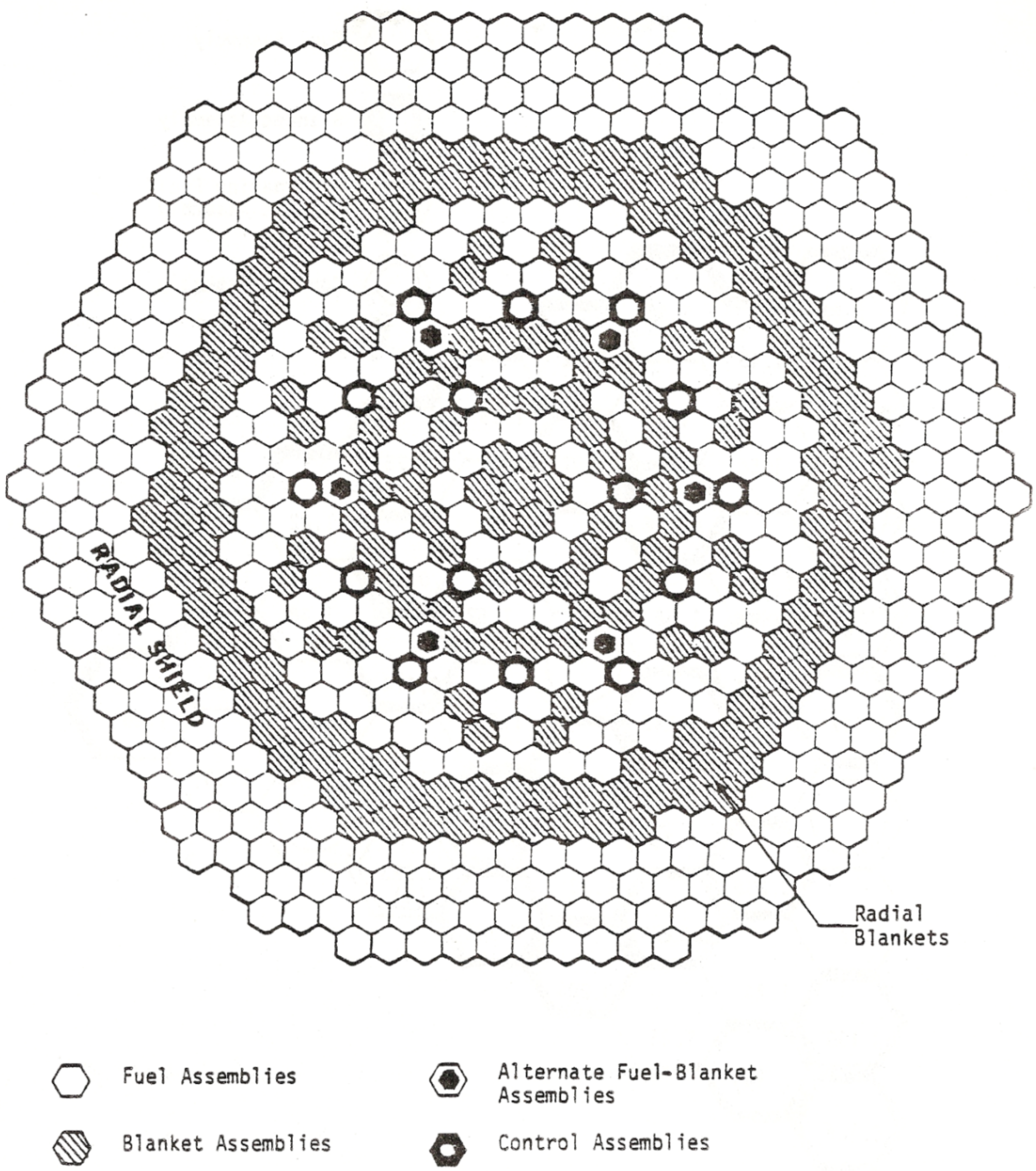


Figure C.15 The CRBR core map

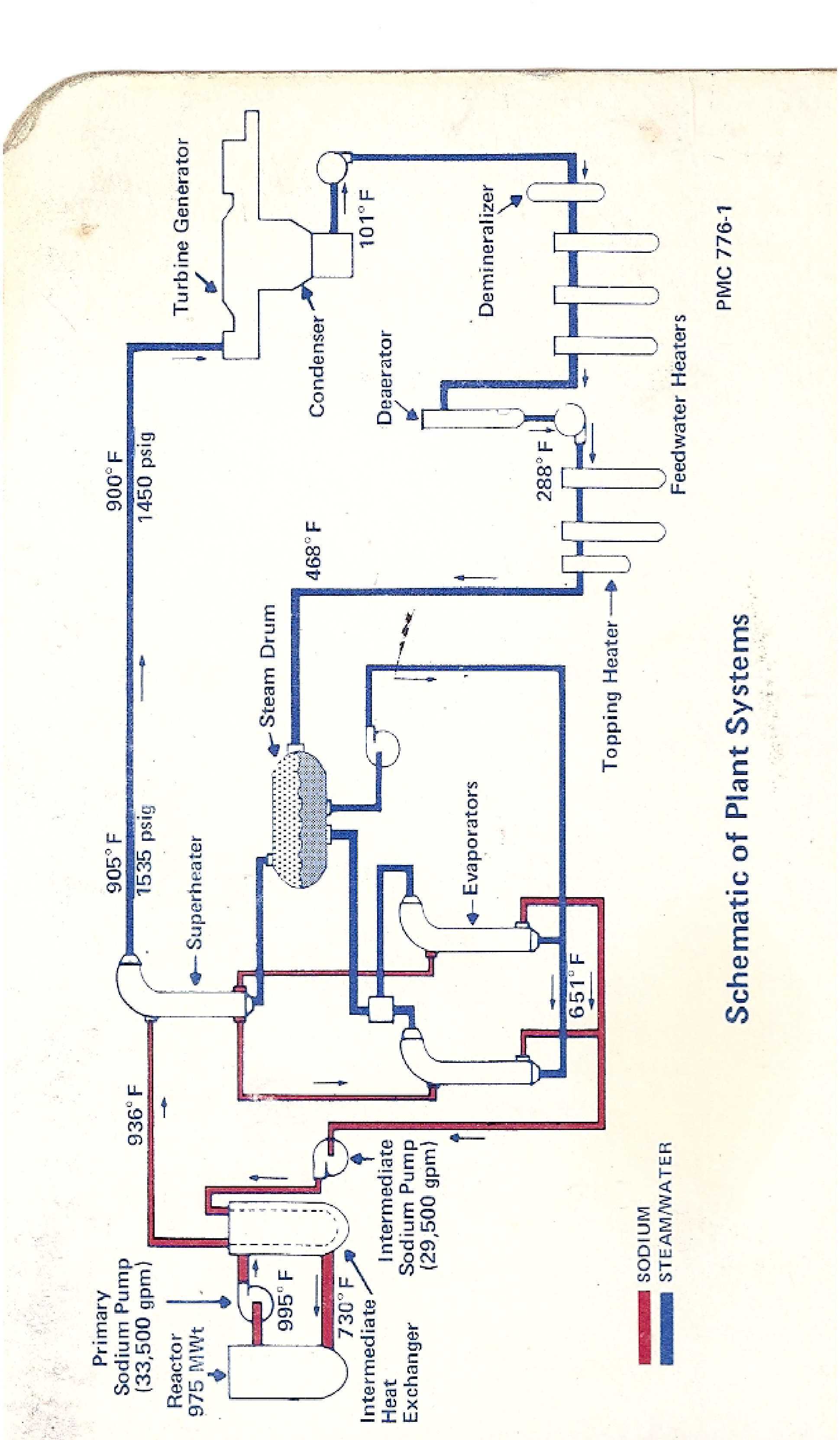


Figure C.16 The CRBR heat transport system

Table C.2 CRBR Design Parameters

Owner:	Energy Research and Development Administration
Architect-Engineer:	Burns and Roe, Incorporated
Lead Reactor Manufacturer:	Westinghouse Electric Corporation
Constructor:	Stone & Webster Engineering Corporation
Reactor Manufacturers	
Reactor, Primary Heat Transport System:	Westinghouse Electric Corporation
Intermediate Heat Transport, Steam Generation Systems:	General Electric Company
Refueling, Auxiliary, Maintenance Systems:	Rockwell International Corporation
Location:	On the Clinch River in Oak Ridge, Tennessee, about 12 miles southwest of downtown
Acreage:	100 acres of a 1364-acre TVA site
Estimated Cost:	$1.95 billion
Transmission System:	161 KV – Tennessee Valley Authority
Coolant:	Liquid metal (sodium)
Cooling Method:	Mechanical draft wet cooling tower requiring approximately 3,584 gpm makeup water
Construction Force at Peak:	2400
Operating Personnel:	160-200
Initial Startup:	1983

OVERALL PLANT

Generator Output, MWe	380
Thermal Power, MWt	975
Gross Plant Heat Rate, Btu/kWh	8,881
Plant Availability Factor	.85
Number of Primary Loops	3
Containment Diameter, ft	186

REACTOR

Fuel Material	Pu/U Oxide
Cladding Material	SS 316
Fuel Rod Diameter, in	.23
Fuel Rod Pitch Diameter Ratio	1.26
Number Fuel Rods/Assembly	217
Number Core Assemblies	198
Number Blanket Assemblies	150
Core Height/Diameter, ft	3.0/6.2
Maximum Cladding Wall Temperature, °F	1,215
Linear Power Rating, Peak/Avg., kW/ft	14.5/7
Peak Fuel Burnup, MWd/T	150,000
Fuel Volume Fraction	.325
Breeding Ratio	1.2
Doubling Time, yr	23

PRIMARY HEAT TRANSPORT SYSTEM

Reactor Outlet Temperature, °F	995
Reactor Inlet Temperature, °F	730
Pump Flow Rate at Pump Temperature, gpm	33,700
Pump Developed Head at Design Flow, ft Na	450

INTERMEDIATE HEAT TRANSPORT SYSTEM

Hot Leg Temperature, °F	936
Cold Leg Temperature, °F	651
Pump Flow Rate, gpm	29,500
Pump Developed Head at Design Flow, ft Na	330

STEAM GENERATION AND TURBINE SYSTEM

Turbine Cycle	Straight Expansion
Superheater Outlet Temperature, °F	905
Superheater Outlet Pressure, psig	1,535
Steam Flow Rate, Total, 10^6 lb/hr	3.34
Feedwater Temperature, °F	468

COMPONENT DIMENSIONS

Reactor Vessel Diameter, ID, ft	20.25
Reactor Vessel Height, ft	54.67
Primary and Intermediate Pumps Diameter, ft	8.5
Primary and Intermediate Pumps Height, ft	20.0
Intermediate Heat Exchanger Diameter, ft	8.8
Intermediate Heat Exchanger Height, ft	52.1
Steam Generator Diameter, ft	4.33
Steam Generator Height, ft	65.0

II. Major Lessons Learned

Five general lessons learned were reported by the licensing leadership of the CRBR project (43):

1. **Maintain a totally open approach.** The technical staff at the NRC was acknowledged as being quite competent and willing to work hard during the entire licensing process. The applicant willingly disclosed "hard spots" to the NRC staff on a regular basis. This developed needed trust.
2. **Keep economics in mind.** The applicant reported spending about $1 million per day for the total plant project. For Phase I, licensing was not on the critical path, so the pace

was a bit slower and there was somewhat less pressure to yield to points expressed by the NRC if they were believed by the applicant to provide little real safety value. However, for Phase II, licensing was on the critical path. Hence, decisions were sometimes made to accede to NRC-requested changes—even if of questionable safety value in the judgment of the applicant. This was sometimes done to keep the process moving ahead.

3. **All legally allowed actions are possible.** The applicant was able to obtain an early site preparation permit, despite having to fight off lawsuits issued to prevent such work. They successfully proceeded by simply taking advantage of the legal system that allowed such exemptions.

4. **Don't be afraid of being sued.** The licensing leaders at CRBR strongly believed that any applicant WILL be sued, irrespective of actions taken. Hence, it is best to assume that suits will be filed and the best defense is to hire very competent lawyers and provide top technical staff to defend actions taken in good faith.

5. **Have the design nearly complete before starting the licensing process.** This is a lesson that has been learned by the entire nuclear community by now. An incomplete design is understandably very hard to license.

Two common-sense lessons learned were also stated by the CRBR licensing staff:

1. **Establish a small office of about 5 persons within a block of the NRC offices.** This greatly facilitates communication, because frequent face-to-face meetings can take place on a regular basis—allowing many issues to be resolved without the undue expenditure of efforts.

2. **Have more experts in meetings than normally needed.** Most of the issues raised at formal meetings can be answered on the spot if the right experts are in the room. This is especially helpful during ACRS meetings because issues somewhat removed from the mainstream of thought often come up in such meetings. With immediate resolution, a huge expenditure of time and lengthy written responses can be avoided.

Two lessons learned were suggested for licensing new and unusual reactors:

1. **Provide tutorials for new NRC staffers.** The CRBR project invested considerable time and effort into providing fast reactor tutorials for "fresh" NRC staffers to help bring them up to speed as soon as possible. They believed this was a win-win situation.

2. **Categorize the General Design Criteria (GDC) of Appendix A to 10 CFR 50 into three groups:**
   - Those truly applicable (and with which the applicant complies)
   - Those truly not applicable
   - Those complied with in principle, although not in the way compliance is achieved for a LWR

The reason for such categorization is that the cited GDC was written for the standard LWR community and, therefore, does not strictly apply to sodium-cooled fast reactors. Such a categorization helped streamline the licensing process.

Two other comments were noted by the CRBR licensing leaders:

1. They felt the review for an actual license is considerably more demanding than a technical review. They noted the expenditure of approximately 100 man-years of effort

per year, which they felt was nearly an order of magnitude larger than the effort expended for the FFTF technical review. This author cannot directly assess this observation, and would accept it with a bit of skepticism. However, it might be true that the NRC would be relatively more diligent when knowing that a formal license is at stake.

2. They also felt that far too much time was expended on the BDBA events. This is the same observation made by this author for the FFTF review. In the case of CRBR, both the NRC and the ACRS eventually agreed that Class 9 events (BDBAs) were not credible.

As a final observation, the CRBR licensing leaders noted a comment made during an introductory meeting with the NRC staff. A member of the applicant's staff commented, "I know you all believe this will be a tough process because you think that a liquid-metal fast breeder reactor (LMFBR), with its fast spectrum, its small $B_{eff,}$ its Pu inventory, its positive void coefficient, and its chemically reactive coolant, is inherently more difficult to make safe than an LWR. Believe me, when you become thoroughly familiar with these reactors, you will agree that they are inherently more forgiving than LWRs and accidents develop so much more slowly that they are therefore easier to license." At the time of licensing termination, many of the NRC staff members apparently agreed.

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APPENDIX C.2

EBR-II TEST AND OPERATING EXPERIENCE

Prepared for the U.S. Nuclear Regulatory Commission
John I. Sackett, INL

Executive Summary

The Experimental Breeder Reactor-II (EBR-II) operated for 30 years as a very successful test and demonstration sodium-cooled fast-reactor power plant. As a complete power plant, it was the site where the reliability of the system was demonstrated and sodium operating and maintenance technology was established. As an irradiation test facility, it was the site where oxide, metal, carbide and nitride fuels were developed. (Oxide fuel for the Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor Project (CRBRP) was qualified and metal fuel was extensively developed for EBR-II.) As an operational-safety test facility, it was the site where the self-protecting response of a metal-fueled reactor was demonstrated for Anticipated Transients without Scram and the benefits to safety were quantified in a probabilistic risk assessment (PRA). It was also where the safety of operation with breached fuel was demonstrated. As the Integral Fast Reactor (IFR) prototype, it was the site where proliferation-resistant reprocessing and recycling of fuel was demonstrated and fuel containing minor actinides was fabricated and irradiated. When it was decommissioned, its sodium coolant was drained and reacted to produce an acceptable form for disposal and residual sodium was passivated. It provided the impetus for developing and qualifying forms for geologic storage of waste from fuel reprocessing. In short, the EBR-II experience and test program established the viability of sodium-cooled fast reactor power plants.

Figure C.17 Experimental Breeder Reactor II

Introduction

There is an important partnership between fast and thermal reactors because fast-spectrum reactors can burn as fuel the waste that thermal reactors produce (primarily long-lived minor actinides). Studies have indicated that anywhere from 10 percent to 20 percent of the fleet should be fast reactors to effectively manage this waste depending on the rate of growth of nuclear deployment. Further, fast reactors can greatly extend the fuel supply (approaching a factor of 100). Extending fuel supply was the promise of fast-reactor development at the dawn of the nuclear age. The Experimental Breeder Reactor-I (EBR-I), a fast reactor, was the first reactor in the world to produce electricity (December of 1951). The Experimental Breeder Reactor-II (EBR-II) followed, producing power in 1964 and operating for 30 years as a complete power plant. Based on this and extensive international experience, the technology has been shown to be successful.

International Experience

Fast reactor experience is extensive. Small test fast reactors similar in size to the U.S.'s EBR-II have been operated in many other countries to develop and test the technology: Rapsodie in France, the Dounreay Fast Reactor (DFR) in the UK, Kompakte Natriumgekühlte Kernreaktoranlage Karlsruhe (KNK) in Germany, Jōyō in Japan, the Fast Breeder Test Reactor (FBTR) in India, and BOR-60 in Russia. Of these, EBR-II, KNK, and DFR were complete power plants.

In the United States and Russia, small, specialized fast-spectrum test reactors were operated to address questions of physics: the Southwest Experimental Fast Oxide Reactor (SEFOR) and EBR-I in the U.S. and BR-2 and BR-5/BR-10 in Russia.

The next generation of fast reactors was made up primarily of complete power plants that incrementally increased power levels over the test reactors that preceded them. These reactors were Fermi 1 in the U.S., Phénix in France, the Prototype Fast Reactor (PFR) in the UK, SNR-300 in Germany, Monju in Japan, and BN-350 in Russia.

France and Russia operated larger commercial plants, Superphénix (France) and BN-600 (Russia). In addition, the United States constructed and operated a second research reactor, the Fast Flux Test Facility (FFTF), but without an electricity-generating system. The United States also pursued design of the Clinch River Breeder Reactor Project (CRBRP), which was cancelled before construction was completed. A similar fate befell the German fast reactor, SNR-300.

All of these reactors were and are cooled with sodium. Sodium supports a fast-neutron spectrum because of its low neutron moderation and absorption. It has excellent thermal conductivity and high heat capacity, which allows high power density in the core. Its relatively low density reduces pumping power requirements and its large margin to boiling allows operation at atmospheric pressure. The coolant is also chemically compatible with structural materials, which minimizes corrosion in plant cooling systems. However, an inert atmosphere covering the sodium is needed because it is reactive with air. Sodium at temperature will burn if exposed to air and special fire-suppression systems are an important part of reactor design.

Sodium-cooled reactor operating experience is extensive and has resulted in the following major conclusions:

Positives:

Fast reactor fuel is reliable and safe, whether it is metal or oxide. Cladding failure does not lead to progressive fuel failure during normal or off-normal reactor operation.

High burnup of fast reactor fuel is achievable, whether the fuel is metal or oxide. Acceptable conversion ratios (either as breeders or burners) are also achievable with either fuel type.

Sodium is not corrosive to stainless steel components immersed within it.

Table C.3 International Fast Reactor Experience

Reactor	Country	Dates of Operation	Power (MWt)
EBR-I	U.S.	1951–1963	1.0
EBR-II	U.S.	1964–1994	62.5
Fermi 1	U.S.	1963–1972	200
FFTF	U.S.	1980–1992	400
CRBRP	U.S.	Cancelled (1983)	975
Rapsodie	France	1967–1983	40
Phénix	France	1973–	563
Superphénix	France	1985–1997	3000
BR-5/BR-10	Russia	1958–2002	8
BOR-60	Russia	1968–	60
BN-350	Russia	1972–1999	1000
BN-600	Russia	1980–	1470
Jōyō	Japan	1982–	140
Monju	Japan	1980–1992	714
DFR	UK	1959–1977	72
PFR	UK	1974–1994	600
KNK-II	Germany	1972–1991	58
FBTR	India	1985–	42.5

Leakage in the steam generating system with resultant sodium-water reactions does not lead to serious safety problems. Such reactions are not catastrophic, as previously believed, and can be detected, contained, and isolated.

Leakage of high-temperature sodium coolant, leading to a sodium fire, is not catastrophic and the fire can be contained, suppressed, and extinguished. There have been no injuries from sodium leakage and fire (operation at near atmospheric pressure is an advantage to safety).

Fast reactors can be self-protecting against Anticipated Transients without Scram when fueled with metal fuel. Load-following is also straightforward.

Sodium-cooled reactors have demonstrated passive transition to natural convective core-cooling and passive rejection of decay heat.

Sodium-cooled reactors have demonstrated reliable control and safety-system response.

Operators of sodium-cooled reactors have demonstrated effective systems for purity control of sodium and cleanup.

Operators of sodium-cooled reactors have demonstrated efficient reprocessing of metal fuel, including remote fabrication.

Low radiation exposures (less than 10 percent of those typical for LWRs) are the norm for operating and plant maintenance personnel.

Emissions are quite low, in part because sodium reacts chemically with many fission products if fuel cladding is breached.

Maintenance and repair techniques are well developed and straightforward.

Electromagnetic pumps operate reliably.

Negatives:

Steam generators have not been reliable and are expensive to design and fabricate.

Sodium heat-transport systems have experienced a significant number of leaks because of poor quality control and difficulty with welds. Also, because of sodium's high thermal conductivity, many designs did not adequately anticipate the potential for high thermal stress on transients.

Many problems with handling fuel in sodium systems have occurred, primarily because of the inability to visually monitor operations.

Failure of in-sodium components without adequate means for removal and repair has resulted in costly and time-consuming recovery.

Sodium-cooled fast reactors have been more expensive than water-cooled-reactor systems.

Reactivity anomalies have occurred in a number of fast reactors, requiring careful attention to core restraint systems and potential for gas entrainment in sodium flowing through the core.

Operational problems have been encountered at the sodium/cover-gas interface, resulting from formation of sodium oxide that can lead to binding of rotating machinery and control-rod drives and contamination of the sodium coolant.

EBR-II Design Description: Keys for Success

EBR-II suffered few of the negatives and its designers and operators were able to develop technology that led to the success of plants that followed. The reason for this success was based on design choices, attention to details of construction, disciplined operation and maintenance, and an aggressive test program that developed a deeper understanding of the technology.

EBR-II was a complete power plant, producing 20 MW(e) with a conventional steam-turbine (with superheating). The reactor produced 62.5 MW(t).

Table C.4 EBR-II Operating Parameters

Power Output, thermal	62.5 MW(t)
Power Output, electric	20 MW(e)
Reactor Inlet Temperature	700°F
Reactor Outlet Temperature	883°F
Flow Rate Through Core	9,000 gpm
Volume of Primary Sodium	89,000 gal
Sodium Temperature to Superheaters	866°F
Sodium Temperature from Evaporators	588°F
Steam Temperature	820°F
Steam Pressure	1,250 psig
Feedwater Temperature	550 F
Fuel	Metal 63% enriched
Primary System Configuration	Piped Pool
Steam Generator Design	Duplex Tube

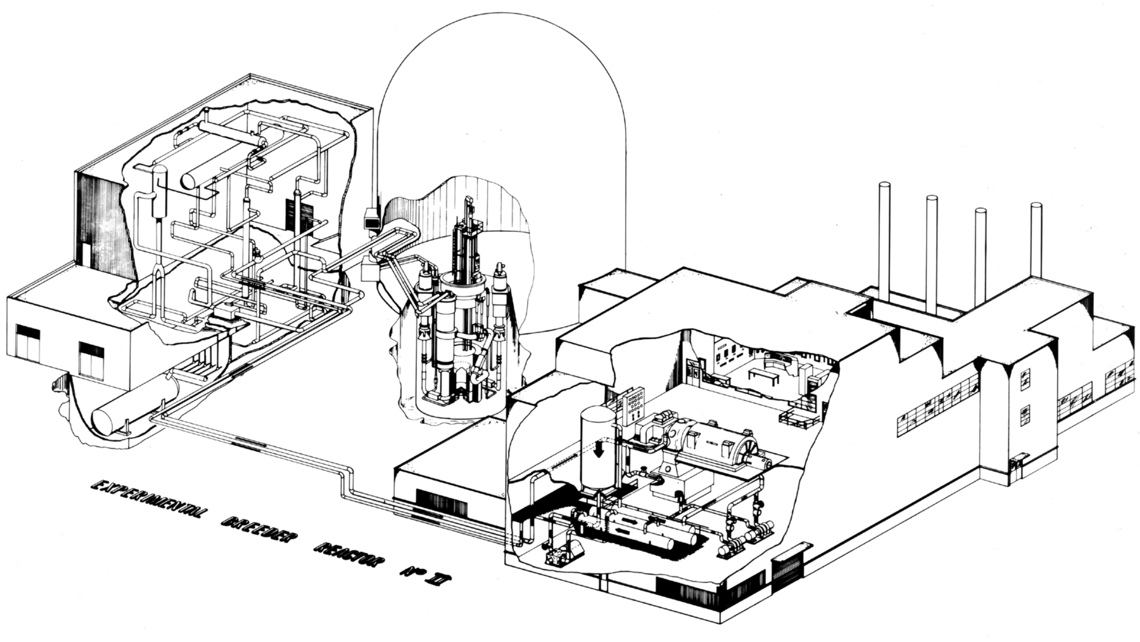

Figure C.18 EBR-II was a complete power plant

EBR-II was a sodium-cooled reactor with a piped-pool configuration. That is, two centrifugal pumps drew the coolant from a tank of sodium and then piped it to a plenum at the bottom of the core. After the sodium had flowed through the core, piping conducted it to an intermediate heat exchanger where it transferred its heat to the secondary sodium system. This configuration allowed for leakage at the connections at the outlet of the pumps and at the intermediate heat exchanger, because primary sodium would simply leak back to the tank from which it was drawn. This also allowed “ball and socket” connections at the pumps, which facilitated their removal and replacement. The tank, which was a right circular cylinder, was kept at a nearly uniform and constant temperature 371 degrees Celsius (C) (700 degrees Fahrenheit (F)), which limited thermal stress. Another important feature was that the tank included no penetrations through the wall; all penetrations were through the top. This also limited the risk for thermal stress, weld failure, and sodium leakage.

A guard tank surrounded the primary tank; an annulus between them allowed the detection of sodium leakage. The guard tank was in turn surrounded by concrete shielding, which acted as a final containment vessel. Were leakage to occur in both the primary and guard tanks, the core would not be uncovered and would be adequately cooled.

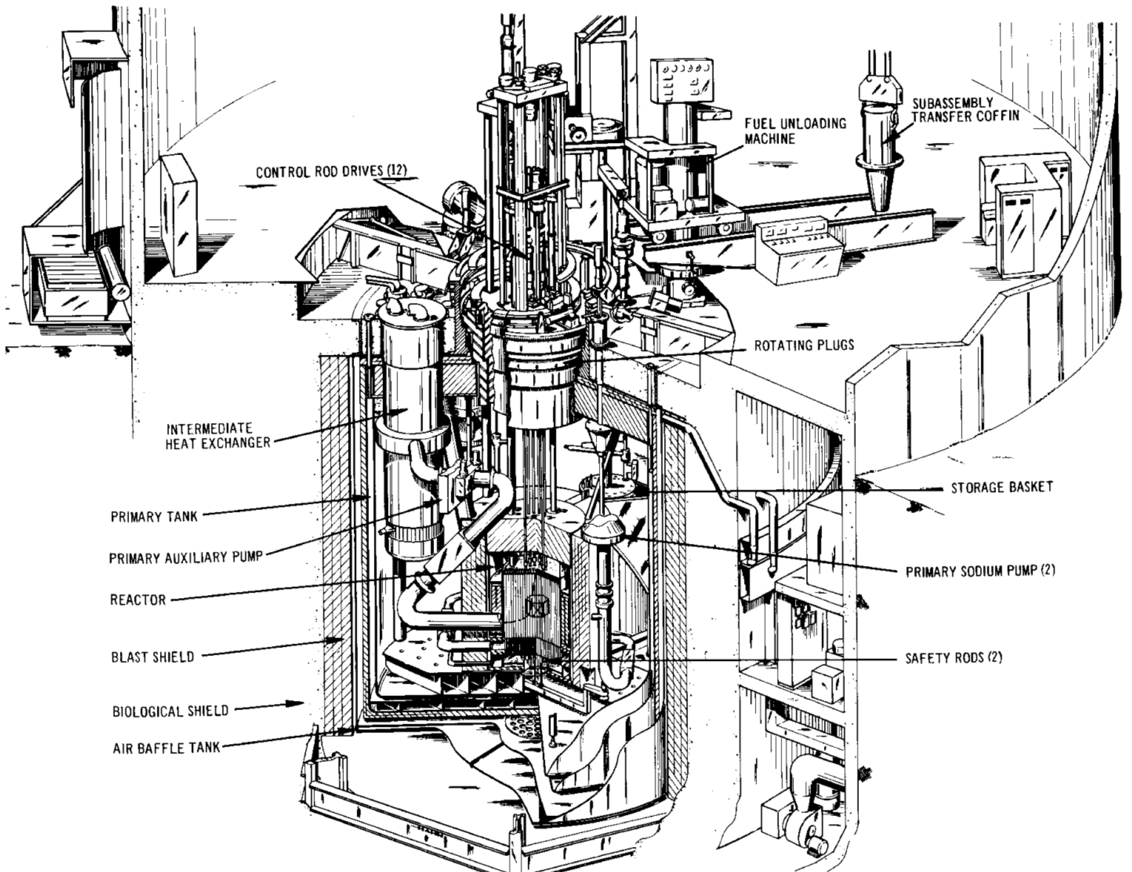

Figure C.19 Cutaway of the EBR-II Primary Tank

An inert gas (argon) filled the space between the tanks and their cover. Because there were penetrations though the cover for rotating machinery (pump shafts and fuel handling systems) and control rods, much attention was paid to seals to prevent ingress of oxygen which would result in formation of sodium oxide. Sodium oxide as a deposit on this equipment would cause binding of the machinery and contamination of the sodium coolant with particulate. Much attention was also paid to instrumentation for detection of oxygen ingress, and this remained a priority through the life of EBR-II operation.

Heat was removed from the primary sodium by three systems: (1) the secondary sodium loop which transferred heat to the steam generators, (2) thimbles immersed in the primary sodium and filled with sodium-potassium, which removed decay heat by natural convection, and (3) forced flow of air through the annulus between the primary tank and its guard tank, which also removed decay heat. Because decay heat removal did not depend on the secondary sodium loop, sodium in that loop could be drained to a storage tank for maintenance or in the event of a sodium leak. The secondary sodium loop was designed to ensure that a severe reaction between sodium and steam would not endanger the reactor. The steam generators themselves were double-walled (a tube in a tube) to minimize the potential for leakage. The tube sheets were configured with a plenum between the two tube sheets at each end, which provided a path for sodium or steam to travel if one of the tubes were to fail, facilitating detection.

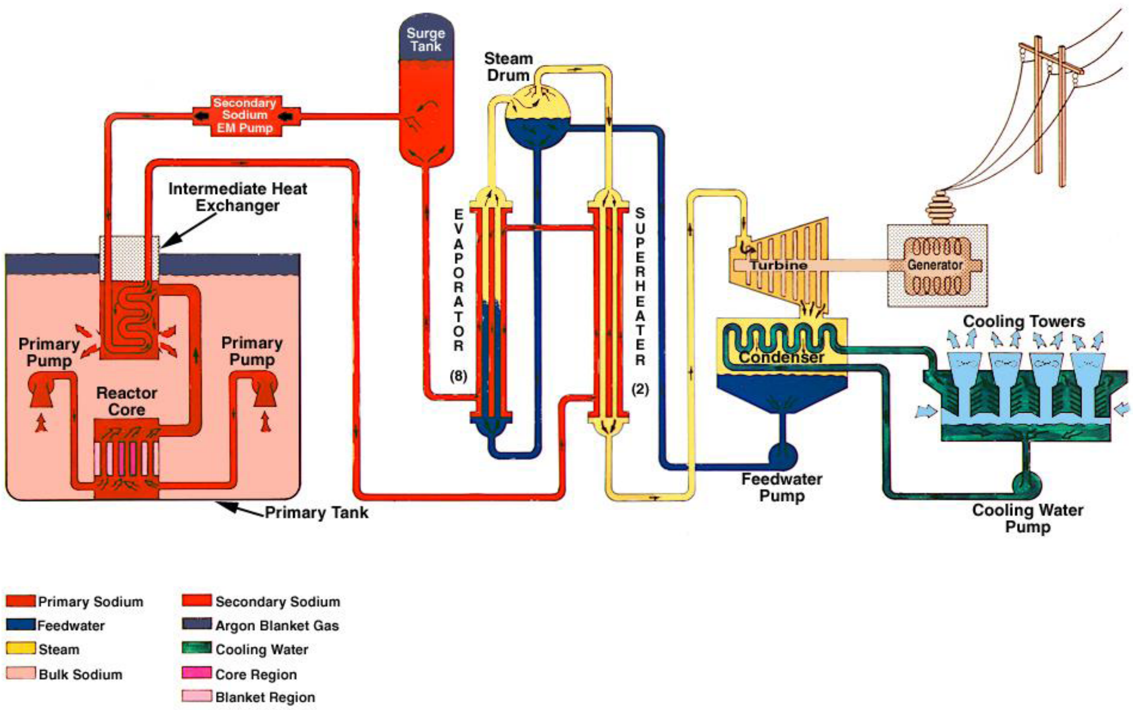

Figure C.20 EBR-II Heat Transfer Path

The EBR-II containment building was a domed cylinder designed to withstand a pressure of 24 psig. The design pressure was determined from analysis of a massive sodium fire, assuming that primary sodium was somehow sprayed as aerosol into containment (such an event is hypothetical). Interestingly, the extent of the fire is limited by available oxygen so a smaller containment is better, but the containment must be big enough to allow fuel handling, removal of major primary system components, and other activities. The result was a rather large containment building. It was a welded steel structure lined with concrete to provide the ability to withstand the high temperatures that would be associated with a sodium fire in the building. The building was pressurized slightly to ensure cleanliness of the atmosphere in the building and all exhaust went through HEPA filters. Periodic pressure tests were conducted to verify leak-tightness.

One of the more distinctive aspects of the EBR-II containment was a fuel transfer tunnel that attached the building to an adjacent Fuel Cycle Facility (FCF). Spent fuel from EBR-II was transferred from the holding basket in the primary tank to a shielded cask which was then lowered through a hatch to the tunnel. The cask was then moved through this tunnel on rails, after which it was mated with a transfer hatch at the FCF. Many thousands of transfers of fuel were made in this manner, and during the first five years of operation, ~35,000 reprocessed fuel pins were returned to the reactor.

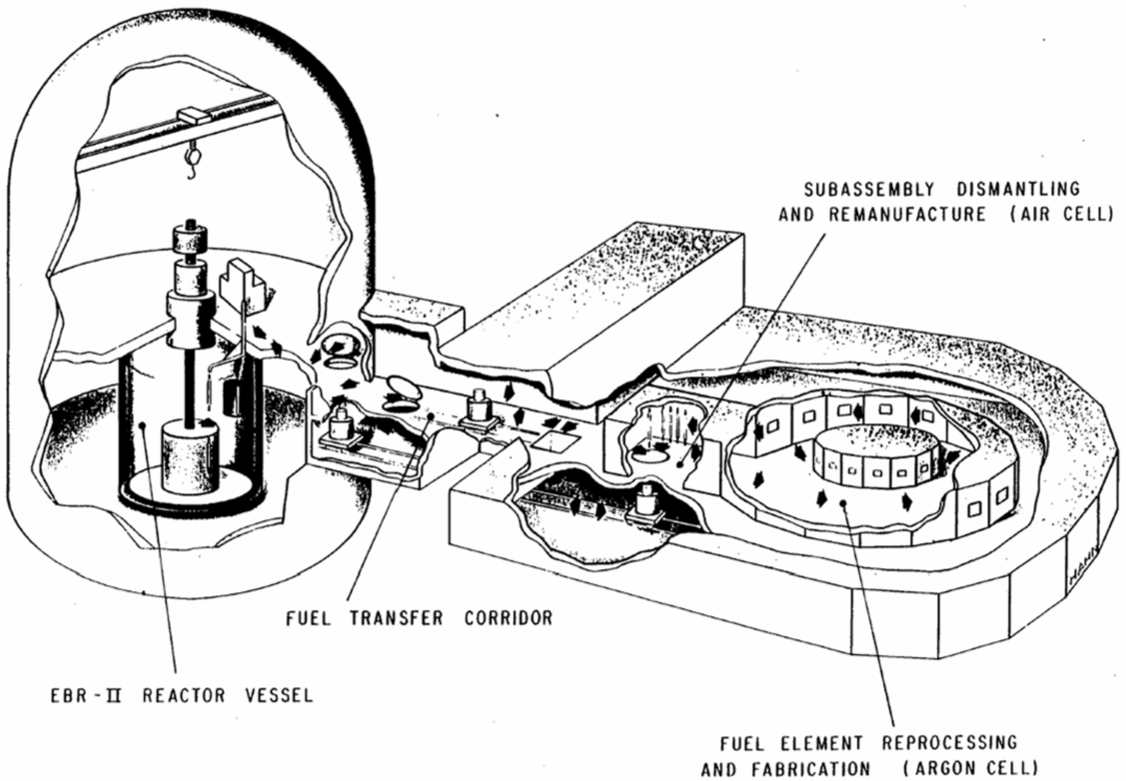

Figure C.21 Layout of the EBR-II Containment

To facilitate fuel handling, a fuel-storage basket capable of storing a large fraction of the core was placed in the primary tank. Because the containment building and the primary tank of sodium were accessible during reactor operation, the storage basket would be preloaded with fresh fuel, which would be exchanged for spent fuel when the reactor was shut down for refueling. This greatly facilitated core unloading/loading, which typically took 3 days. Spent fuel could then be transferred on a schedule determined by cooling requirements for decay heat generation.

The fuel-handling system was one of the more complicated features of EBR-II design. Because fuel handling is done in the blind (because sodium is opaque), the equipment had to be precise with many checks and interlocks to ensure that transfers were being made properly. The main fuel assembly gripper was a straight pull through a penetration in a rotating plug in the top of the primary tank cover. This plug was one of two, placed within a larger plug at an eccentric position which allowed positioning of the gripper over any core location. The control rods also penetrated this smaller rotating plug, which required that they be disconnected and withdrawn before rotation of the plugs. When a fuel assembly was withdrawn by the gripper, it was captured by a transfer arm which positioned the assembly above a desired location in the fuel-storage basket. The basket could be rotated, and then raised to accept the assembly, which was then detached from the transfer arm. The systems worked well, with a few exceptions as discussed in later in this report.

Sealing of penetrations to the atmosphere above the primary tank was given special attention in the EBR-II design. The large rotating plug in particular represented a special challenge. To provide a seal while the plug was rotated, a dip ring was immersed in an alloy of tin-bismuth, which was heated until molten for rotation of the plug. When the plug was secured, the alloy was cooled, sealing the interface. This arrangement created many problems for operation and maintenance. Frequent manual cleaning of this trough was necessary to avoid sticking the rotating plug, which would have created serious problems for recovery.

The intermediate heat exchanger (sodium to sodium) was a conventional shell and tube design, with primary sodium flowing to the shell at the top and exiting at the bottom while the secondary sodium flowed in tubes from the bottom. An electromagnetic pump was immersed in the primary sodium, providing forced flow for a smooth transition from forced to convective primary flow in event of a loss of power to the primary pumps (a feature later determined to be unnecessary).

Purification of the sodium was accomplished by in-line cold traps which cooled sodium to the point that sodium oxide would solidify and collect on stainless-steel wire mesh. Later, special graphite traps were added to clean the sodium of cesium, a fission product associated with extensive run-beyond-cladding breach testing in EBR-II. Both systems worked well.

Sodium leak detectors were installed throughout the plant and were of two main types: smoke detectors and “spark plug”-type detectors that would sense the presence of liquid sodium. In the steam generator building, acoustic monitors and hydrogen detectors were installed to detect a sodium-water reaction. In the event of a sodium-water reaction, blowout diaphragms and panels were installed to relieve pressure away from the reactor building, and fast actuating valves would dump the secondary sodium to a storage tank.

Control of the reactor required two operators, one controlling sodium flow in the secondary system (to maintain a constant reactor inlet temperature) and the second operator controlling reactor power through control-rod movement. Primary coolant flow was held constant. (More on this later; it was found that the reactor would load follow easily, responding through reactivity feedback as inlet coolant temperature changed in response to changes in power demand. No operator action is required in such a case.)

EBR-II Operating History

EBR-II was extremely successful as a test reactor; arguably the most successful ever as measured by the scope of what was accomplished. The test programs successfully addressed issues of safety, operability, maintainability, security and sustainability. Although EBR-II operation was not without problems, major problems which occurred in other fast reactor systems were successfully addressed or avoided at EBR-II.

Early EBR-II Milestones

- Site Preparation Begins 5/1957
- All Construction and Component Installation Complete 12/1962
- Primary System Filled with Sodium 2/1963
- Approach to Power Begins 7/1964
- Reactor Operated at 30 MW(e), T-G Synchronized with Site Loop 8/1964
- First Spent Fuel Reprocessed in FCF 9/1964
- Completed Demonstration of Fuel Cycle Closure Approximately 9/1969

- 35,000 Fuel Pins Recycled Back into EBR-II
- Reactor Power Increased to 62.5 MW(t) 9/1969

EBR-II was constructed and operated at the ANL-W site in Idaho. An important feature of this site was that all of the nuclear facilities needed for fast reactor development were co-located there, which created a synergism between testing programs and expertise that greatly benefited all. Besides the EBR-II reactor, there was the Fuel Cycle Facility which reprocessed EBR-II fuel, the Transient Reactor Test Facility (TREAT), which subjected fuel to severe overpower transients as part of an extensive safety testing program, the Zero Power Physics Reactor (ZPPR), a large critical facility to mock up fast reactor cores and conduct important physics measurements, the Hot Fuel Examination Facility (HFEF) for post-irradiation examination of fuel and materials, the Fuel Manufacturing Facility (FMF) for production of EBR-II fuel, the Sodium Components Maintenance Shop (SCMS) for repair and maintenance of reactor components operating in sodium, and the Sodium Process Facility (SPF) used to produce sodium hydroxide as part of EBR-II decommissioning.

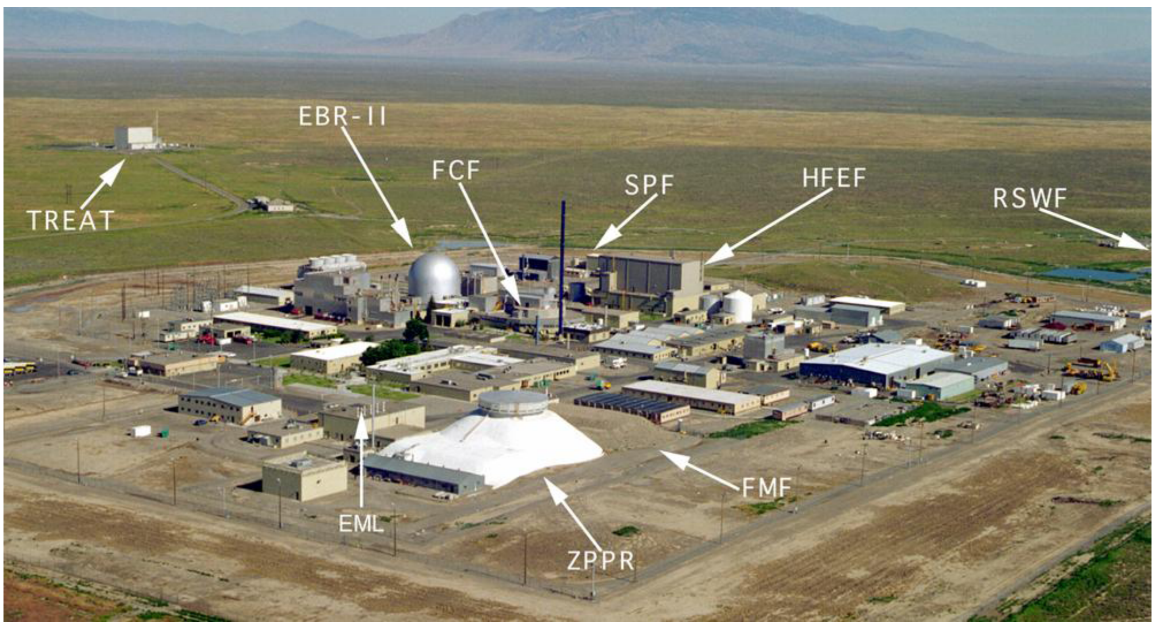

Figure C.22 Argonne National Laboratory West; now INL Material and Fuels Complex

The mission of EBR-II went through four distinct phases. The first was as a complete power plant with co-located fuel reprocessing. The second was as an irradiation facility, testing fuels for later fast reactors, primarily oxide fuel for the FFTF and the CRBRP. The third was as an operational safety testing facility, subjecting fuel and the plant to off-normal conditions such as operation of fuel with breached cladding (and ultimately, in the reactor’s inherent-safety demonstration tests, to anticipated transients without scram). The fourth was as the Integral Fast Reactor prototype, including demonstration of new reprocessing and recycling technology. (Decommissioning could actually be considered a fifth phase that yielded important information about the technology of sodium processing for disposal.)

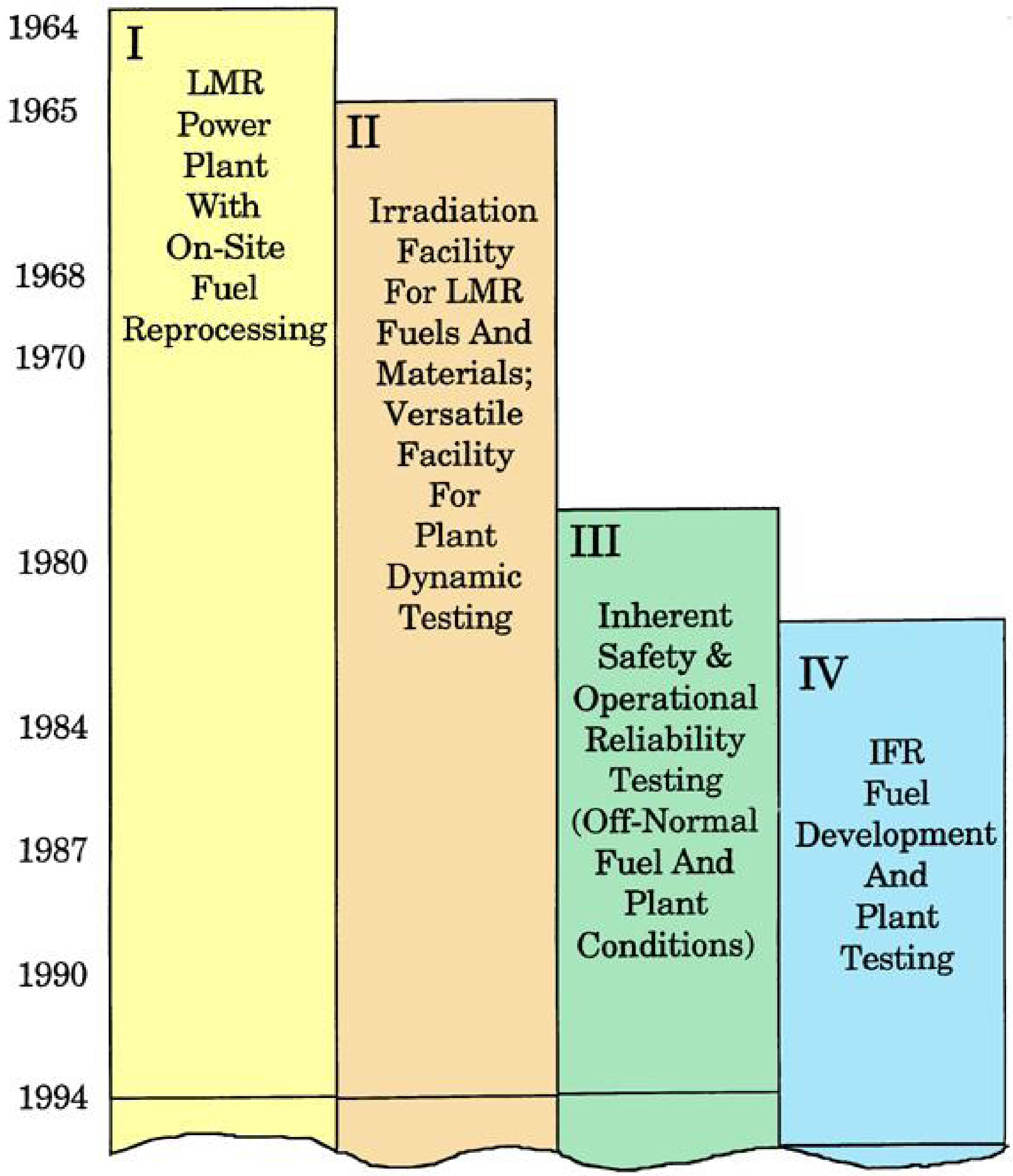

Figure C.23 EBR-II missions over 30 years of operation

Mission I, Power Plant Operation

The power plant operated reliably for 30 years. Capacity factors approached 80 percent even with an aggressive testing program. Maintenance techniques were proven, with personnel exposure to radiation less than 10 percent of that for a comparable Light-Water-Cooled Reactor (LWR). Effective sodium management was demonstrated, including successful suppression of a fire from a major sodium leak early in EBR-II's operation. The steam generators operated quite well, with no failures or leaks in the systems, a testament to the duplex-tube design.

- achieved high plant capacity factors
  - capacity factors approached 80 percent even with an aggressive testing program
- proved maintenance techniques
  - very low exposure for personnel, excellent safety record
- demonstrated sodium management
  - sodium leaks well managed
- demonstrated fuel reprocessing
  - 35,000 fuel pins reprocessed

Fuel reprocessing was also very successful, with over 35,000 fuel pins reprocessed and recycled to the reactor in the first five years of operation. This demonstrated the viability of remote casting of metallic fuel elements and non-aqueous reprocessing of spent fuel using a simple melt refining process.

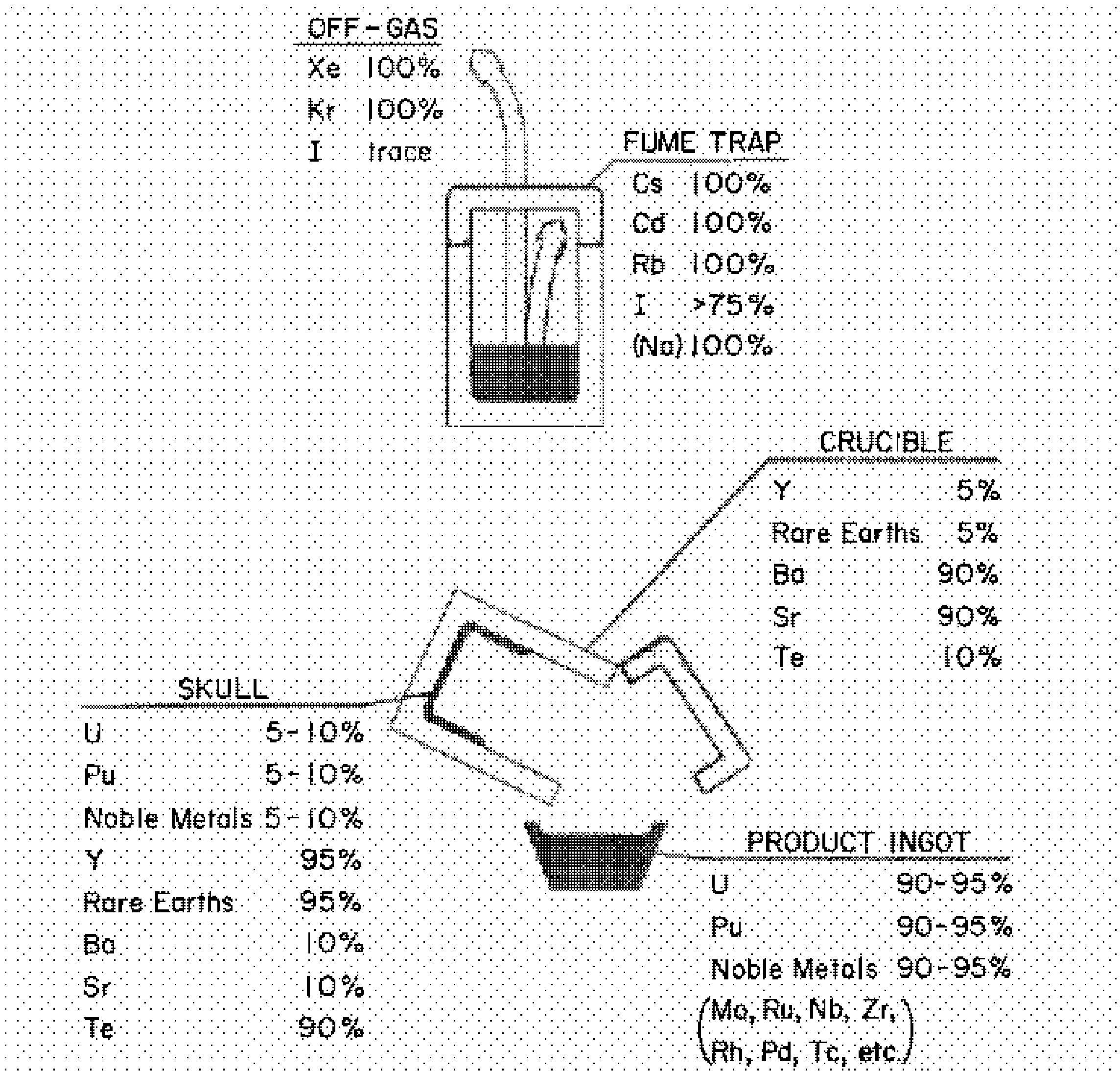

Figure C.24 Melt refining process

Several key features of design and characteristics of the system contributed to the excellent performance of EBR-II as a power plant. The first was the sodium coolant. Sodium is compatible with the reactor materials in the primary circuit, with no corrosion found after 35 years at temperature. Sodium also has a high boiling point (greater than 893 degrees C (1,640 degrees F) at atmospheric pressure) that allows the primary and secondary systems to be low-pressure. Consequently, there was no potential for high-pressure ejection of coolant. This feature is important for maintenance activities and is a major reason that there were no injuries from sodium leakage over the course of EBR-II operation. Already mentioned was the low exposure experienced by maintenance personnel.

The pool-type primary system also provided distinct advantages. Its large thermal capacity limited the severity of thermal transients and therefore stress on the primary tank and components submerged within it. The piped pool configuration allowed the majority of the primary sodium to be at reactor inlet temperature, further increasing the capacity of the sodium to absorb heat in the event of an upset. All primary system components were submerged in this relatively cold pool of sodium, which proved to be beneficial for their operating reliability and ease of removal for maintenance or repair. The pool-type primary system also minimized the potential for leakage of primary sodium, because all penetrations were through the top of the vessel. The only leakage encountered was in smaller systems, such as sodium sampling and purification, which were outside the primary tank and contained small inventories of sodium.

Major Incidents in EBR-II Operation

Early in operation of EBR-II (1968), a major sodium leak occurred in the secondary sodium system. Nearly 100 gallons of hot sodium spilled to the floor in the secondary sodium “control” room where sodium was sampled and purified. Repairs were being made to a bellows-seal isolation valve in the secondary-sodium plugging loop, during which personnel would freeze the sodium in the line, cut out a section, and then re-weld the section into the original line. Unfortunately, the frozen sodium plug did not extend far enough beyond the removed section, and when it was welded into the pipe, the sodium melted and spilled to the floor. A major fire erupted but was contained and extinguished by application of Metalex (a mixture of salts which cover the burning sodium and starve the fire of oxygen). Cleanup was accomplished in 13 days and there were no injuries. Firefighting techniques were found to be effective. Maintenance procedures were changed and no further incidents of this type occurred. (Freezing sodium in a line, cutting out a section for repair and re-welding it was a common practice through the life of EBR-II. Such operations were conducted on small piping associated with sampling and purification systems. Large pipes, such as for the secondary sodium systems, were drained before work maintenance was conducted.)

The second major incident was damage to a fuel assembly during fuel handling in April 1978, which bent the assembly so that it could not be removed from the fuel-storage basket. Fuel handling in sodium must be done without visual reference and all operations are done remotely. When an attempt was made to engage the assembly upper adaptor with the fuel-handling arm as part of the procedure to remove it from the storage basket, it was found that the upper adaptor was out of position and could not be engaged. A technique was developed for profiling the assembly by mechanical means, using the fuel-handling equipment to characterize its position and configuration. Following this work, a mock-up of the storage basket, the deformed assembly and the fuel-transfer system was constructed to develop the tools and procedures for removal of the assembly. Removal was accomplished in May 1979 using a specially designed shaft and gripper that penetrated one of the nozzles in the cover of the primary tank. Reactor operation was not impacted and fuel handling from the storage tank proceeded normally for other assemblies located within it. The techniques developed and experience gained proved to be valuable for fuel-handling system design and beneficial for the second incident associated with fuel handling at EBR-II. It was found that the damage occurred because the assembly had not been fully seated in the storage basket and the assembly contacted the lower shield plug of the primary tank cover when the storage basket was raised.

On November 29, 1982, a fuel assembly was dropped over the EBR-II core as it was being transferred from the fuel-storage tank. The incident was discovered when no assembly was present for the exchange between the transfer arm and the core fuel assembly gripper. Extensive checks were made to verify that the assembly was not located in the storage basket or the transfer arm and then a search began to determine its location. The assembly had been dropped somewhere between the storage basket and its intended location in the core.

Care had been taken in design to provide extensive interlocks to ensure that movement of fuel-handling equipment did not begin until assemblies were securely gripped, and manual operation of the transfer operation was such that checks could be made manually. However, in this instance the assembly had become disengaged from the transfer arm and fallen. It was found that the transfer arm and storage basket were misaligned, preventing the assembly upper adapter to be fully seated and locked before transfer.

Mechanical probes were used to locate the assembly and precisely identify its position. As before, a full-scale mockup was constructed and tools and procedures were developed to retrieve the assembly. The major retrieval tool was a stainless-steel cable extending as a loop beyond a stainless-steel tube which penetrated the top cover. (A number of spare nozzles had been provided through the cover in the original design, a decision which proved to be very valuable). The loop was maneuvered into position manually and the noose pulled tight, snagging the assembly upper adaptor so it could be retrieved. (This process was aided not only by the ability of the operator to feel resistance but also by acoustic monitors installed in the tank which detected the sound from contact with equipment). The assembly was then moved to a position where it hung from the noose and could be engaged by the transfer arm; it was handled normally from that point. The total operation took less than a month but, in this case, did require the reactor to be shut down. However, advantage was taken of the down time to conduct preventative maintenance normally scheduled for the spring shutdown, so the overall impact on reactor operation was minimized.

Over 40,000 fuel assembly transfers were made without incident in the 30 years of operation of EBR-II, so these incidents were certainly rare. However, mishaps during fuel handling can have a significant impact on reactor operation and every reasonable precaution needs to be taken to prevent them. Besides robust fuel-handling systems and extensive interlocks, the EBR-II experience demonstrated the importance of operator tactile feel and acoustic monitoring for operation of the equipment. Much of the success of the EBR-II fuel-handling experience, for example, resulted from the fact that motion of the rotating transfer arm was manual, allowing the operator to verify through a “wiggle” test that the arm had successfully engaged the assembly before it was released by the core gripper. Under-sodium viewing technology is now available as another guard against fuel-handling errors.

Another lesson learned from EBR-II operation was the importance of anticipating problems and providing design features to accommodate them. For example, in anticipation of an assembly falling from the transfer arm after it had cleared the core, a catch basket was provided that would funnel the assembly to a position where it could be easily retrieved. Spare nozzles had also been provided to support special operations in the primary tank. Of note, each of the primary pumps was removed for maintenance twice during the course of EBR-II operation, facilitated by designs and equipment that anticipated the need.

Mission II: Fast Reactor Fuel Development

After the initial demonstration of EBR-II as a complete power plant, the reactor core was reconfigured to enhance its capability as an irradiation test facility for fuels development. The inner blanket surrounding the core was replaced with a stainless-steel reflector which increased the flux levels and provided a smaller flux gradient across the core. The irradiation testing mission was directed primarily at development of oxide fuel for FFTF and CRBR, but it was also important to improve the performance of the EBR-II metal fuel. In addition, nitride and carbide fuels were tested, but not to the degree that oxide fuels were developed.

EBR-II metal driver fuel was significantly improved over the course of the 30-year operating life of the reactor. Early in the development of metal fuel, failures in the cladding were seen at burnups as low as 1 percent. The reason was that the fuel would swell against the cladding, exerting enough force to cause it to fail. The solution was quite simple; the gap was increased between the fuel pin and the cladding which allowed the fuel to swell until the fission gas was released from the pin, stopping the swelling. Fission gas was released at about 2 atom percent burnup when the gas bubbles in the fuel would interconnect, creating a porous fuel structure that allowed the fission gas to be released. This interconnected porosity would then backfill with sodium, further enhancing the thermal conductivity of the fuel and resulting in very low fuel-centerline temperatures. The second modification to the fuel pin design was to increase the gas-plenum volume to accommodate the fission gas that was released. With these design changes, burnups of in excess of 20 atom percent were achieved. In fact, the limit was not reached, but it is likely to be related to filling the fuel pores with solid fission products to a degree sufficient to again initiate fuel swelling.

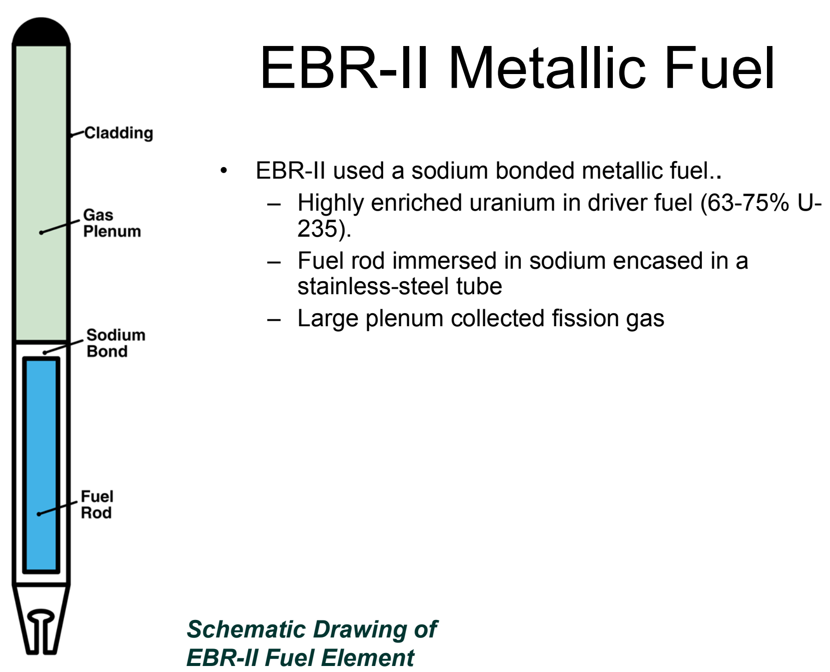

Figure C.25 EBR-II Fuel Element

Perhaps one of the most important aspects of metal fuel was its ease of fabrication. Metal fuel pins were produced in 100-pin lots by simple injection casting. Glass molds were lowered into molten fuel and then the system was pressurized, forcing fuel into the molds. The molds were removed, allowed to cool, and removed, after which the pins were cut to length. They were then placed into the cladding tubes which contained a small amount of molten sodium as a thermal bond and a cap was then welded to close the tube. This process produced ~150,000 fuel pins and was carried out both at the EBR-II Fuel Manufacturing Facility (FMF) and by commercial vendors. It was also accomplished remotely and because of its simplicity was done without difficulty.

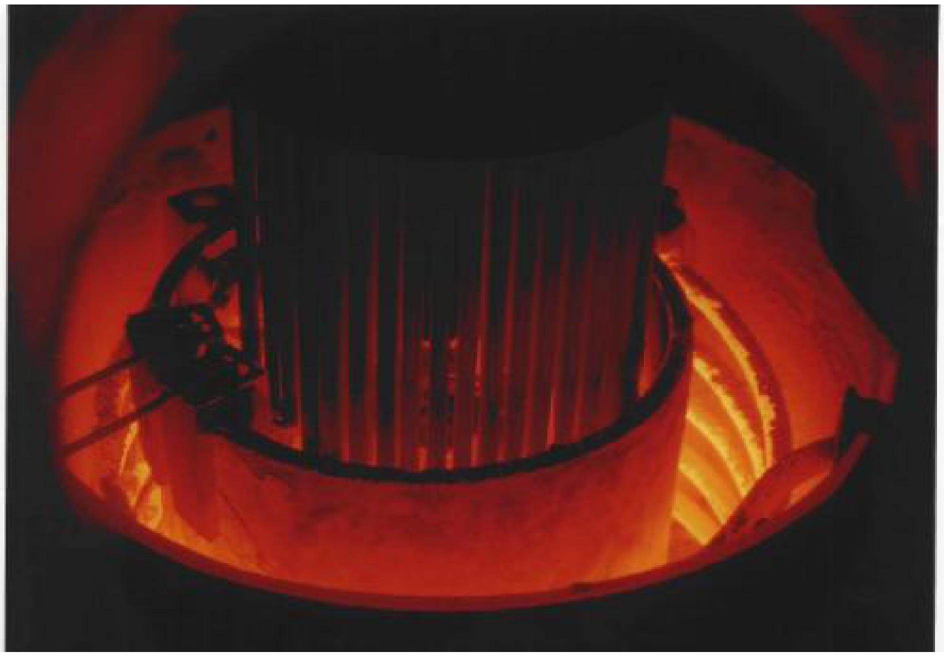

Figure C.26 EBR-II fuel casting furnace

Figure C.27 Metal fuel after casting

In addition to metal fuel development for EBR-II, eight full-sized assemblies (1800 pins) of metal fuel were irradiated to high burnup in FFTF without failure. This work was done as part of a plan to convert the FFTF core from oxide fuel to metal, but the reactor was shut down before the conversion could be accomplished. Those assemblies have been returned to the HFEF at the INL where they are available for examination.

Minor Actinide Fuel Has Been Fabricated and Irradiated

- Three full-length pins containing minor actinides were successfully fabricated and irradiated to 6 percent burnup.
- As-fabricated composition was: 68.2 percent U, 20.2 percent Pu, 9.1 percent Zr, 1.2 percent Am, and 1.3 percent Np.
- Approximately 40 percent of the initial Am was lost during casting, primarily because of volatile impurities of Pu-Am feedstock (3 atom percent Ca and 2,000 ppm Mg).
- Judicious selection of the cover gas pressure during the melt preparation and the mold vacuum level during casting is expected to reduce the Am loss by approximately 200 times.

A full range of metal fuel compositions was tested, including uranium-zirconium and uranium-zirconium-plutonium mixtures, with and without additions of minor actinides. Peak cladding temperatures reached 620 degrees C with maximum in-reactor exposures of 5 years. An important conclusion is that the metal is a versatile and “forgiving” fuel design, able to accommodate a wide range of compositions.

Excellent Steady-State Irradiation Performance

- Over 40,000 EBR-II Mark-II (75 percent smear density U-Fs) driver fuel pins were successfully irradiated through the early 1980s.
- When the IFR Program was initiated in 1984, 10 percent Zr replaced 5 percent fissium, and a total of 16,800 U-Zr and 660 U-Pu-Zr fuel pins were irradiated in the next 10 years. U-Pu-Zr fuel reached peak burnup of approximately 20 percent.
- In addition, eight full metal fuel assemblies were irradiated in FFTF. The lead test achieved peak burnup of 16 percent. One assembly contained U-Pu-Zr, which achieved peak burnup of 10 percent.

As noted earlier, oxide fuel was also demonstrated to be viable, operating to high burnup and achieving the smear densities and power ratings desired. (Details of oxide fuel development and experience will be given by other authors.) The major difference is approach to reactor safety. Metal fuel provides a large degree of self-protection in response to off-normal events; oxide fuel does not, as explained in the following discussion.

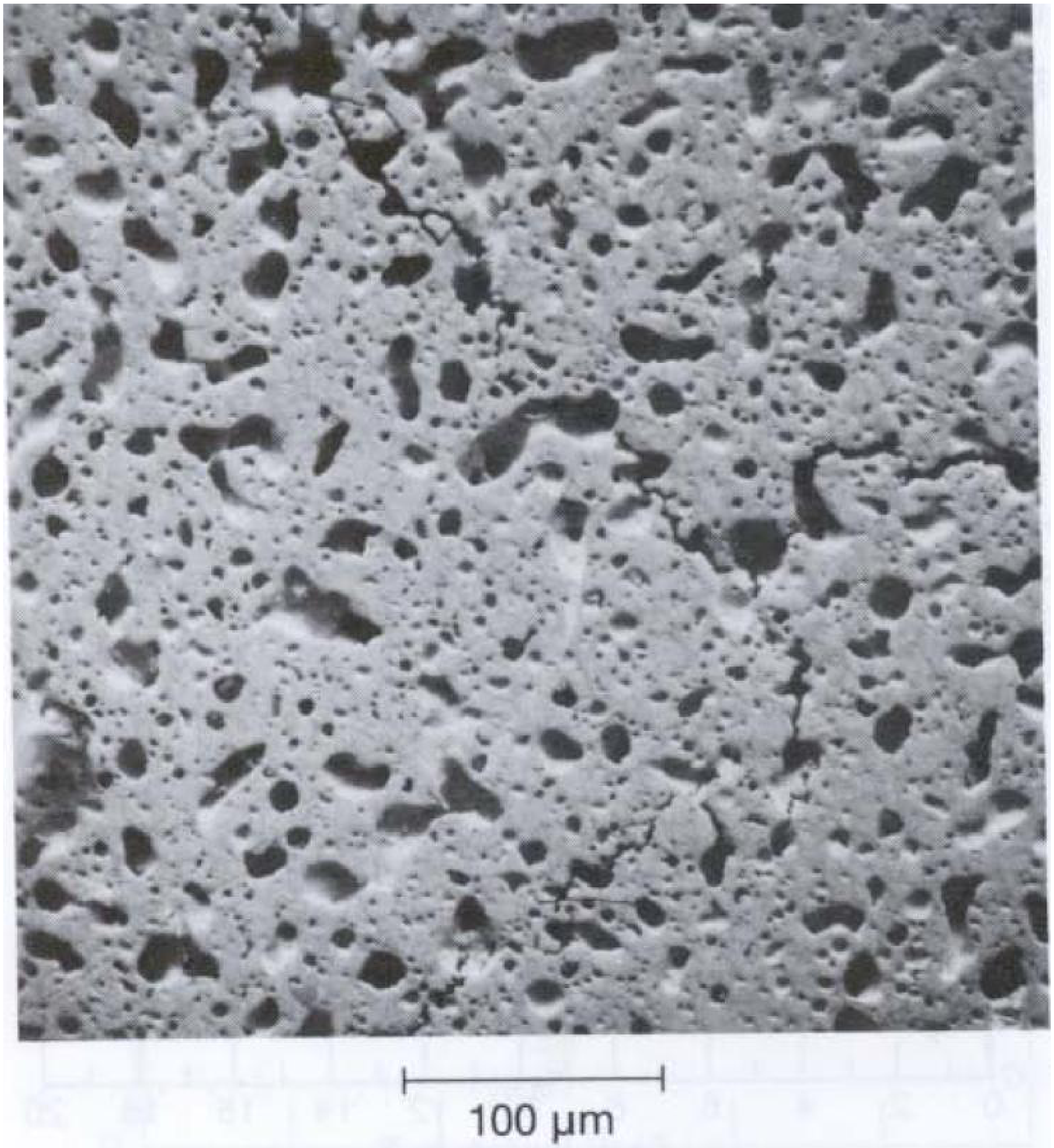

Figure C.28 Interconnected pore morphology of irradiated U-10Zr metal fuel

Mission III: Operation Safety Testing

When the FFTF began operation, taking on a major role in irradiation-testing of fuels and materials, EBR-II was able to conduct more aggressive operational-safety tests. These involved integral plant-safety tests as well as fuel-safety tests. The interest for fuel was its performance with breached cladding under both steady-state and transient overpower conditions.

A particular concern for oxide fuel is formation of sodium oxide as a reaction product with the sodium once fuel is exposed to the coolant. Sodium oxide is less dense than the fuel and can tend to split the cladding, causing progressive failure. The EBR-II program of run-beyond-clad breach testing supported the safety case for oxide fuel for both the Monju reactor in Japan and the CRBRP reactor in the United States. The testing was extensive and included operational transients in EBR-II as well as more aggressive tests in TREAT. The result of this work was data which demonstrated the safety of continued operation of oxide fuel with breached cladding, forming the safety basis for Monju.

The question also arose about the performance of metal fuel with breached cladding, because testing of oxide fuel in the reactor would mask failure of cladding for metal fuel. The EBR-II driver fuel had to be qualified to operate safely for extended periods with breached cladding. Metal fuel has an advantage in that it is chemically compatible with the sodium. (Sodium is used in the fuel pin to enhance thermal conductivity between the metal fuel and the cladding.) Extensive tests, including both steady-state and transient overpower conditions, demonstrated that metal fuel was completely compatible with the sodium coolant and that a breach in cladding would not grow. The safety case was made that breached cladding in metal fuel could be safely accommodated; no fuel loss would be expected.

EBR-II was modified to accommodate fission-gas release by installing the cover-gas cleanup system which captured the noble gases Xe and Kr. (Chemically active fission products, like Cs and I were captured in the sodium and subsequently cleaned by the sodium-cleanup systems.) The cover-gas cleanup system used cryogenic cooling to capture these gases in an activated charcoal bed, working very well over the remaining life of the plant.

The most dramatic safety tests were those involving the whole plant, leading to the inherent safety demonstration tests conducted in April 1986. The EBR-II plant was subjected to all of the Anticipated Transients without Scram (ATWS) events without damage, demonstrating the self-protecting characteristics of a metal-fueled fast reactor.

The first of these was loss of all pumping power with failure to scram, simulating a station blackout with failure to scram. The reactor was brought to 100 percent power and the pumps were turned off, allowing them to coast down and coolant flow to transition from forced to natural-convective flow. Testing and analysis over the previous 4 years had been conducted to accurately model the reactor for this event, and the system responded as expected. Special in-core temperature monitoring had been provided as a safety system to scram the reactor if temperatures rose to unexpectedly high levels, but they did not.

Temperatures initially rose rapidly as the cooling flow decayed, but the increase in temperature introduced sufficient negative reactivity feedback that the power was also reduced rapidly, resulting in peak core coolant temperatures that were higher than for normal operation (approximately 704 degrees C (1,300 degrees F) vs. 477 degrees C (890 degrees F) at normal operation) but not high enough to damage the fuel. There was also significant margin to sodium boiling temperature, which would occur at approximately 893 degrees C (1,640 degrees F). The system power came down rapidly, reducing peak core temperatures until they equilibrated at an average temperature very close to that of normal operation.

A point to be emphasized is that no fuel or core damage occurred during this event, unlike what would occur in a conventional reactor system. In fact, this was the 45th test of ATWS events on this core and the reactor was restarted for a subsequent test that same afternoon.

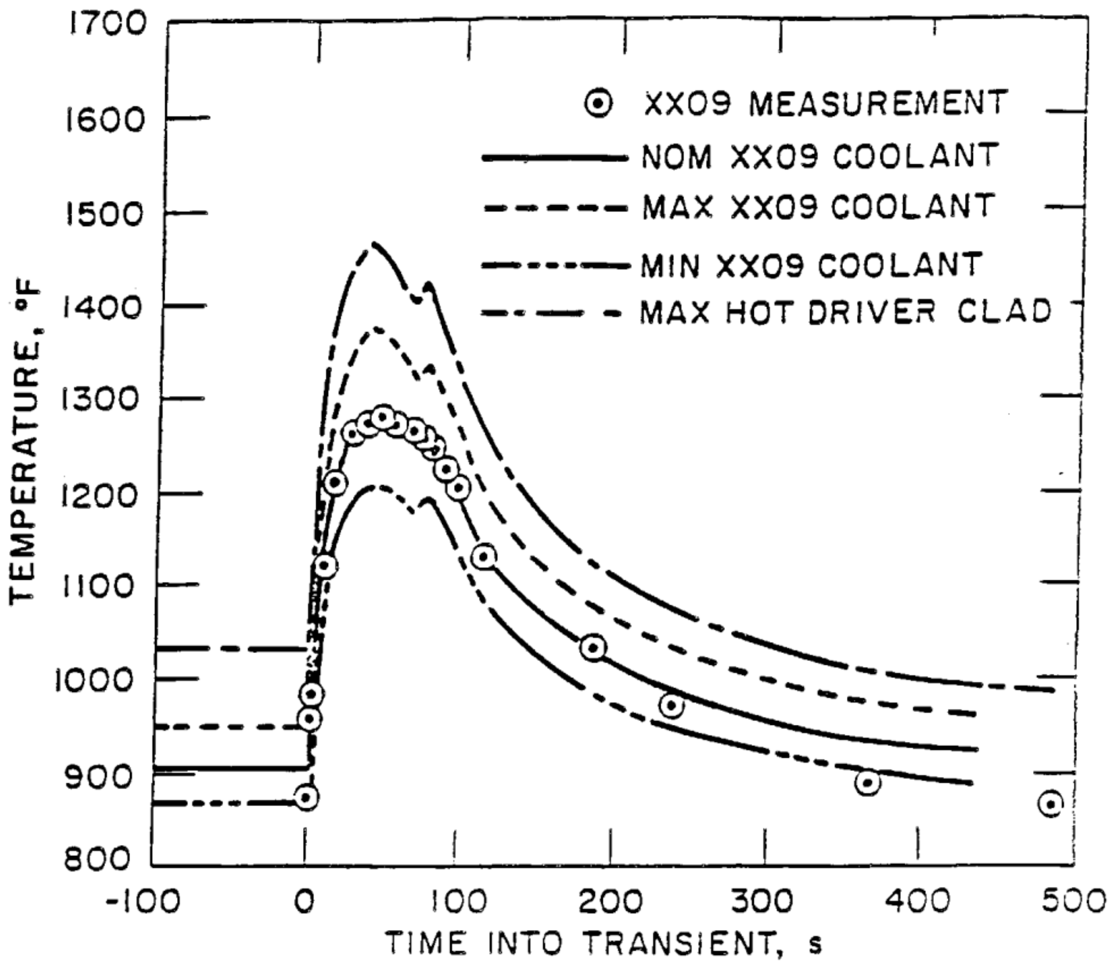

Figure C.29 TEST 45, loss of flow without scram from 100 percent power

Key Contributors to Inherent Passive Safety

- Has large margin to sodium boiling temperature.
- Pool design provides thermal inertia.
- Has low stored Doppler reactivity because of high thermal conductivity (hence, low temperature) of metal fuel.
- Hence, the inherent passive safety characteristics are achieved only in the IFR-type fast reactors.

Key Steps in Test

- establish 100 percent power
- insert special SCRAM protection for the test
- bypass loss-of-flow SCRAMs
- turn off the pumps

The second test subjected the reactor to loss of heat sink without scram. The reactor was brought to 100 percent power and flow was stopped in the secondary heat transfer system, blocking the transfer of heat to the steam generators. As the reactor inlet temperature rose, negative reactivity feedback reduced power to the point that the temperature difference across the core was reduced; peak coolant temperature never increased. The reactor temperatures equilibrated at an average temperature very close to that of normal operation.

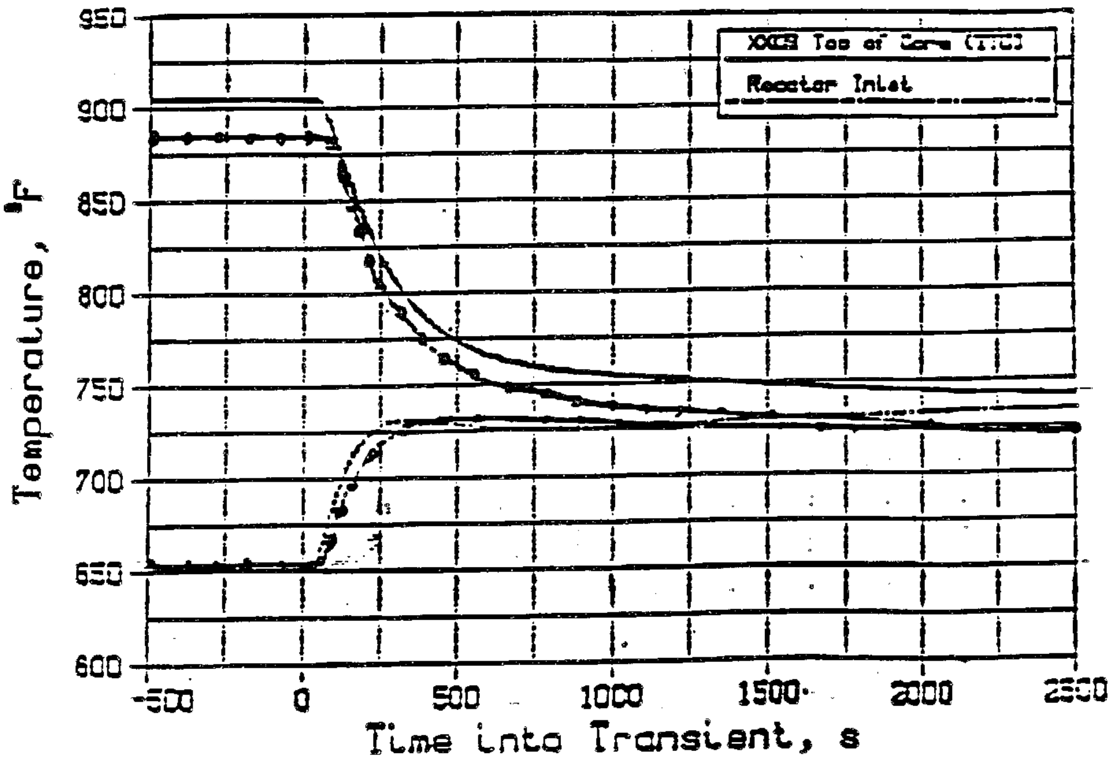

Figure C.30 Loss of heat sink without scram from 100 percent power

Key Steps in the Test

- Establish 100 percent power.
- Stop all flow in the intermediate sodium loop.
- Monitor the passive power reduction and the leveling of tank temperature.

This behavior results because of the very strong negative feedback associated with neutron leakage as the coolant temperature increases and the lack of a strong positive reactivity feedback from Doppler effects as the fuel centerline temperature falls. Metal fuel operates with a very low centerline temperature and therefore little Doppler feedback reactivity. For events involving loss of cooling or loss of a heat sink, coolant temperature rises, power falls, and—if one has a high Doppler coefficient of reactivity in the system, as with oxide fuel—the positive reactivity feedback will delay power reduction, with the result that the sodium will boil. Boiling sodium will insert significant positive reactivity, likely leading to a severe overpower transient. For this reason, a metal fuel core is self-protecting against undercooling events while an oxide fuel core is not.

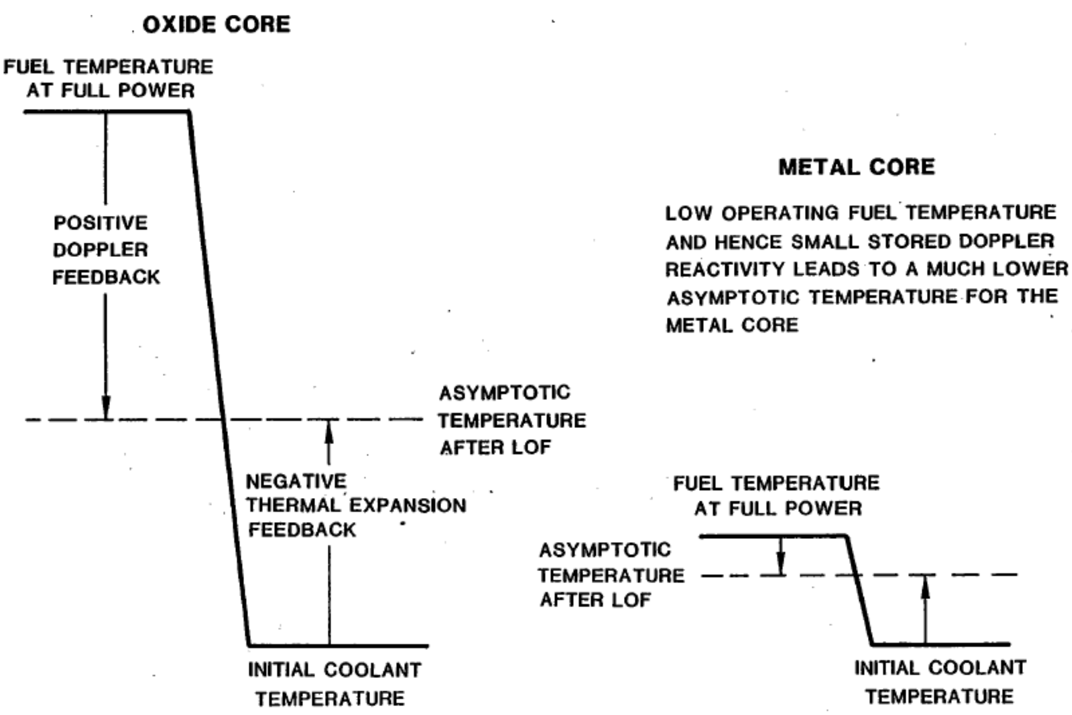

Figure C.31 Oxide versus Metal Core Reactivity Feedback Comparison

It was earlier thought that a high Doppler coefficient of reactivity was important to protect against severe overpower events, so tests were done in TREAT with metal fuel to determine its performance under such conditions. Many metal fuel pins from EBR-II were subjected to severe overpower events which took the pins to failure. It was found that the relatively low melting temperature of the fuel was important, because it softened and then flowed like toothpaste in a tube before breaching the cladding. This flow of fuel occurred rapidly and would be effective in introducing large negative feedback during severe transients, acting as an effective self-protecting mechanism. Also, metal fuel cladding failures typically occurred at 4 times nominal power, higher power than typical for oxide fuel (which typically failed at 3 times nominal power).

Further tests were conducted to determine the load-following characteristics of the reactor, which are very good. Metal fuel is not adversely affected by cyclic changes in power and temperature, which (coupled with its strong tendency to maintain a constant average core temperature) greatly facilitates its ability to load-follow. EBR-II could be easily controlled by fixing the control-rod position and controlling power demand at the steam turbine. A full range of safety and load-following tests were conducted, including (for example) rapid run-up of the primary pumps to their maximum capacity, which would cool the core, insert positive reactivity, and raise power level. No damage occurred to the fuel or core through all of these tests.

A level-1 probabilistic risk assessment (PRA) was completed to quantify these results. It was shown that risks associated with EBR-II operation were substantially lower (an order of magnitude less) than those associated with typical LWR plants. The EBR-II risks would have been lower still except for its seismic response. (Subsequent plants employ seismic isolation to mitigate even this risk.) An important result from this work was the finding that acts of commission (purposely disconnecting a pump, etc.) would not lead to core damage.

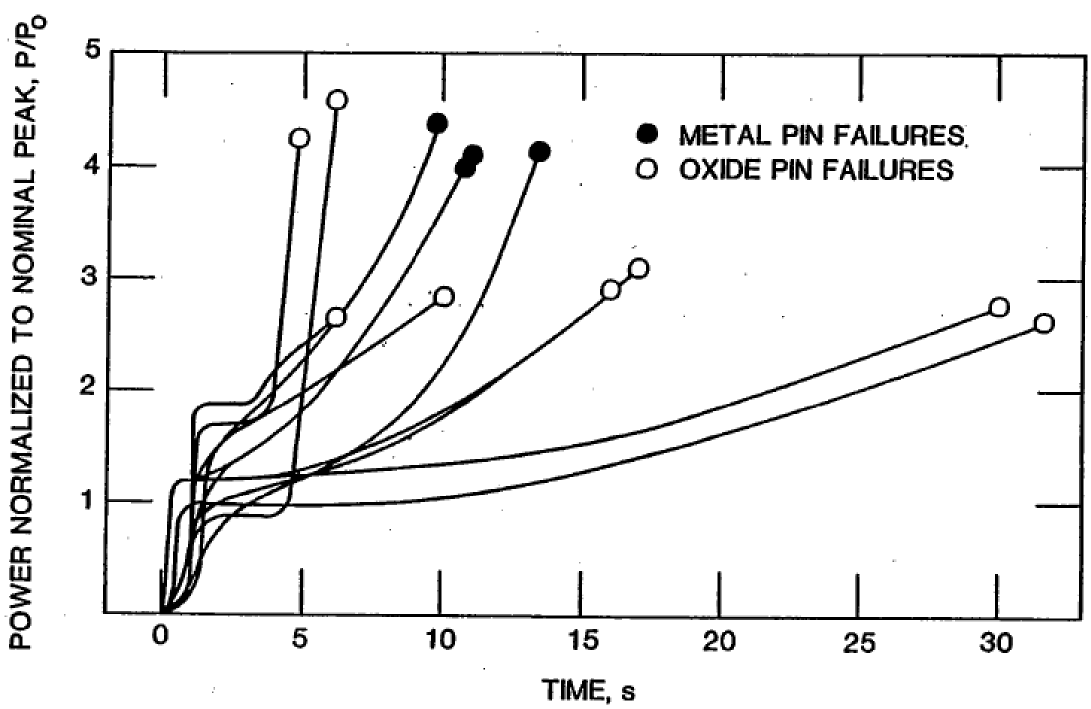

Figure C.32 Transient overpower tests to failure in TREAT

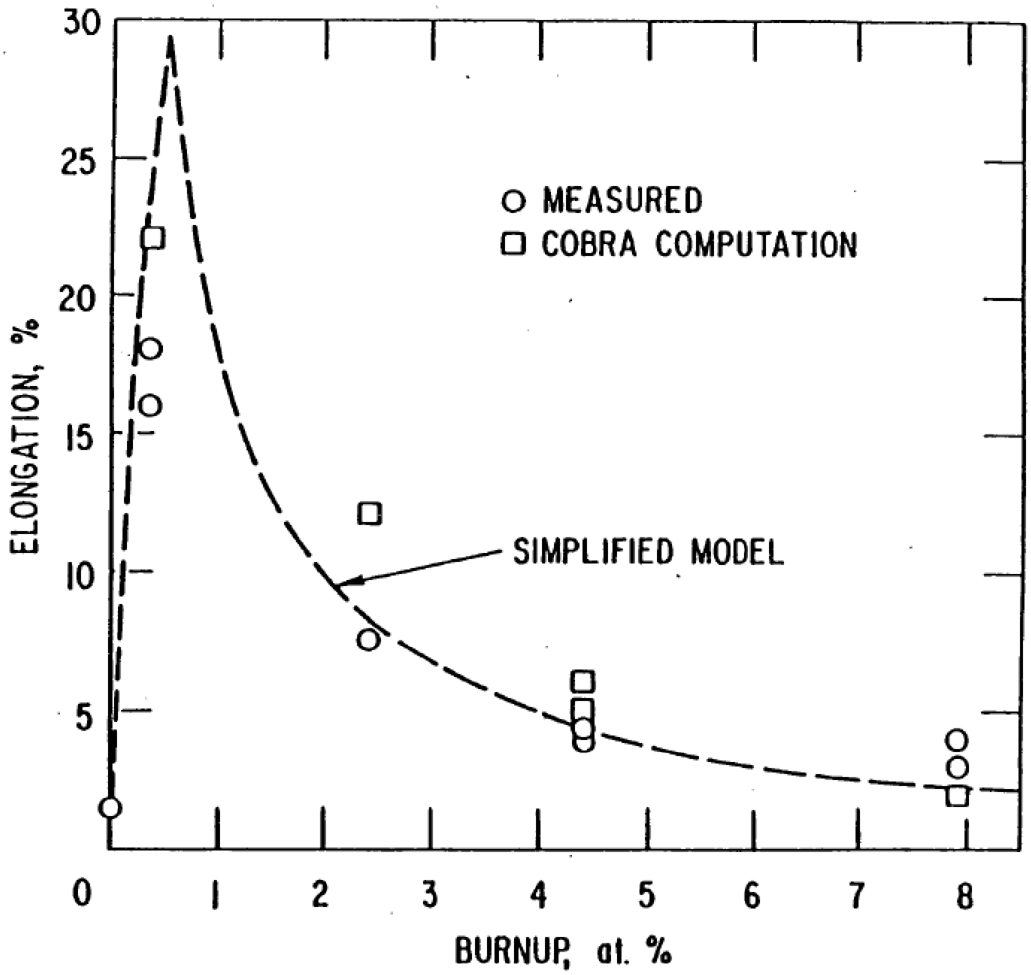

Figure C.33 Axial movement of metal fuel before pin breach

Mission IV: The Integral Fast Reactor (IFR)

With all that was learned through Mission III of EBR-II operation, the results were integrated into an approach to fast-reactor design termed the Integral Fast Reactor (IFR). A new feature of the approach was a reprocessing technology that accommodated fuel containing actinides and that offered proliferation resistance. The Fuel Cycle Facility (FCF) was refurbished and equipment was installed to conduct the work. The heart of the process was an electro-refiner into which the chopped EBR-II spent fuel was placed. The potassium/lithium chloride salt in the electro-refiner was kept at 500 degrees C and the fuel dissolved into it, leaving the cladding hulls behind. The anode for the electro-refiner was the fuel basket from which the fuel was dissolved and two types of cathodes were employed, a solid cathode on which uranium was deposited and a liquid cadmium cathode within which a mixture of uranium and transuranics were deposited. The reason that the system offers proliferation resistance is that it is virtually impossible to cleanly separate Pu. Through a quirk of nature, the free energies of Pu and the minor actinides in the salt are so closely aligned that it is virtually impossible to adjust electro-refiner voltages to distinguish between them for transport of material.

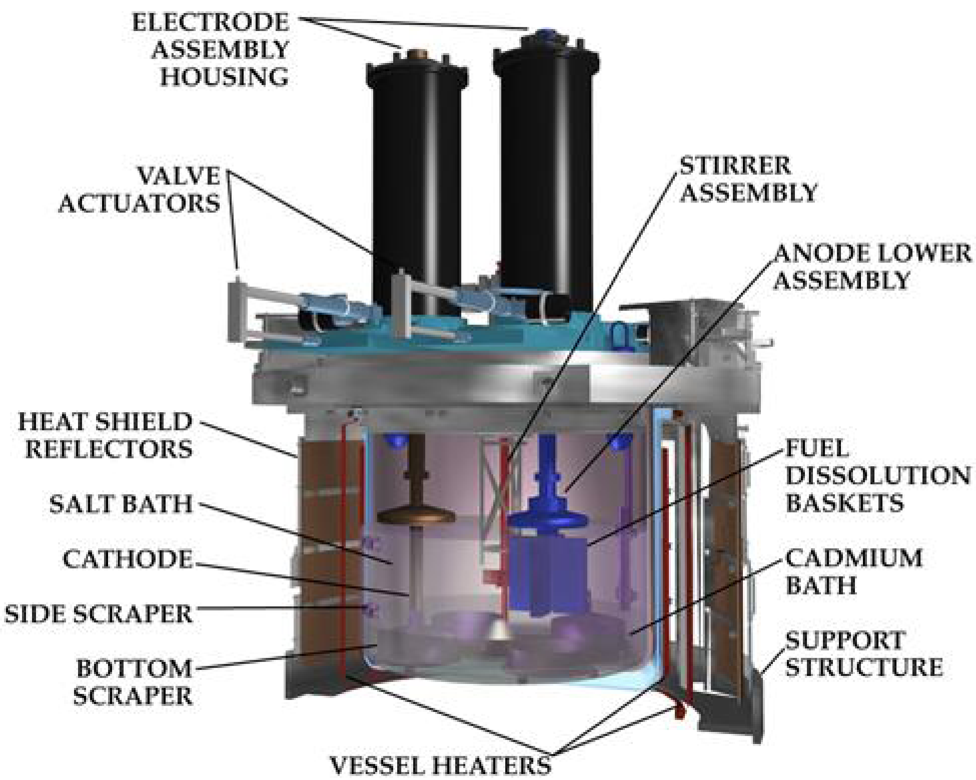

Figure C.34 The electro-refiner for reprocessing EBR-II spent fuel

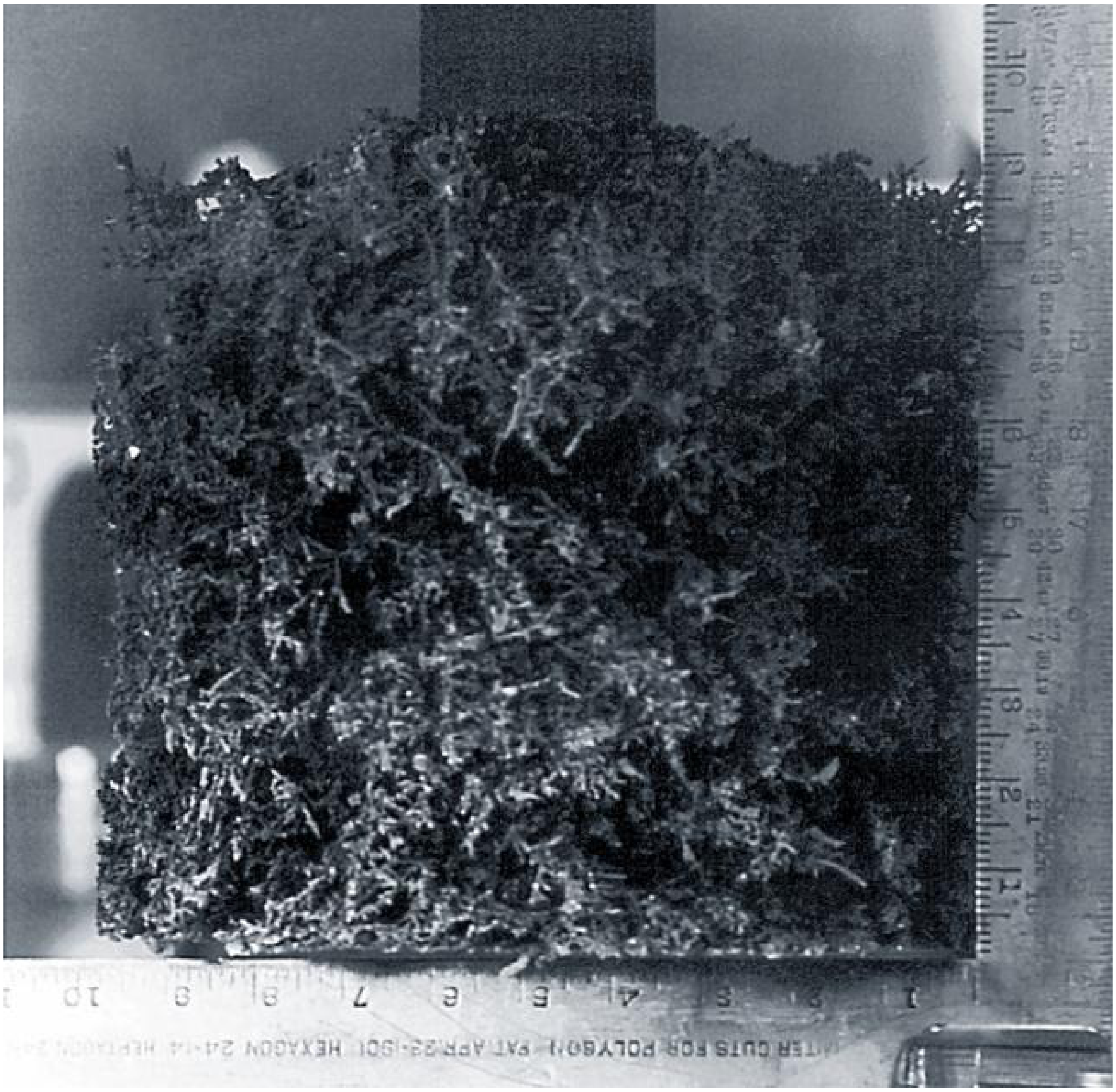

Figure C.35 Uranium collected on the electro-refiner's cathode

Spent fuel was first chopped and then loaded into the anode basket. Uranium was then electro-transported to a solid cathode. Subsequently, Pu and a mixture of minor actinides were transported to a liquid cadmium cathode. From there, material was taken to a cathode processor where clinging salt was distilled from the product, to be returned to the electro-refiner. The fuel product was then consolidated into an ingot for subsequent casting into fuel.

Table C.5 Free Energies of Chloride Formation

Free Energies of Chloride Formation at 500°C, -kcal/g-eqCl

Elements that remain in salt (very stable chlorides)		Elements that can be electrotransported efficiently		Elements that remain as metals (less stable chlorides)	
$BaCl_2$	87.9	$CmCl_3$	64.0	$ZrCl_2$	46.6
CsCl	87.8	$PuCl_3$	62.4	$CdCl_2$	32.3
RbCl	87.0	$AmCl_3$	62.1	$FeCl_2$	29.2
KCl	86.7	$NpCl_3$	58.1	$NbCl_5$	26.7
$SrCl_2$	84.7	UCl_3	55.2	$MoCl_4$	16.8
LiCl	82.5			$TcCl_4$	11.0
$CaCl_2$	80.7			$RbCl_3$	10.0
$LaCl_3$	70.2			$PdCl_2$	9.0
$PrCl_3$	69.0			$RuCl_4$	6.0
$CeCl_3$	68.6				
$NdCl_3$	67.9				
YCl_3	65.1				

An important aspect of the fuel cycle was the production of waste forms suitable for geologic storage. One was ceramic and the other metallic. To produce the ceramic waste form with the electro-refiner, salt was cleaned of active fission products by flowing it over a zeolite bed, which was then consolidated into a ceramic waste-form after the addition of glass frit. To create the metallic waste form, the cladding hulls and noble metals were recovered from the anode basket and the bottom of the electro-refiner and cast into a metal ingot. Extensive leach tests were conducted on these waste forms, which were qualified for long-term geologic disposal. Because the waste could be free of actinides, required storage times were on the order of hundreds, not thousands of years.

This work was overseen by a special committee of the National Academy of Sciences which issued a final report supportive of the technology.

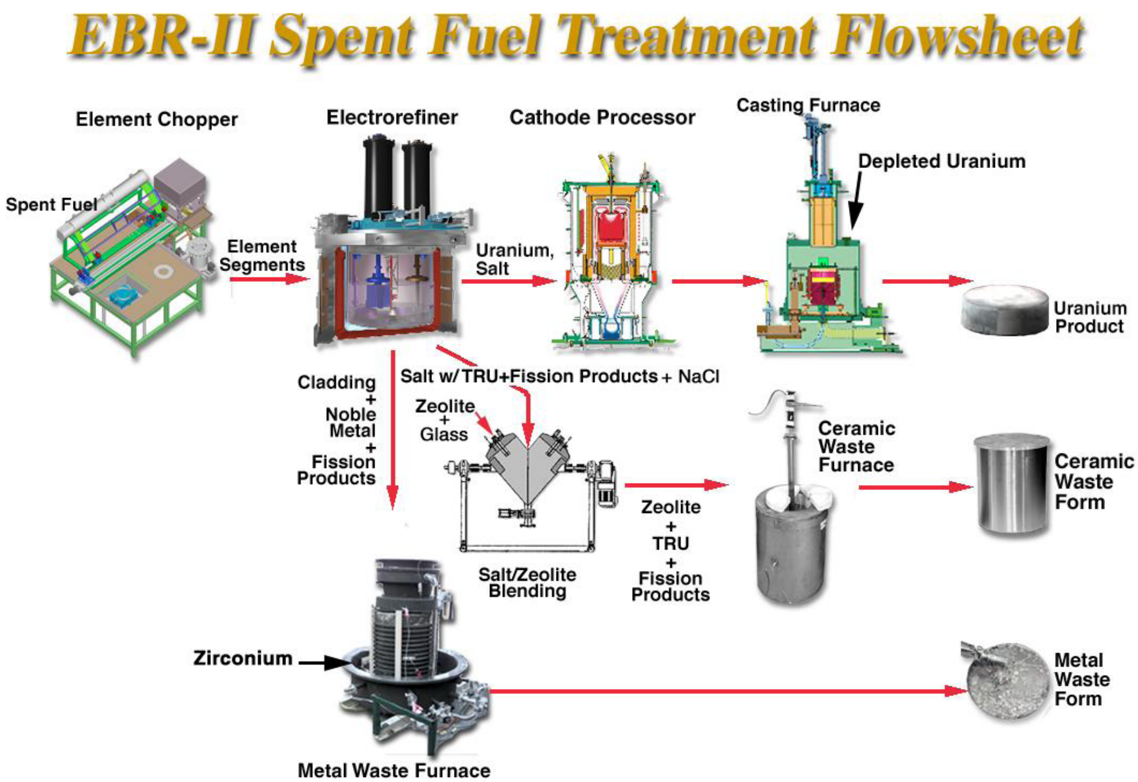

Figure C.36 EBR-II spent fuel treatment

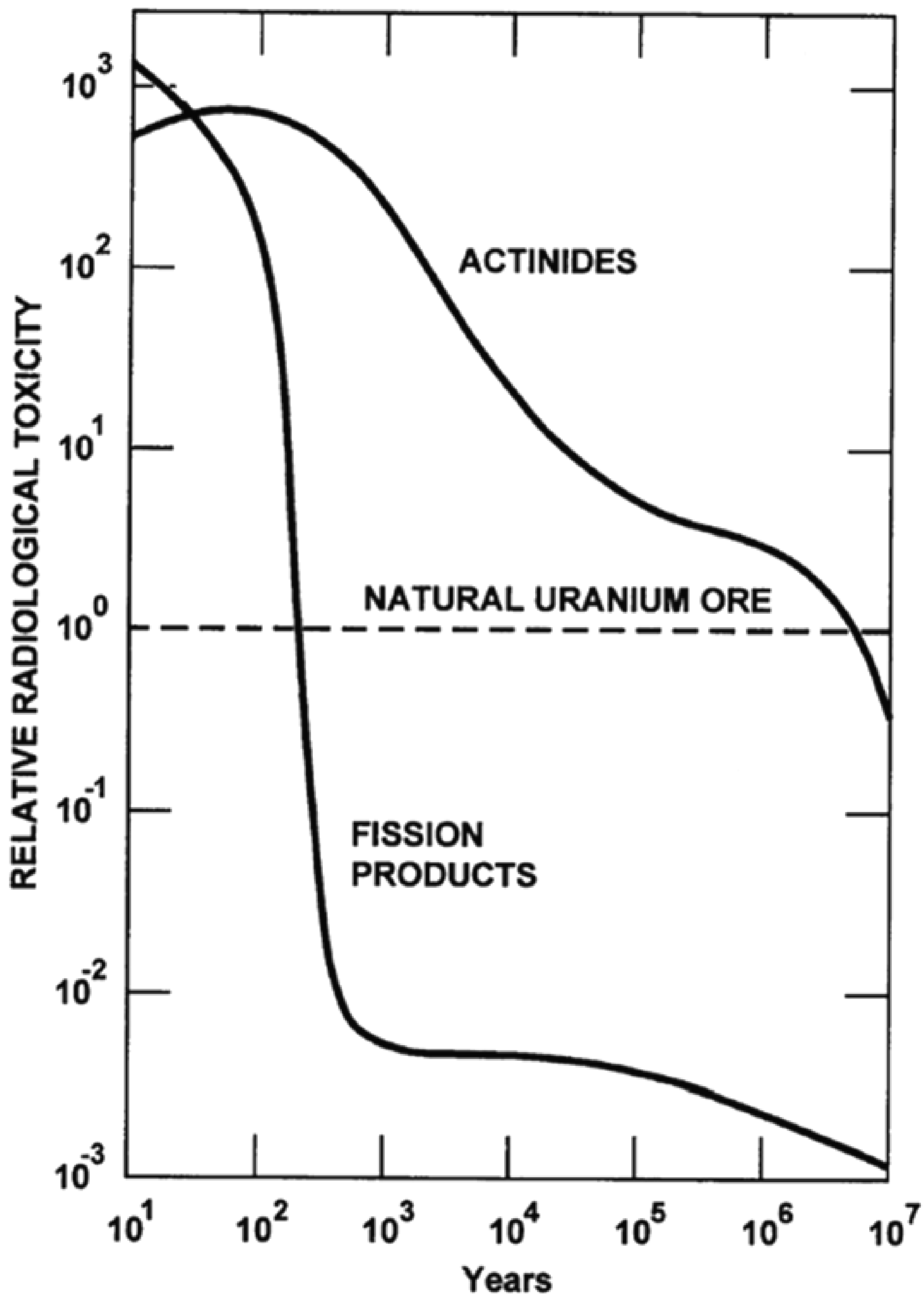

Figure C.37 Relative storage times of nuclear waste with and without actinides

Decommissioning of EBR-II

The first phase of decommissioning involved the removal of all fuel and blanket assemblies, 637 in all. This involved cleaning each assembly of sodium, transferring it to a hot-cell facility for disassembly and repackaging, and then transferring it to interim storage. As a final step, EBR-II fuel is being reprocessed for recovery of uranium and production of waste forms suitable for geologic storage, as described previously. Defueling was accomplished over 14 months without difficulty.

The next phase concentrated on the technology for dealing with sodium coolant to produce a waste form suitable for landfill disposal. It was also important that residual sodium left in the coolant systems after draining be fully reacted so it would not pose a long-term hazard. The 89,000 gallons of primary sodium was thoroughly cleaned of fission products (especially Cs-137) and sodium oxide. It was then transferred to a sodium-processing facility where it was reacted with water to produce sodium hydroxide at a concentration of 73 percent by weight. The sodium hydroxide at this concentration is a solid product that could be stored in drums at a DOE landfill. In addition to the EBR-II sodium, sodium drained from the Fermi 1 reactor was reacted and disposed of in this manner.

One of the more interesting challenges was passivating the residual sodium that remained in the primary system in order to place the system in a radiologically and industrially safe condition. After a number of laboratory tests, a solution was found. Moist CO_2 was introduced to the primary system at a controlled rate, and the reaction rate of the water vapor with the sodium was monitored by observing the evolution of hydrogen. It was uncertain how long the reaction would continue, because pools of sodium form a “scab” at the surface; however, it was found that over time, the CO_2 would permeate this surface layer and the reaction would continue to completion, although the process could take several years. The volume of residual sodium has now been reduced to a point that it would be safe to flood the primary tank with water.

An important observation after the sodium was drained from the primary tank was that the condition of the tank and the components submerged in sodium was pristine. There was absolutely no corrosion of the stainless steel after 35 years in contact with hot sodium.

Lessons Learned

The extensive program of operation and testing at EBR-II has established sodium-cooled fast reactors as a viable technology to support a nuclear renaissance. The ability of fast reactors to manage nuclear materials for waste and fuel has been demonstrated, along with advantages for safety, operability and reliability. Cost remains the major issue, but there are opportunities for significant cost reductions by taking advantage of the self-protecting nature of the reactor system to simplify design. The major conclusions are that:

- A pool-type, metal-fueled LMR nuclear generating station can be reliably operated and maintained with large margins of safety to workers and the public.
- Sodium system maintenance is straightforward and safe, facilitated by low pressure in operating systems.
- Sodium spills and fires are manageable, principally because of the lack of high-pressure driving fluid; no personnel injuries have been associated with leaking sodium systems.
- Sodium is highly compatible with reactor materials, facilitating long life.
- Attention must be given to maintaining purity of the inert gas covering sodium systems to avoid sodium oxide buildup on systems penetrating the interface.
- Personnel radiation-exposure levels are very low, typically less than 10 percent of those for LWR systems.
- A metal-fueled LMR nuclear generating station can be passively safe, offering self-protection against anticipated transients even without safety-system action.

- Safety benefits have been quantified by PRA, which demonstrates very low levels of risks.
- Metal fuel offers exceptional benefits for reprocessing and recycling, conversion, load-following, passive safety, and benign behavior in degraded condition.
- Fuel-handling systems require much design, operating, and maintenance attention to ensure reliability.

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APPENDIX C.3

INDUSTRY PERSPECTIVES AND EXPERIENCE IN THE DESIGN OF LIQUID-METAL-COOLED FAST REACTORS

Sterling Bailey, PhD, PE
General Electric Nuclear Energy (retired)

Introduction

The objectives of this paper are to provide fundamental information on the major design features of liquid-metal-cooled fast reactors, which are often referred to as "Liquid Metal Reactors" or "LMRs," with primary focus on the reactor core; to highlight their differences from light-water-cooled reactors (LWRs); to describe the design process used in the U.S. LMR programs; and to illustrate the experimental validation processes for key core features. The information presented is from the perspective of an industry technology manager in the U.S. LMR program.

Author's Background

Dr. Bailey received a B.A. in Physics from the University of California at Berkeley and an M.S. and Ph.D. in Nuclear Engineering from Stanford University. He initially worked for General Electric (GE) Nuclear Energy on boiling-water reactor (BWR) physics and core design and then on fast reactor technology. He participated in the physics analysis of the Southwest Experimental Fast Oxide Reactor (SEFOR) project, later led the GE nuclear design team for fast reactors, and subsequently was responsible for GE's advanced reactor engineering, including physics, thermal-hydraulics, mechanical design, and instrumentation and control. He was an active participant in the Department of Energy's (DOE's) liquid-metal fast breeder reactor (LMFBR) Base Technology Program and the design and analysis of the Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor Project (CRBRP). Dr. Bailey was the general manager of the DOE/National Aeronautics and Space Administration (NASA) SP-100 space reactor program, which used a high-temperature advanced LMR, and currently works with DOE and NASA on a small LMR for powering a lunar base.

Liquid-Metal-Cooled Fast Reactor Basic Principles and Design Features

Liquid-metal-cooled fast reactors use a fast neutron spectrum reactor core with no material added to moderate the neutron energy. The higher-energy neutron flux allows the reactor to take advantage of the high energy cross sections of the fuel and fertile materials and also avoid some of the parasitic capture from other core constituents. This provides a better neutron economy for breeding fissile material, recycling spent fuel, and burning actinide wastes from spent fuel compared to moderated light water reactors. The liquid metal coolant, typically sodium, provides very effective heat transfer from the reactor core, roughly 100 times as effective as water. The liquid metal coolant also has very low vapor pressure at operating conditions, which allows low-pressure piping. The high boiling point of the liquid metal also provides a large margin between operating temperatures and coolant boiling.

A typical fission energy spectrum is shown in Figure C.38, which illustrates that most of the neutrons released by the fission process have energies greater than 100 keV. Figure C.39 illustrates the energy dependence of the fission cross section for the most important fissile isotopes, ^{235}U and ^{239}Pu, as well as the fertile isotope ^{238}U. Note that neutrons with energy greater than 1 MeV can cause fission in ^{238}U. This is also true for some of the trans-plutonium isotopes. The energy dependence of typical capture cross sections is shown in Figure C.. The lower values of the cross sections at high energies require a higher neutron flux level to achieve equivalent reaction rates. This means that the fuel and structural materials in a LMR may be exposed to greater neutron fluence than in a LWR. The hard neutron spectrum also results in considerably longer mean free neutron path lengths, which simplifies some of the core nuclear design considerations compared to LWRs.

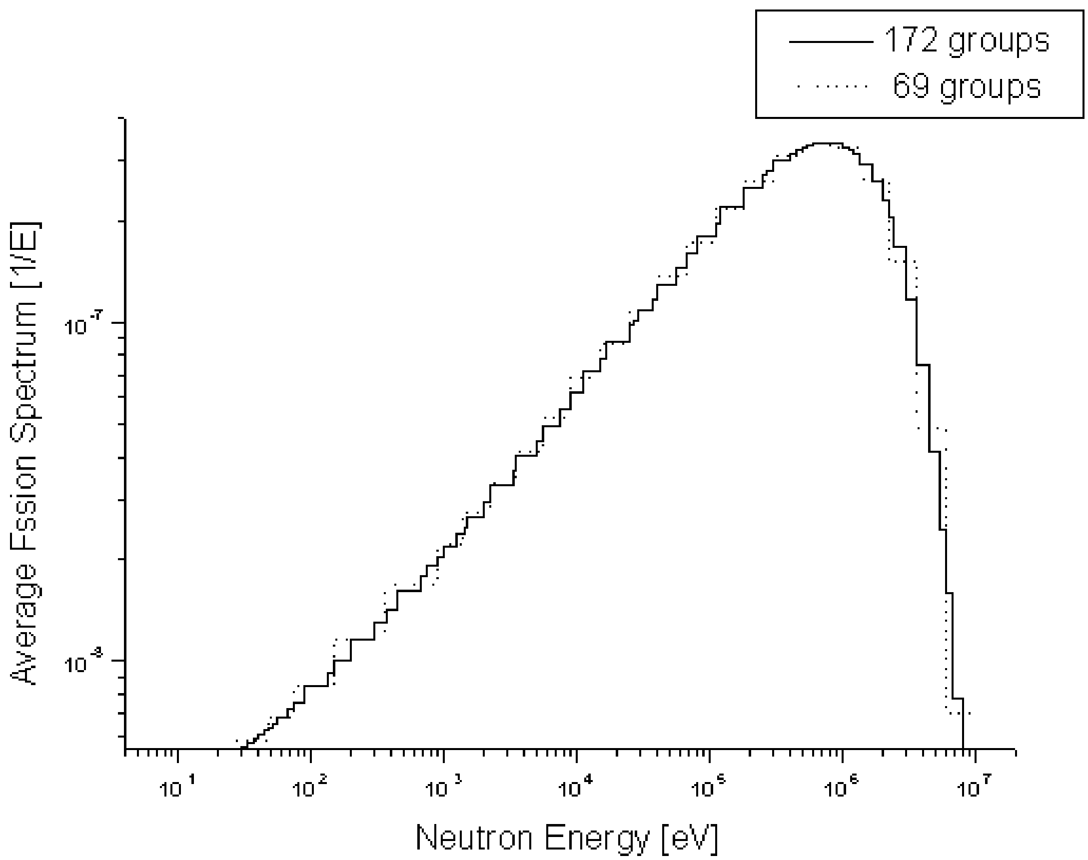


Figure C.38 Typical fission neutron energy spectrum

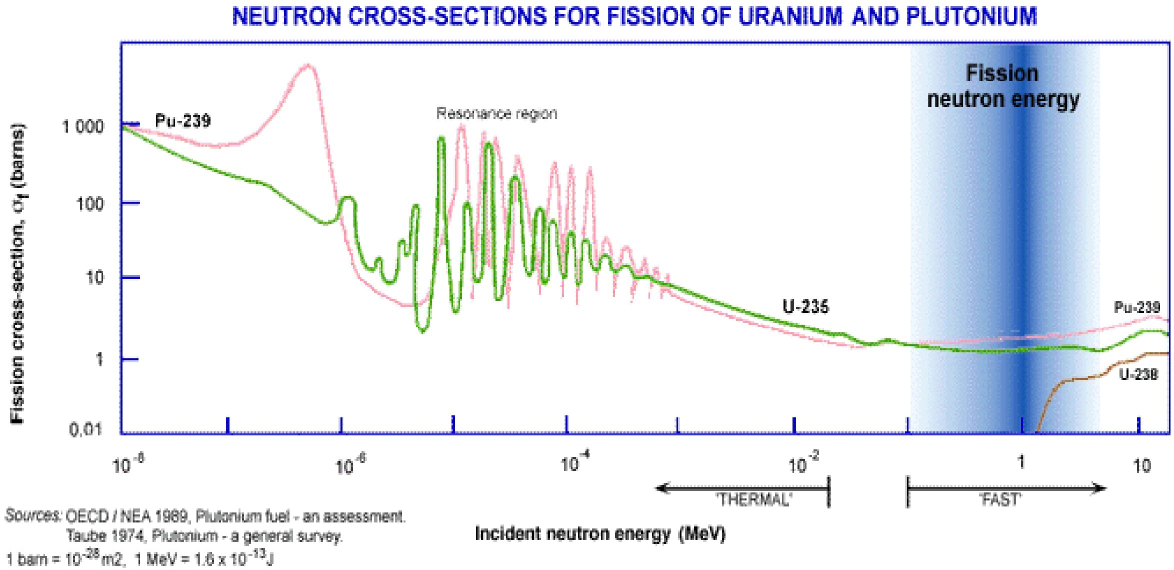


Figure C.39 Typical fission cross sections vs. neutron energy

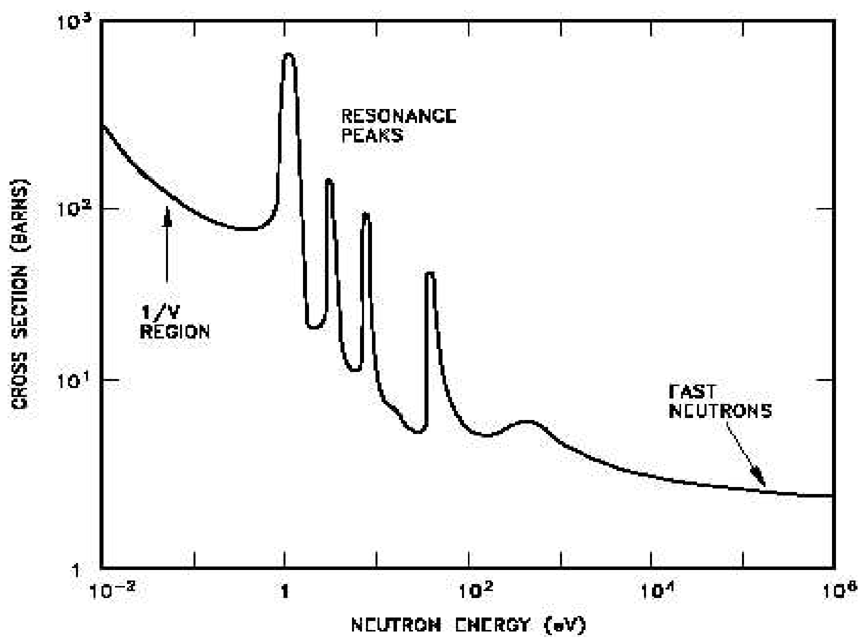


Figure C.40 Typical capture cross section energy dependence

The use of an unmoderated, or “fast,” neutron spectrum has several significant consequences on the performance and design of LMRs:

- Enables significant net breeding of fissile material
- Facilitates recycle of spent fuel—can increase energy from natural uranium by factor of ~60 compared to once-through LWRs

- Burns actinide waste isotopes—greatly reduces waste-management challenge
- Lower cross sections at higher energies leads to longer neutron mean free paths
- Higher fissile enrichments, approximately 20 to 35 percent
- Higher flux level required for equivalent reaction rate and higher fluence to fuel and materials
- Power distribution flatter, less sensitive to local geometry
- Higher leakage can make reactivity more sensitive to dimensional changes
- Mid-energy U^{238} resonances contribute to significant Doppler reactivity coefficient
- Lower reactivity reduction with burnup
- Fuel burnup of 150,000 to 200,000 MWd/T achievable compared to 30,000 to 50,000 MWd/T for LWRs
- Shorter neutron lifetime and reduced delayed neutron fraction impact dynamic behavior

The use of liquid metal coolant, with its excellent heat transfer properties, allows a relatively compact core design, minimizes temperature differences between the clad and coolant and within the flow field, and facilitates heat removal in many postulated accident scenarios. Sodium is the coolant chosen for most LMRs. The high boiling point of sodium (883 degrees C) and low vapor pressure at operating temperatures (<0.1 psi) allows operation at essentially atmospheric pressure, avoiding high-pressure piping and components. This also provides a large margin between the operating and boiling points of the coolant. However, the sodium coolant produces a radioactive isotope, ^{24}Na, under neutron irradiation. Hot sodium also has a very exothermic reaction with water. These two considerations lead to the incorporation of an intermediate coolant loop to isolate the radioactive primary sodium from the power-conversion components and to significantly reduce the potential consequences of a sodium leak and subsequent sodium-water reaction.

Sodium is very compatible with the structural materials commonly used for fuel cladding, vessels, piping, and components (Type 316 stainless steel and similar alloys), as long as the contaminants, primarily oxygen, in the sodium are kept sufficiently low. Piping used in the EBR-II reactor still clearly showed identification marks on the inside bore after more than 25 years of operation.

The key materials and operating temperature range for typical terrestrial LMRs are shown in Table C.6.

Table C.6 Typical LMR Materials and Operating Conditions

Fuel	Enriched UO_2 or PuO_2-UO_2 or Pu/U-Zr metal alloys or actinides in either form
Fertile Blanket	$^{238}UO_2$
Clad	Type 316 stainless steel or advanced alloys
Coolant	Sodium
Structure	Type 316 stainless steel
Control	B_4C enriched in ^{10}B
Core Outlet Temperature	500–550°C

The fuel material is typically a sintered pellet or a metallic rod contained in cylindrical clad with welded end caps. Figure C.41 illustrates a typical fuel pin design, in this case from the

U.S. FFTF reactor, and the nose piece that is used to hold fuel pins together to form a fuel assembly or fuel bundle.

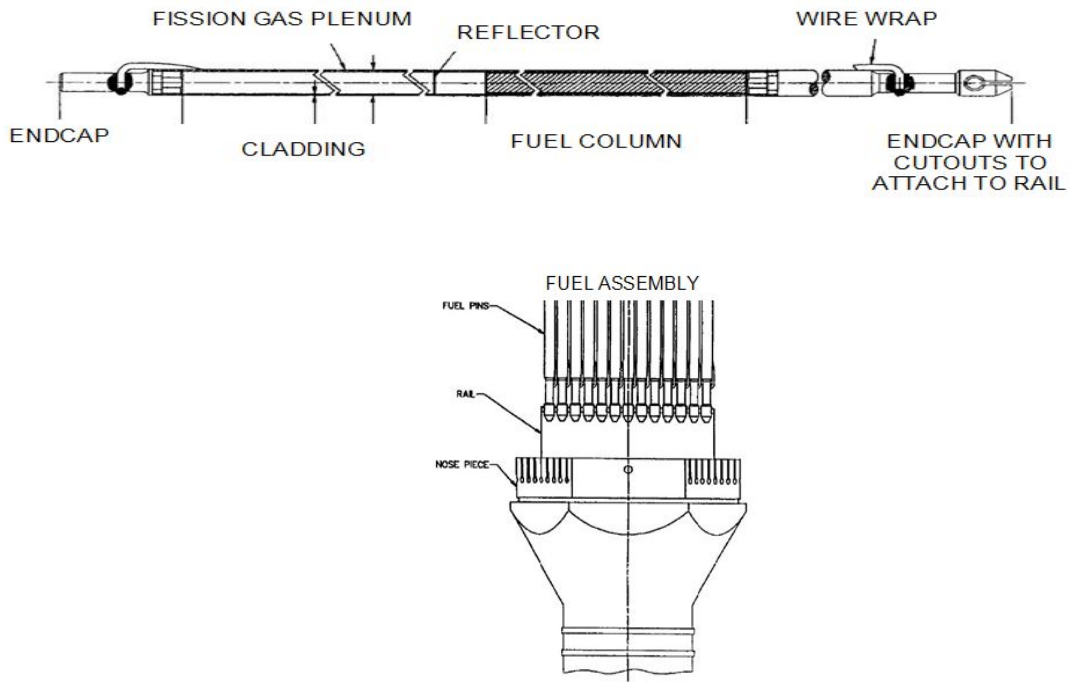


Figure C.41 Typical LMR fuel pin and nose piece design

The fuel pins are arranged in a tightly packed triangular array usually spaced from one another by wire wraps and then inserted into a hexagonal fuel assembly. The end caps are welded onto the clad and create a hermetically sealed structure. The fuel pin is typically back-filled with an inert gas such as helium to enhance the heat transfer from fuel to clad. The fission gas plenum provides space for gaseous fission products to accumulate and reduces the clad stress from internal pressure. Typically the enriched fuel length is on the order of three feet and the clad outer diameter is about 0.25 inches with a thickness of about 20 mils.

Figure C.42 illustrates the FFTF fuel assembly design.

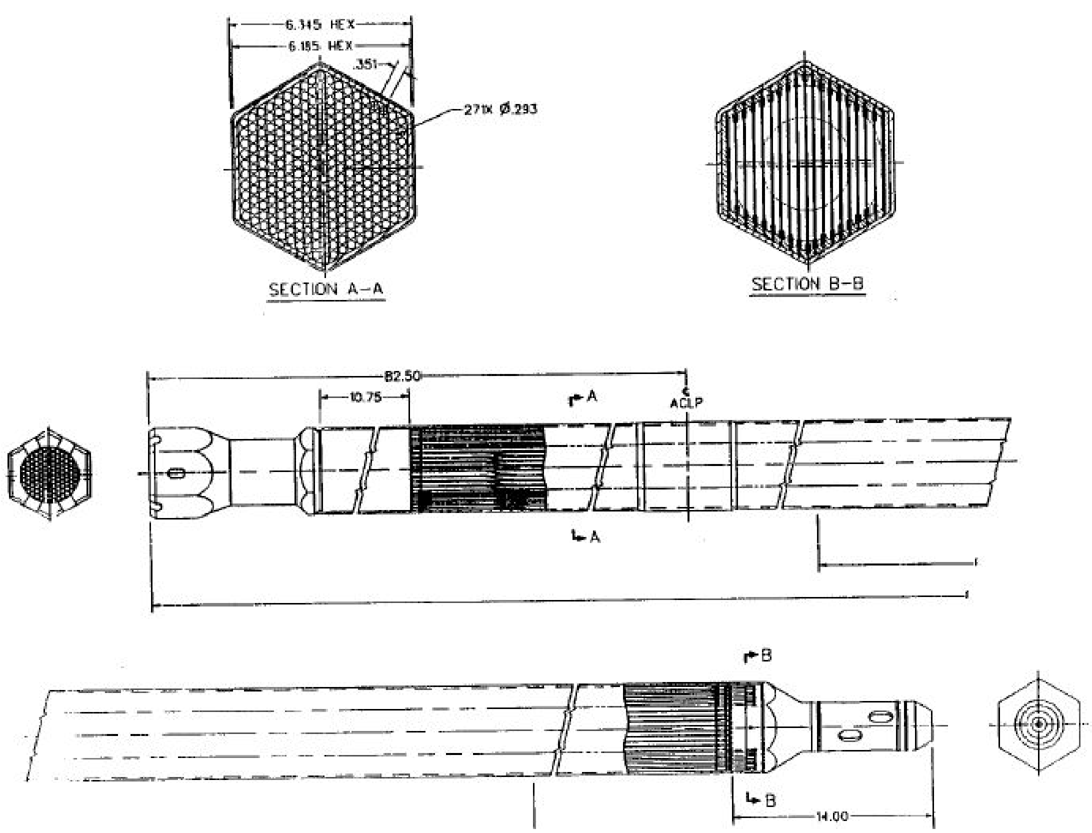


Figure C.42 FFTF fuel assembly design

The fuel assemblies are arranged with control assemblies, shutdown assemblies, breeding blanket assemblies, and reflector or shielding assemblies to create the reactor core. Figure C.43 shows one typical LMR core arrangement designed for burning actinides in recycled U–Pu fuel. In this arrangement, two enrichment zones are used to reduce the radial power peaking across the core. The control assemblies consist of enriched B_4C pellets in clad that can be moved into or out of the core vertically and change reactivity by several dollars. These control assemblies keep the core in a subcritical cold shutdown condition until the planned startup. Withdrawal of some control assemblies allows the core to be critical and come to operating temperature. Additional slow withdrawal compensates for the small reactivity decrease as the fuel is partially burnt up. Burnup of 15 to 20 percent of the heavy metal fuel content can be achieved with established LMR technology.

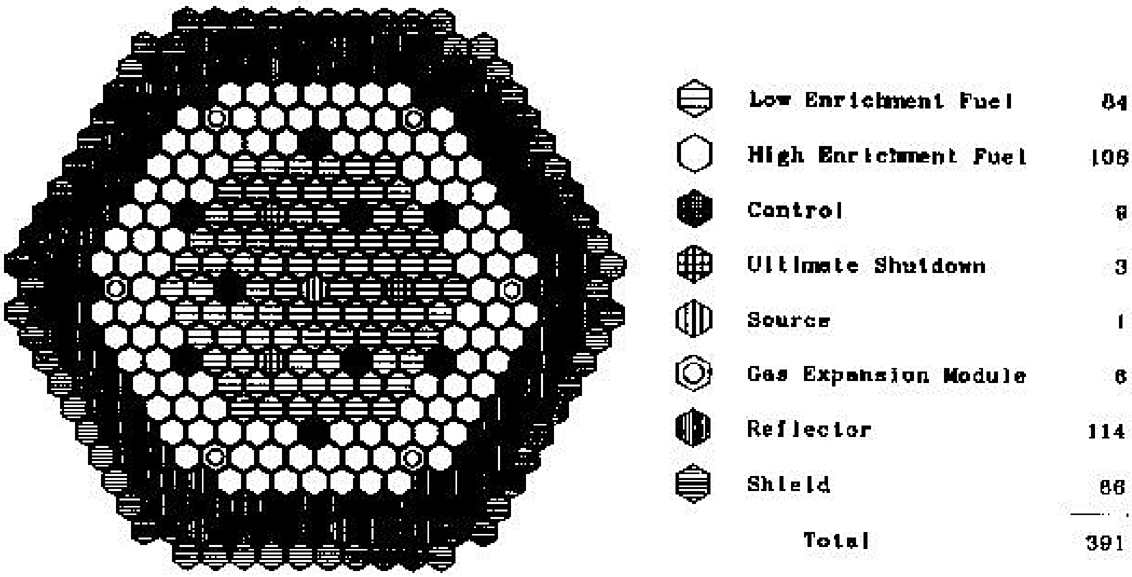


Figure C.43 Typical LMR core configuration

The Ultimate Shutdown and Gas Expansion assemblies provide additional reactivity control to assure final shutdown and to mitigate postulated accident conditions. The Reflector assemblies reduce neutron leakage from the core and improve overall neutron economy. For cores designed to breed fissile material, the Reflector assemblies would be replaced by fertile blanket assemblies. Thus, a LMR plant can be changed from a fissile burner to a fissile breeder by a relatively simple change in the type of assemblies loaded into the core matrix. The shield assemblies provide some of the neutron and gamma shielding required.

The core assemblies are inserted into a lower grid structure within the reactor vessel. The nose piece for each assembly typically contains flow orifices that control the amount of coolant flow delivered to the assembly and provide a relatively flat radial temperature profile.

LMR Plant Configurations

An LMR plant typically consists of the reactor core and associated instrumentation and control, a vessel, a primary heat transport system (PHTS), an intermediate (or secondary) heat transport system (IHTS), a steam generator, a power conversion turbine/generator, containment, shielding, and associated balance-of-plant systems. These systems can be configured in either of two basic arrangements, loop or pool. Typical LMR pool and loop configurations are shown schematically in Figure C.44.

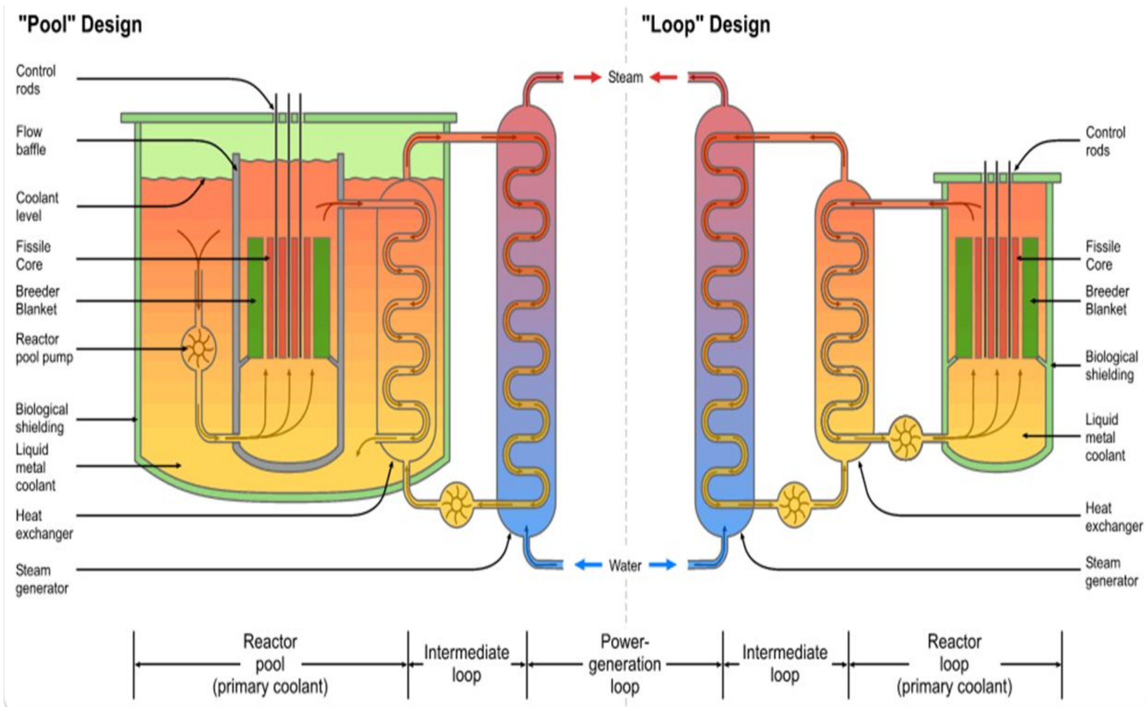


Figure C.44 Typical pool and loop LMR configurations

In the pool configuration, all of the primary sodium loop components are contained in the reactor vessel along with the core and the intermediate heat exchanger. The intermediate loop pump and steam generator are outside the vessel. This arrangement keeps all of the radioactive sodium within the vessel and can reduce the total shielding required. The pool arrangement generally provides a more compact plant than a loop configuration.

In the loop arrangement, the reactor core, fertile blankets, near-core shielding, control elements, and flow plenums are contained in the reactor vessel while the remainder of the primary heat transport system (PHTS), the intermediate heat transport system (IHTS), and the other systems are outside the vessel. This arrangement provides more access to the IHTS components and the primary pump than the pool configuration.

These different configurations impact many aspects of the design, such as economics, inspectability, maintainability, and response to postulated failures. The optimal choice often depends on the specific size and application. However, the overall plant configuration has only limited impact on the core requirements, performance, or technology. Several plants of each configuration have been successfully built and operated.

Most of the worldwide LMRs have used traditional water Rankine cycle turbine/generators to convert the thermal energy generated by fission into electric power. However, LMRs are not constrained to this power conversion technology. Brayton cycles, either open or closed loop, Stirling, thermoelectric, or essentially any power-conversion technology can be readily coupled with LMRs. The choice will depend on the power level, operating temperature, and power conversion technology maturity and reliability, as well as economics for any specific project.

The relatively simple layouts shown in Figure C.44 become much more complicated when all of the necessary subsystems and components are included. Figures C.45, C.46, and C.47 illustrate the progression from the simple configuration schematic to a more complete pool design, plant layout and actual physical plant with supporting infrastructure.

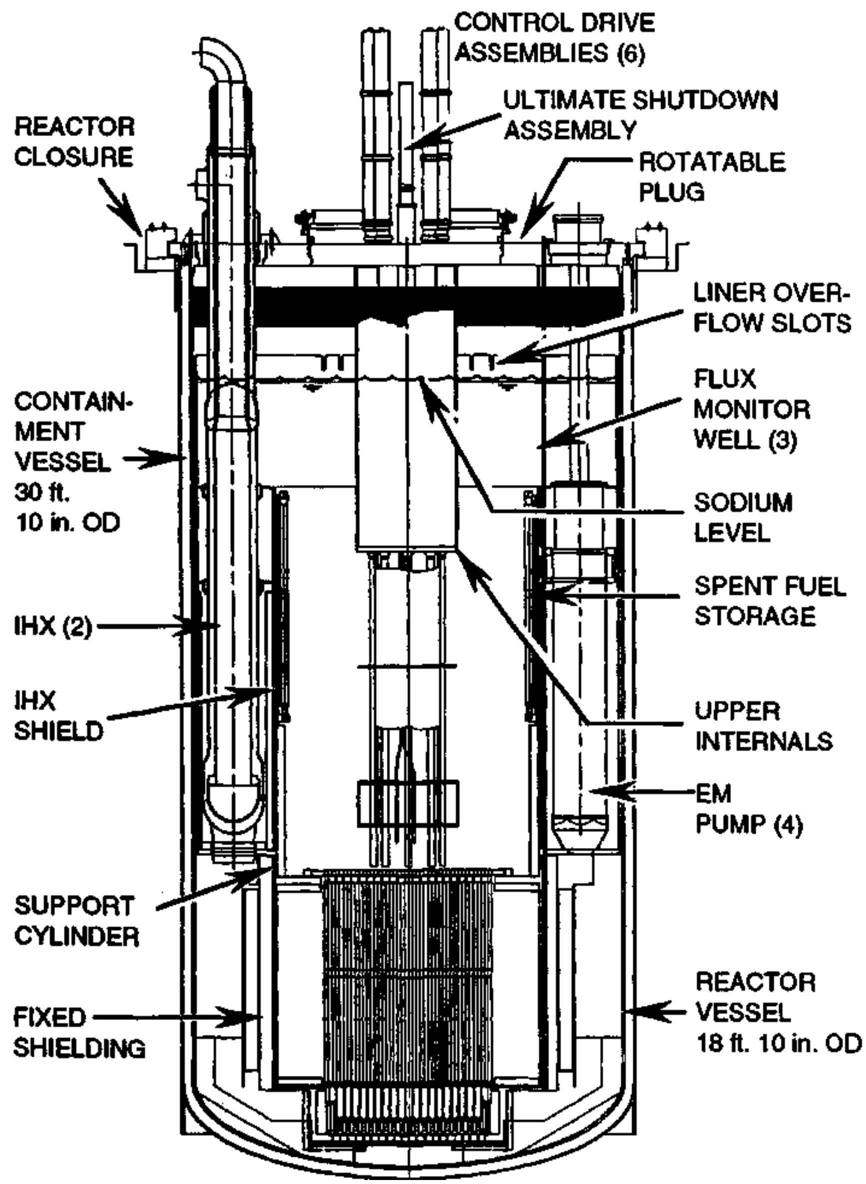


Figure C.45 More complete pool design layout

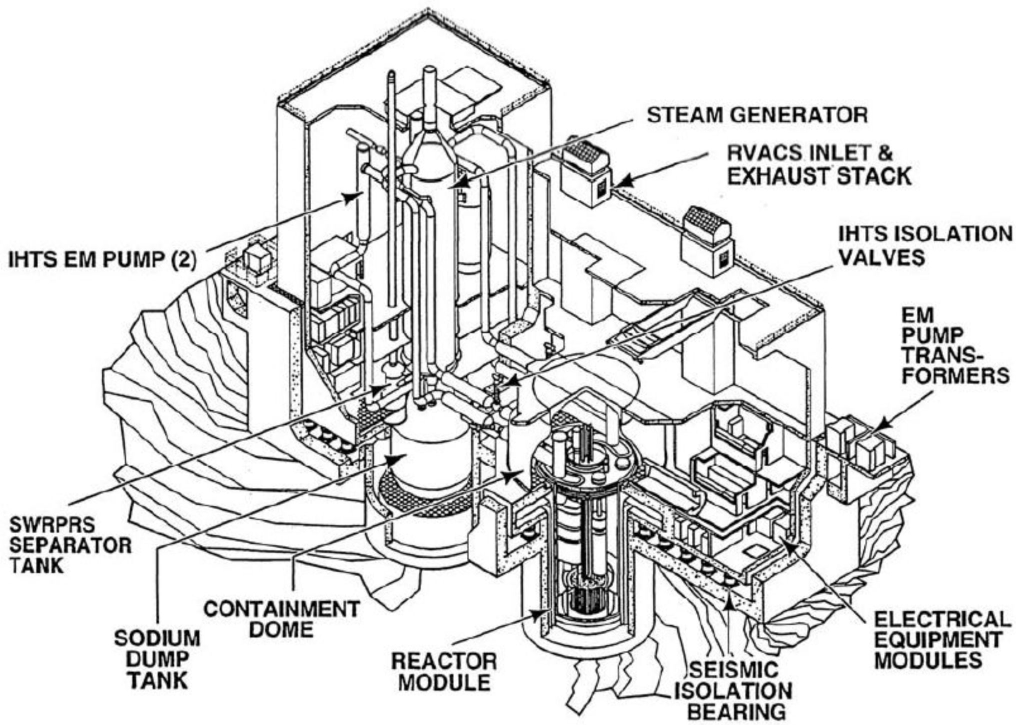


Figure C.46 LMR plant layout including power-conversion subsystems

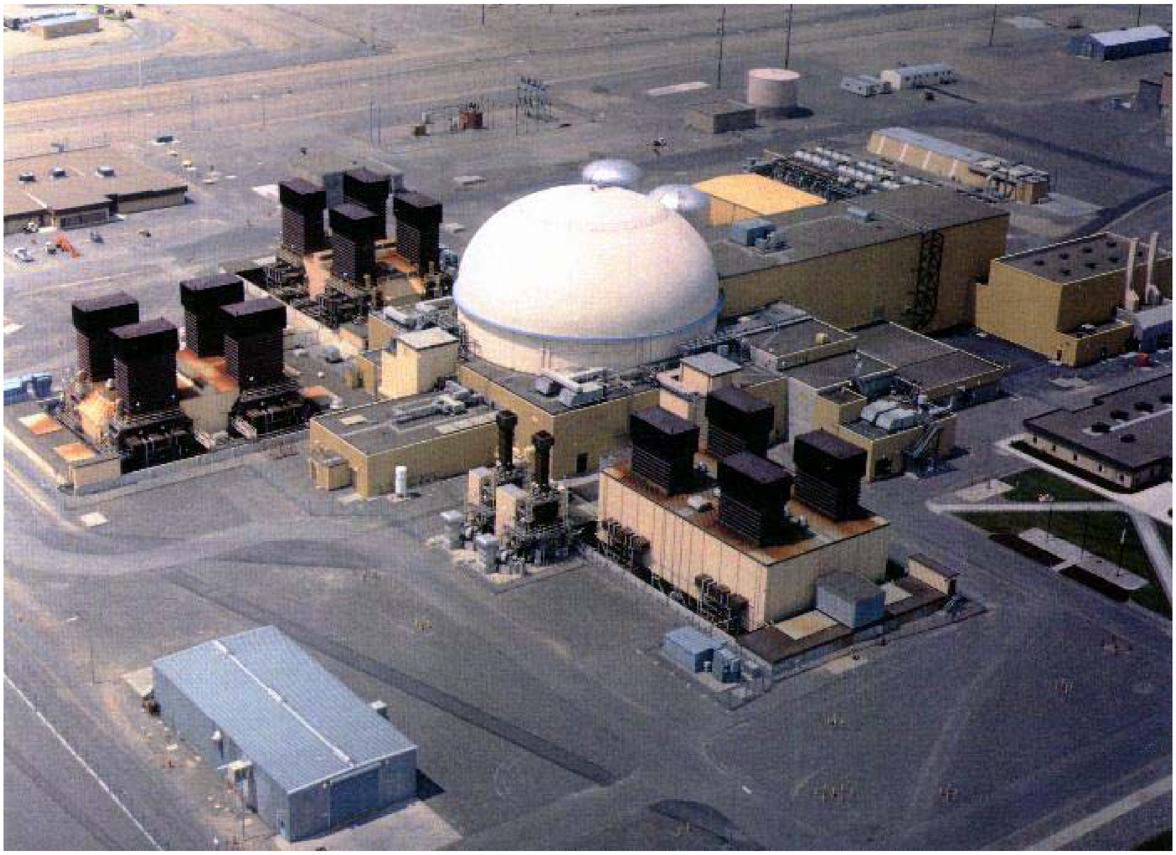


Figure C.47 FFTF LMR plant in Richland, Washington

Brief History of U.S. and International LMRs

There is a very extensive database of LMR design, construction, and operating history, as well as an extensive technology knowledge base from the LMR technology development programs. In the 1940s to early 1950s, several countries, including the U.S., started working on fast reactor technology. In 1946, the U.S. Clementine fast reactor became critical and this was the first operational fast reactor. In 1951, the first electricity produced by a nuclear reactor was generated by the EBR-I liquid metal cooled fast reactor. The key U.S. LMR projects since 1950 are summarized in Table C.7.

Table C.7 U.S. LMR Projects after 1950

- Experimental Breeder Reactor-I (EBR-I)
  - Operated 1951–1964
  - World's first electricity from a nuclear plant
- Experimental Breeder Reactor-II (EBR-II)
  - Critical 1961, power operation 1964–1994
  - Major contribution to fuels and materials testing
- Enrico Fermi Nuclear Generating Station (Fermi)
  - Operated 1963–1972
  - First attempt at commercial LMR plant
- Southwest Experimental Fast Oxide Reactor (SEFOR)
  - Operated 1969–1972
  - Definitive measurement of oxide-fueled LMR Doppler feedback
- Fast Flux Test Facility (FFTF)
  - Operated 1980–1992
  - Established world record for fuel performance
- Clinch River Breeder Reactor (CRBR)
  - Design began 1969, 1982 NRC site preparation approval
  - Funding cut off by Congress 1984
- Extensive design studies for commercial LMRs
- 1964 1000-MW(e) designs
  - Separate designs developed by GE, Westinghouse, Combustion Engineering, and Allis-Chalmers
  - Oxide and carbide fuel studied
  - Varying core aspect ratios and layouts
- 1967–69 Follow-on 1000-MW(e) studies
  - Focus on U-Pu oxide fuel
  - Loop and pool configurations
  - Different core configurations
- Reduced effort with focus on FFTF and CRBR
- 1977 President Carter deferred commercialization tasks, emphasized non-proliferation
- Studies such as Advanced Liquid Metal Reactor (ALMR) continued at lower level
- Current Global Nuclear Energy Partnership (GNEP) studies reflect renewed interest in LMRs

The extensive worldwide LMR experience is summarized in Table C.8 below, which is taken from the 2006 Revision of the International Atomic Energy Agency's (IAEA's) Fast Reactor Database.

Table C.8 Worldwide LMR Experience Summary

Experimental Fast Reactors

Plant	Dates of major events				
	Start of construction	First criticality	First electricity generation	First full power operation	Final shutdown
Rapsodie (France)	1962	Jan. 1967		Mar. 1967	Apr. 1983
KNK-II (Germany)		Oct. 1972	Apr. 1978	1978	Oct. 1991
FBTR (India)	1972	Oct. 1985	1994	1996	
PEC (Italy)	Jan. 1974	project cancelled			
JOYO (Japan)	Feb. 1970	Jul. 2003*		Oct. 2003*	
DFR (UK)	1954	1959	1962	1963	1977
BOR-60 (Russian Federation)	1964	1968	1969	1970	
EBR-II (USA)	June 1958	**	Aug. 1964	1965	1998
Fermi (USA)	Aug. 1956	Aug. 1963	Aug. 1966	Oct. 1970	1975
FFTF (USA)	June 1970	Feb. 1980		Dec. 1980	1996
BR-10 (Russian Federation)	1956	1958		1959***	Dec. 2003
CEFR (China)	May 2000	To be determined			

Demonstration or Prototype Fast Reactors

Phénix (France)	1968	1973	1973	Mar. 1974	****
SNR-300 (Germ.)	1973, finished in 1985; in 1991 the Government announced that SNR-300 should not proceed to commence operation				
PFBR (India)	2003	To be determined			
MONJU (Japan)	1985	1994	1995		
PFR (UK)	1966	1974	1975	1977	Mar. 1994
CRBRP (USA)	project cancelled				
BN-350 (Kazakhstan)	1964	Nov. 1972	1973	mid 1973	Apr. 1999
BN-600 (Russian Federation)	1967	Feb. 1980	Apr. 1980	Dec. 1981	not determined
ALMR (USA)	not determined				
KALIMER-150 (Republic of Korea)	not determined				
SVBR-75/100 (Russian Federation)	not determined				
BREST-OD-300 (Russian Federation)	not determined				

Table C.8 Worldwide LMR Experience Summary (continued)

Commercial Size Reactors

Plant	Dates of major events				
	Start of construction	First criticality	First electricity generation	First full power operation	Final shutdown
Super-Phénix 1 (France)	1976	1985	1986	1986	1998
Super-Phénix 2 (France)	project subsumed into EFR				
SNR 2 (Germany)	project subsumed into EFR				
DFBR (Japan)	not determined				
CDFR (UK)	project subsumed into EFR				
BN-1600 (Russian Federation)	project subsumed into BN-1800				
BN-800 (Russian Federation)	2002*	2012	to be determined		
EFR	not determined				
ALMR (USA)	not determined				
SVBR-75/100 (Russian Federation)	not determined				
BN-1800 (Russian Federation)	not determined				
BREST-1200 (Russian Federation)	not determined				
JSFR-1500 (Japan)	not determined				

U.S. LMR Design Process

The design process for U.S. LMRs has consistently followed a logical approach of establishing requirements for safety, operational performance, and economics and applied relevant regulations and technological and programmatic constraints. Lessons learned were factored into the evolution of the design process and were documented based on the engineering practices of the responsible organization. However, because the early projects did not use uniform design methods, fabrication controls, or documentation standards, much of the data from the early U.S. LMR program is difficult to use in modern engineering practice. In addition, those efforts had a component of “trial and error philosophy” characteristic of early technology development.

The U.S. LMR program became considerably more disciplined beginning in the late 1960s. In this period the U.S. Atomic Energy Commission (AEC) brought several managers from the Naval Reactors program into the civilian advanced reactors program. A disciplined system of Reactor Development and Technology (RDT) Standards for LMRs was instituted that reflected some of the methodology and lessons learned from Admiral Hyman G. Rickover’s programs. The resulting RDT Standards approach was applied to the LMR Base Technology development program and to the design, construction and operation of FFTF and the design, analysis, and fabrication of components for the CRBRP. Over 300 RDT Standards related to LMRs were issued and were frequently updated as new information became available. Included in the RDT Standards is the process for creating and maintaining the System Design Descriptions (SDDs) for a project. The SDD-31 covered the reactor core, and the FFTF and CRBR SSD-31s are especially useful for ongoing LMR core design work.

An important current benefit from this rigor is that the data and conclusions from prior work can be effectively used now and satisfy strict QA requirements as well as good engineering practice because of the prior rigorous QA and documentation requirements. An example of this process (and of RDT standards) is RDT F9-7, which prescribes the structural design criteria to be used for FBR core components. This standard is complemented by F9-8 and F9-9, which give the Guidelines for Analysis and the Rationale for the structural criteria respectively. With the downturn of the U.S. LMR program, most of the LMR-focused RDT standards were cancelled in the 1979 to 1996 period, but some were converted to DOE Office of Nuclear Energy (NE), American Society for Testing and Materials (ASTM), or American Society of Mechanical Engineers (ASME) standards. In addition, copies of the previous standards are preserved in archives and can be reactivated for current LMR design work.

In 1970 the American Nuclear Society (ANS) established a working group to develop principal design criteria for LMRs that would be consistent with the general design criteria requirements of Appendix A, “General Design Criteria for Nuclear Power Plants,” to Part 50, “Domestic Licensing of Production and Utilization Facilities,” of Title 10 of the *Code of Federal Regulations* (10 CFR 50) This group produced American National Standards Institute (ANSI)/ANS-54.1-1989, “General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant.”

This standard provides top-level design guidance that was incorporated in the FFTF and CRBRP design process and is applicable to current LMR designs.

The EBR-II, FFTF, CRBRP, and DOE LMR Base Technology programs were purposely structured as multi-organizational efforts. The intent was to apply the best talent to each task, provide cross-checks (facilitated by natural competitiveness), and promote energetic, full discussion of technical issues. The major participants were industry organizations who were expected to design, build, and operate LMR plants (primarily Westinghouse, General Electric, Babcock and Wilcox, Combustion Engineering, and Atomics International); DOE national laboratories who were to develop specific LMR technologies to be used in the plants (primarily Argonne National Laboratory (ANL) East and West, Hanford Engineering Development Laboratory (HEDL), Pacific Northwest National Laboratory (PNL), Brookhaven National Laboratory (BNL), Oak Ridge National Laboratory (ORNL), Liquid Metal Engineering Center (LMEC), and Los Alamos National Laboratory (LANL)); and electric utilities, mainly represented by groups like ESADA (Empire State Atomic Development Associates) and EPRI (Electric Power Research Institute). Universities also participated but with much smaller roles. There was also considerable international exchange, primarily with England, France, Japan, and Germany.

The benefit of this very large group of participants was that many points of view were brought forward and extensively debated, different testing and analysis methods were tried and inter-compared, and the varying interpretations of test data were argued out. When the debating ended, the consensus results were very sound and provided solid technical bases for design and successful operation. This is borne out by the nearly perfect operation of the LMR reactor cores; essentially all of the LMR plant problems have been related to out-of-core components.

LMR Technology Validation Process and Examples

A significant part of the U.S. LMR development and design strategy after the mid-1960s was to conduct a robust Base Technology Development program with extensive experiments covering all essential aspects of LMRs so that there was a very high probability of successful fabrication and operation once a design was finalized. This technology development work was subject to the RDT Standards and was carried out in a disciplined and well-documented manner. Table C.9 is a partial listing of the technology areas investigated in the Base Technology program.

Table C.9 LMR Base Technology Program Areas of Focus (partial list)

- Safety
- Reactor core physics
- Fuels and materials
- Structural design methods and guidelines
- Thermal hydraulics
- Pumps – mechanical and electromagnetic
- Steam generators
- Sodium chemistry
- Self-actuated shutdown mechanisms
- Control elements and drives
- Instrumentation and control methodology
- Quality assurance and reliability

Reactor Core Physics

The reactor core physics area provides an illustrative example of the type of technology validation process used in the LMR Base Technology Development program. Several industrial firms, national laboratories, and university organizations participated in the program with tasks assigned, coordinated, and monitored by AEC/DOE headquarters. Measurements were made of fundamental cross-section data and processed into the Evaluated Nuclear Data File (ENDF-B) format for the isotopes important to LMR operation and safety. International cooperation at this level was encouraged. The analytical methods used to predict criticality, neutron- and gamma-flux spatial and energy distributions, and reaction-rate distributions using diffusion theory, transport theory, and Monte Carlo techniques were evaluated against experimental data and empirical calibration factors; error estimates were developed as well. The intercomparison of these methods also provided insight into the implications of the approximations to the Boltzmann equation made in practical design codes.

Some of the most powerful experimental tools used in the LMR core physics technology validation were the “critical experiment” Zero Power Reactor (ZPR) and Zero Power Physics Reactor (ZPPR) facilities run by Argonne National Laboratory. Figure C.48 shows the ZPR-6 and ZPPR facilities.

Zero Power Reactor 6 (ZPR-6)

Figure C.48 LMR critical experiment facilities

The Zero Power Physics Reactor (ZPPR) at the Idaho National Laboratory (INL).

Image Credit: *Argonne National Laboratory*

Figure C.48 LMR critical experiment facilities (continued)

In both of these facilities, small sheets of fuel, structure, control, and coolant materials are put into long metal drawers that are then inserted into matrix structures in the two halves of the experimental rig. The actual composition of the reactor core design can be closely approximated by the correct drawer loading. Because the neutron mean free path is on the order of a few inches, the heterogeneity of the critical experiment loading is not “seen” by the neutrons and the actual neutron behavior in the reactor design is very closely mimicked in the critical experiment. After the core simulation is loaded into the two halves, they are slowly driven together. Reactivity is measured during this process and with the correct loading the array becomes exactly critical when the two halves are just touching. Fine shim control allows the critical experiment to be kept with k = 1.0 and operate at essentially zero power generation. Instrumentation placed in and around the matrix allows measurement of the neutron- and gamma-flux distributions, power distributions, etc. Changing the simulated core loading allows experimental investigation of the reactivity impacts of postulated accident configurations or other off-normal events. Additionally, heated elements may be used to examine the temperature feedback of particular materials.

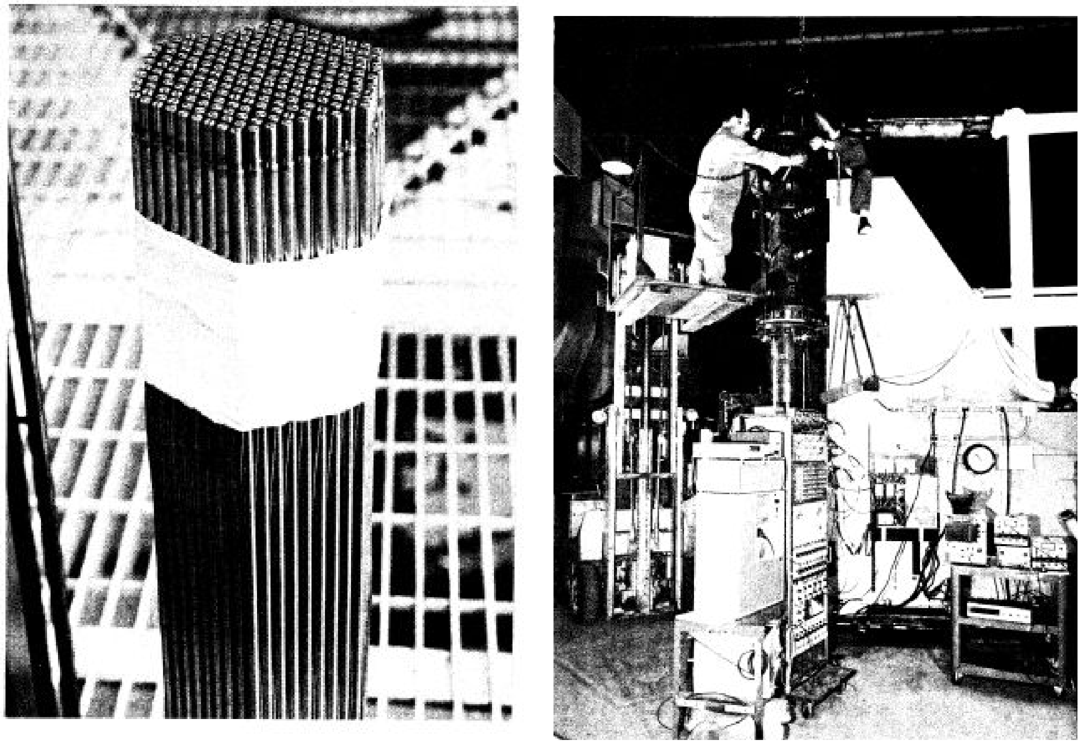
Extensive critical experiments have been conducted for LMRs, from very small designs, through the CRBR design, to potential commercial LMR designs with a variety of homogeneous and heterogeneous core configurations. The critical experiment data for SEFOR and FFTF provided especially accurate and useful data for these plants.

Thermal Hydraulics

Another illustrative technology-validation area addresses the coolant's thermal hydraulic behavior. It is necessary to have accurate prediction of the heat transfer from the fuel pins to the coolant and the transport of heat within the vessel and throughout the PHTS and IHTS. Of particular interest is assuring adequate flow distribution within and among the fuel assemblies to validate the flow orifice design. The sodium coolant can also potentially cause flow-induced vibration and wear to the fuel pins, which are spaced by wire wraps and include some clearance to facilitate loading. The fuel assembly design must preclude premature pin failure from this phenomenon. Sodium can also induce high cycle fatigue by thermal cycling. This could occur if sodium streams of differing temperatures impinge on a structure before fully mixing. Thermal stratification of sodium in the vessel is another potential design issue.

The LMR Base Technology Development program conducted extensive analytical methods development and experiments to address these issues and provide sound design guidance to assure acceptable performance for normal and postulated off-normal conditions. An example of the type of tests that were performed is shown in Figure C.49.

In this test sodium was pumped through the simulated fuel assembly at varying rates and the vibration was measured by accelerometers. Different wire wrap spacing parameters were used to understand the conditions for onset of unacceptable vibration. The results were then factored in to the fuel assembly design, which was documented in FFTF SDD-31.



Simulated Fuel Assembly Vibration Flow Test Rig

Figure C.49 FFTF fuel assembly flow vibration tests

Another example of the LMR thermal hydraulic validation testing is illustrated in Figure C.50, a schematic of the Core Flow Mockup, which was fabricated and run to assure satisfactory coolant flow within the vessel and components of the CRBR design.

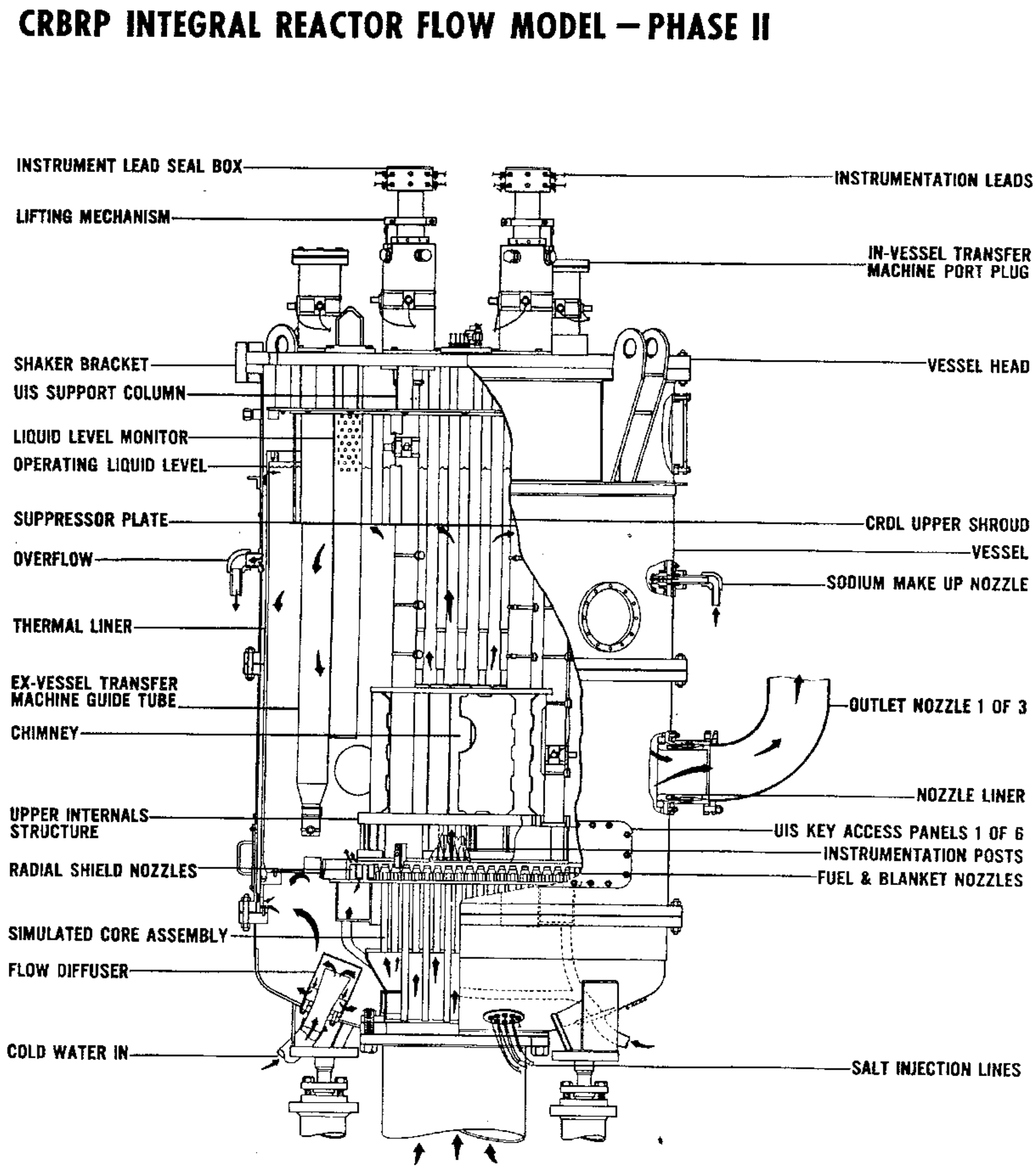


Figure C.50 CRBR core flow mockup

A final example of LMR thermal hydraulic testing technique is shown in Figure C.51. This example comes from the SP-100 space reactor program which developed LMR technology for a very-high-temperature LMR to be used for nuclear electric propulsion in space.

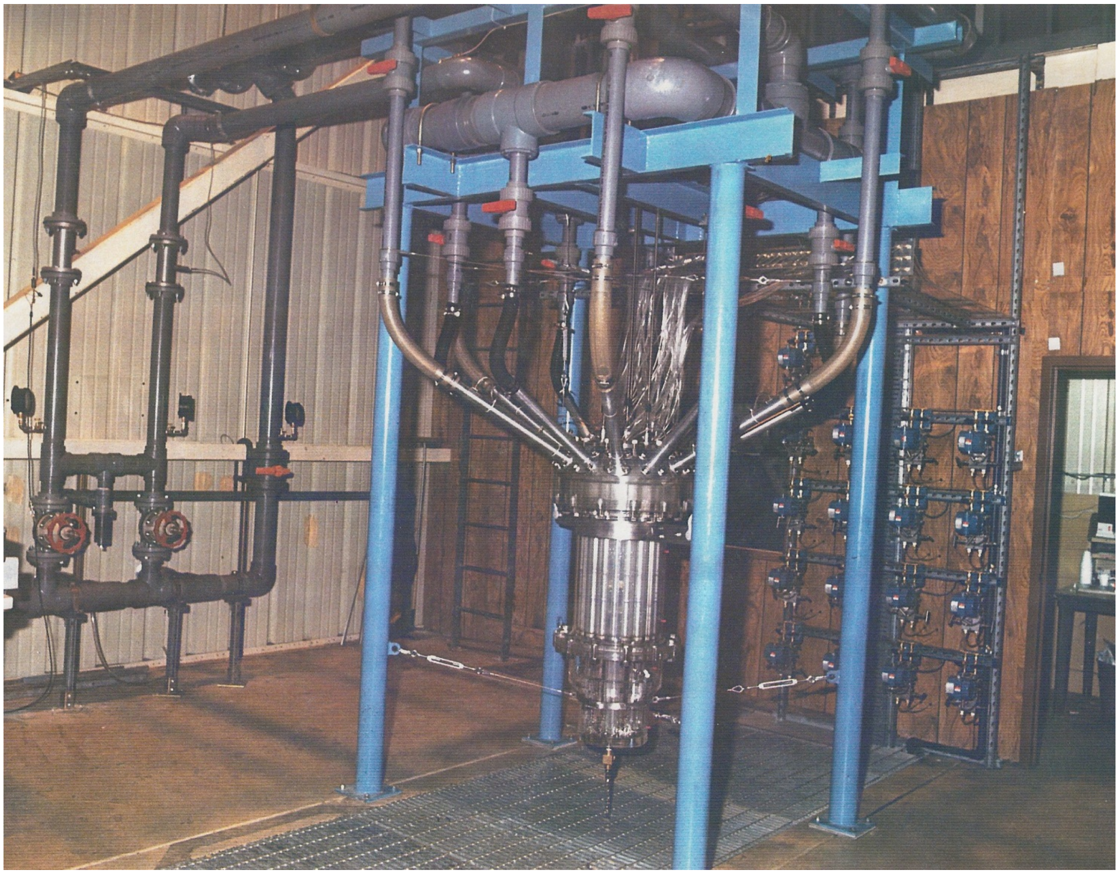


Figure C.51 Core flow test with water (SP-100 LMR)

In this test, water at room temperature is used to simulate the liquid metal coolant at operating temperature. The similitude between these two fluids produces very similar behavior and therefore permits an unusual testing methodology. The simulated reactor vessel and internals are fabricated of relatively transparent material to allow viewing of the flow field within the vessel. The water is seeded with a small amount of very small glass particles and a laser Doppler velocimeter is used to map out the flow distribution with varying total flow rates. This allows validation of the effectiveness of the flow orificing among the fuel assemblies and across the plenums.

Summary

The unmoderated fast neutron energy spectrum in LMRs results in several important differences in the reactor core performance and design constraints compared to LWRs. These differences are well understood and the data and design methodologies needed for successful LMR projects are available in several countries, including the United States. The most important aspect of LMRs from a strategic energy perspective is the flexibility in the fuel cycle that results from specific core loading approaches. This allows an LMR to be used to:

1. recycle spent fuel from LWRs and increase the overall energy production by a factor of ~60 by net breeding of fissile material
2. significantly reduce the high-level nuclear waste challenge through recycling and burning of transuranium isotopes

The liquid metal coolant in LMRs allows a very compact core design with greater power density than LWRs. However, irradiation of sodium produces a radioactive sodium isotope and sodium reacts exothermically with water, which mandates an intermediate heat transfer loop between the reactor core and the steam generator. The plant may be configured in either a pool or loop arrangement and designs of both types have been successfully operated.

LMR work began in the United States in the mid-1940s and resulted in an LMR, EBR-I, generating the first nuclear electric power in 1951. Many countries instituted LMR programs and more than 20 LMRs have been constructed and operated around the world, resulting in more than 300 reactor-years of experience.

In the mid-1960s, the United States adopted a very disciplined LMR development approach, including an intensive Base Technology Development program and the creation of Reactor Development and Technology (RDT) Standards which covered research, design, fabrication, and operation of LMRs. The AEC/DOE, working with the U.S. reactor industry and national laboratories, developed a well-documented LMR technology database and design methodology. The intent of this system was to minimize the probability of errors in the final LMR. Although many of the LMR RDT Standards have been deactivated by DOE, they are largely recoverable with a modest effort.

The examples of technology validation for reactor core physics and thermal hydraulics in this paper illustrate the depth of the experimental investigations and the magnitude of test facilities required. During the peak of the U.S. LMR program, more than 3000 personnel were involved and a total of well over $10B (not at today's dollar values adjusted for inflation, but in dollar values at the time of expenditure) was spent in the U.S. LMR effort.

The strength of this process is evidenced by the excellent performance of FFTF, which was the world's largest test reactor of its kind. During its 12 years of successful operation, FFTF tested a wide range of nuclear fuels, materials, and systems equipment with very minimal operational problems.

Although the U.S. LMR program peaked by the early 1980s and only significantly reduced LMR tasks have continued since that time, the LMR data, standards, and design processes developed are still largely applicable to current and potential future LMR activities. This is a timely consideration because there is renewed global awareness of the key role that LMRs must play in the energy economy if society is to adequately address the rapidly growing energy demand, to meet the need to minimize long-lived nuclear waste, and to resolve the environment/energy dilemma. It is certainly advisable for the United States to make the most beneficial use of our very sizable LMR investments in addressing these issues.

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APPENDIX D AGENDA FOR COURSE

SODIUM FAST REACTOR TECHNOLOGY COURSE

Sponsored by
Office of Nuclear Regulatory Research

Table D.1 Sodium Fast Reactor Technology Course Agenda

START TIME	TOPIC AREA	ESTIMATED TIME (minutes)
	Day 1 Agenda ***Dates: tbd Location: tbd***	
8:30 a.m.	Course Overview and Outline of Lesson Plans • Assumptions About the Course • Course Objectives • Organization of Technology Course – Modules – Quizzes – Timing • Module Areas • Course Agenda	30
9:00 a.m.	Module 1 – Introduction	30
9:40 a.m.	Module 2 – SFR Neutronics	60
10:50 a.m.	Module 3 – SFR Coolants and Thermal Hydraulics	35
11:30 a.m.	*Quiz on Introduction, Neutronics, Coolants, and Thermal Hydraulics Modules*	20
11:50 a.m.	LUNCH	70
1:00 p.m.	Module 4 - Fuel Characteristics	80
2:30 p.m.	Module 5 – SFR Systems and Components – Session 1	50
3:30 p.m.	Module 5 – SFR Systems and Components – Session 2	65
4:35 p.m.	Day 1 Wrap-Up, Summary, and Questions	10
4:45 p.m.	Adjourn Day 1	

Table D.1 (continued)

START TIME	TOPIC AREA	ESTIMATED TIME (minutes)
	Day 2 Agenda ***Dates: tbd Location: tbd***	
8:00 a.m.	Module 5 – SFR Systems and Components – Session 3	55
9:00 a.m.	*Quiz on Fuel Characteristics and Systems and Components*	20
9:30 a.m.	Module 6 – Safety and Accident Analysis – Session 1	60
10:40 a.m.	Module 6 – Safety and Accident Analysis – Session 2	30
11:20 a.m.	Module 7 – Licensing Issues – Session 1	40
12:00 a.m.	LUNCH	60
1:00 p.m.	Module 7 – Licensing Issues – Session 2	60
2:10 p.m.	*Quiz on Safety and Accident Analysis and Licensing Issues*	20
2:40 p.m.	Module 8 – Containment Systems	30
3:20 p.m.	Module 9 – Selected SFR Operating Experience – Session 1	60
4:20 p.m.	Day 2 Wrap-Up, Summary, and Questions	20
4:40 p.m.	Adjourn Day 2	

START TIME	TOPIC AREA	ESTIMATED TIME (minutes)
	Day 3 Agenda ***Dates: tbd Location: tbd***	
8:00 a.m.	Module 9 – Selected SFR Operating Experience – Session 2	60
9:10 a.m.	Module 9 – Selected SFR Operating Experience – Session 3	50
10:10 a.m.	Module 9 – Selected SFR Operating Experience – Session 4	45
11:05 a.m.	*Quiz on Containment and Selected Operating Experience*	25
11:30 a.m.	LUNCH	75
12:45 p.m.	Module 10 – Summary of PWR, 4S, and PRISM Characteristics – Session 1	50
1:45 p.m.	Module 10 – Summary of PWR, 4S, and PRISM Characteristics – Session 2	50
2:45 p.m.	*Quiz on Summary of PWR, 4S, and PRISM Characteristics*	30
3:15 p.m.	Day 3 and Course Wrap-Up, Summary, and Questions	60
4:15 p.m.	Adjourn Day 3 – Course Completed	

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APPENDIX E CODE EXAMPLE

SAS4A

E.1 NAME OF PROGRAM

SAS4A

E.2 COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHER MACHINE VERSION PACKAGES AVAILABLE

Mainframe (IBM, Cray Inc., Control Data Corporation (CDC), etc.), UNIX workstation (Sun Microsystems, IBM RISC, Hewlett-Packard (HP), or Silicon Graphics (SG)), or personal computer (IBM PC compatible) with FORTRAN compiler.

E.3 DESCRIPTION OF PROBLEM SOLVED

SAS4A is designed to perform deterministic analysis of severe accidents in LMRs. Detailed, mechanistic models of steady-state and transient thermal, hydraulic, neutronic, and mechanical phenomena are employed to describe the response of the reactor core and its coolant, fuel elements, and structural members to accident conditions caused by loss of coolant flow, loss of heat rejection, or reactivity insertion. The initiating phase of the accident is modeled, including coolant heating and boiling, fuel cladding failure, and fuel melting and relocation. SAS4A analysis is terminated on loss of subassembly hexcan integrity. The objective of SAS4A analysis is to quantify severe accident consequences as measured by the generation of energetics sufficient to challenge reactor vessel integrity, leading possibly to public health and safety risk. Originally developed for analysis of sodium-cooled reactors with oxide fuel clad by stainless steel, the models in SAS4A were subsequently extended and specialized to metallic fuel clad with advanced alloys.

E.4 METHOD OF SOLUTION

In space, each SAS4A channel represents one or more subassemblies with either a single-pin model or a multiple-pin model. Many channels are employed for a whole-core representation. Heat transfer in each pin is modeled with a two-dimensional (r/z) heat-conduction equation. Single- and two-phase coolant thermal hydraulics are simulated with a unique, one-dimensional (axial) multiple-bubble liquid metal boiling model. The transient fuel and cladding mechanical behavior model, integrated with fission product production, release, and transport models, provides prediction of fuel element dimensional changes and cladding failure. Fuel and cladding melting and subsequent relocation are described with multiple-component fluid dynamics models, with material motions driven by pressures from coolant vaporization, fission gas liberation, and fuel and cladding vaporization. Reactivity feedbacks from fuel heating (axial expansion and Doppler), coolant heating and boiling, and fuel and cladding relocation are tracked with first-order perturbation theory. Reactivity effects from reactor structural temperature changes yielding radial core expansion are modeled. Changes in reactor power level are computed with point kinetics. Numerical models used in the code modules range from semi-implicit to explicit. The coupling of modules in time is semi-explicit within a multiple-level time-step framework.

E.5 RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEMS

In any channel, there are maximums of 24 axial heat-transfer nodes in the core and axial blankets and 49 axial coolant hydraulics nodes. The number of channels is limited only by the size of the computer memory.

E.6 TYPICAL RUNNING TIME

Running times depend on the complexity of the model and the physical phenomena being analyzed. A few-channel reactor model using only pin-heat transfer, single-phase coolant dynamics, and reactor-point kinetics physical models will generally run orders of magnitude faster than real time on modern computing hardware. A many-channel model using two-phase coolant dynamics and fuel melting and relocation physical models takes significantly longer, with running times that depend on problem complexity.

E.7 UNUSUAL FEATURES OF THE PROGRAM

The physical models in SAS4A are highly detailed numerical representations of reactor accident conditions based on extensive laboratory and test reactor results. The models are specialized to liquid-metal (sodium) cooled fast reactors with oxide or metallic fuel clad with stainless steel.

E.8 RELATED AND AUXILIARY PROGRAMS

Many of the reactor core and coolant loop thermal hydraulic models in SAS4A are shared with the SASSYS-1 computer code.

E.9 STATUS AND AVAILABILITY TO THE NRC

SAS4A Version 3.1 is available for production use at Argonne National Laboratory in the Nuclear Engineering Division. Earlier versions have been exported to domestic U.S. DOE contractors and to research organizations in foreign countries. The SAS4A/SASSYS-1 code package continues to undergo development in response to advanced fast reactor simulation needs.

E.10 STATUS OF VERIFICATION AND VALIDATION (V&V)

The SAS4A code has been verified and validated against other analyses of severe accidents in sodium-cooled reactors. Integral tests of core disruption accidents in sodium fast reactors are not available. The EBR-II tests provide validation for initiating events that did not lead to core damage.

E.11 STRENGTHS OF THE CODE

SAS4A and its precursors have been extensively used for the analysis of severe accidents in sodium-cooled reactors. As such, it is the leading code for this purpose.

E.12 WEAKNESSES OF THE CODE

The code has not been compared against total core disruption accidents because test data is not available for these conditions. The code does not address the core conditions while still in the as-designed configuration; it is coupled with SASSYS-1 for this purpose. Also, the code does not assess the core melt cooling or effects on containment systems from fission product loadings, for example.

E.13 OTHER CODES SIMILAR TO SAS4A

SIMMER is one code with similar objectives.

E.14 MACHINE REQUIREMENTS

The length of the combined SAS4A/SASSYS-1 executable on the Sun Microsystems UNIX system is about 7.2 megabytes, and a data buffer of about 200 kilobytes for each channel is required. Disk storage for potentially large ASCII print and binary plotting data storage files is required.

E.15 PROGRAMMING LANGUAGES USED

Standard FORTRAN 77 is used. System-dependent routines may be supplied for dynamic memory allocation, timing, and system and user identification.

E.16 OPERATING SYSTEM

No special requirements other than a FORTRAN compiler and the usual linker/loader facilities.

E.17 OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS

The distribution of the SAS4A computer code and its documentation are subject to U.S. DOE Applied Technology regulations.

E.18 NAME AND ESTABLISHMENT OF AUTHOR OR CONTRIBUTOR

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E.19 MATERIALS AVAILABLE

1. FORTRAN Source Code
2. Example Problems Input Data and Printed Output
3. Five-Volume Technical Report for Version 3.0 containing detailed model descriptions and user guide

E.20 SPONSORS

U.S. Department of Energy, Office of Nuclear Energy, Science, and Technology.

E.21 REFERENCES

Cahalan, J.E., A.M. Tentner, and E.E. Morris, "Advanced LMR Safety Analysis Capabilities in the SASSYS-1 and SAS4A Computer Codes," *Proceedings of the International Topical Meeting on Advanced Reactors Safety, April 17–21, 1994*, *Pittsburgh, PA*, American Nuclear Society, La Grange Park, IL, pp. 1038–1045.

Cahalan, J.E., and T.Y.C. Wei, "Modeling Developments for the SAS4A and SASSYS Computer Codes," *Proceedings of the International Fast Reactor Safety Meeting, Snowbird, UT, August 12–16, 1990*, American Nuclear Society, La Grange Park, IL, pp. 123–132.

APPENDIX F PAPER ON ADVANCED REACTOR RESEARCH PLAN—INPUT FOR LMR

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July 2006

KEY RESEARCH AREAS

1. Reactor Systems Analysis

This paper will address infrastructure needs for liquid-metal-cooled reactors (LMRs) in the area of reactor systems analysis, which includes thermal hydraulic (T/H) analysis, nuclear analysis, and severe-accident and source-term analysis. Accidents considered for analysis will include events that fall within the licensing basis (design-basis events) and severe accidents (beyond-design-basis events).

(1A) Thermal-Hydraulic Analysis

Background

LMRs use a liquid metal (usually sodium, lead, or a mixture of lead and bismuth) as the primary coolant. Heat from the liquid metal is transferred to water to produce steam in a liquid-metal-to-water heat exchanger. LMR designs are basically of two types: (1) a loop-type design, in which the primary coolant system (piping, pumps, and heat exchangers) are located in a compact loop layout outside the reactor vessel, and (2) a pool-type design, in which the primary coolant system is located inside the reactor vessel. For both options, the primary coolant has a relatively large thermal inertia. A large margin to coolant boiling is achieved by design and is an important safety feature of these systems. LMRs generally operate in the 480 to 540°Celsius © (900 to 1,000°Fahrenheit (F)) coolant outlet temperature range, well below the sodium boiling point of 900°C (1650°F). Key safety features of LMRs are the high thermal conductivity and boiling point of the liquid metal coolant (which results in promoting heat removal through conduction and natural circulation without the complications of a two-phase coolant) and the ability to operate at essentially atmospheric pressure (which reduces primary stresses and lowers the potential for coolant leaks). (Ref.1)

In other words, the sodium coolant is a highly efficient heat-transfer material and has the additional advantage of operating at normal atmospheric pressure. In the typical commercial light-water reactor, the water coolant must be kept at 100 to 150 times normal pressure to keep it from boiling away. But sodium can cool the core at normal pressure, because its boiling point is 300 to 400°C (575 to 750°F) higher than the core's operating temperature. The sodium pool minimizes the possibility of the coolant boiling away during an accident and leaving the core uncovered, which is one of the more serious potential trouble spots in a light-water reactor. By submerging the core in thousands of gallons of liquid sodium, one provides the reactor with an immense heat sink that adds greatly to its safety. If the reactor starts to overheat, the pool can absorb vast amounts of heat and yet stay below its boiling point. (Ref.2)

Other key features found in LMRs: LMRs do not have a traditional emergency core-cooling system (ECCS). Rather, they employ one or more secondary vessels, called guard vessels, that fit around the reactor vessel (and for loop plants also fit around the primary system pumps and heat exchangers) to catch and retain any leaking coolant. For loop plants, the primary piping is also elevated to ensure it is not a low point in the system that could cause coolant draining. LMRs generally rely on natural convection to remove decay heat. Sodium reacts chemically with air and with water, so the design must limit the potential for such reactions and their consequences. To improve safety, LMRs using sodium as a coolant generally employ an intermediate sodium system which acts as a buffer between the primary radioactive sodium and the steam or water in the tertiary system, to prevent radioactive primary sodium from reacting with water in the event of a steam generator tube leak. The intermediate coolant system is operated at a higher pressure than the primary coolant system to prevent primary radioactive sodium from entering the intermediate system. In a reactor using lead or lead/bismuth as the coolant, an intermediate loop might not be used because those coolants do not react chemically with water, although molten lead coming in contact with water could result in a steam explosion. (Refs. 1 and 3).

Purpose

Several of the major T/H issues for the LMR are highlighted below. Many of them are related to the Advanced LMR (ALMR) design based on the PRISM concept that uses sodium as the liquid metal coolant (Refs. 4 and 5) but would apply to other LMR designs using the same systems and components. Because several reactors using sodium are already in existence, a large experience base exists for a sodium-cooled reactor system.

- **<u>Demonstration of Passive Safety Design</u>**. The physical phenomena and design features that are relied on to achieve passively safe response to design-basis transients and anticipated transients without scram should be adequately characterized. An example is axial thermal expansion of the fuel and radial expansion of the core grid plate structure. Research and development to evaluate these physical phenomena and design features and validate their models through experimentation would involve in-pile experiments using a transient test facility (Ref. 6). Assurance of passive safety response, including modeling and the validation of models through experimentation, is an important technology issue (Ref. 7).
- **<u>Electromagnetic (EM) Pumps</u>**. As an example, the ALMR design employs EM pumps that are self-cooled by the surrounding sodium. They are unique to nuclear power plants. They are constructed of a series of coils wrapped in insulation that generate an oscillating magnetic field. Because sodium is an electrical conductor, its movement in the magnetic field creates a pumping force on the sodium. Because it has no moving parts, an EM pump has no coastdown to maintain a safe power-to-flow ratio when electrical current to the coils stops. For this reason, a simulated coastdown is forced on the EM pump by running a synchronous machine in parallel with it. If power is lost, the synchronous machine, which can be thought of as a flywheel and generator combination, will provide a prescribed voltage and current to the EM pump to simulate a coastdown. The coastdown should maintain a power-to-flow ratio in the fuel sufficient to maintain large safety margins for the peak clad and fuel temperature. The response of the EM pumps during a loss-of-flow event is crucial to the outcome from these events.

Two major issues with the EM pumps need a database to accurately predict design-basis accidents (DBAs). These are areas that need a computer model and a database for the phenomena (Ref. 5).

1. The EM pumps, if they are used as the primary pumps in an LMR design, will be crucial to its operation. A sudden loss of pumping ability from a coil failure could lead to excessively high fuel temperatures and/or sodium boiling, which could in turn lead to large reactivity insertions. A prototypical test is needed to demonstrate that the coils that make up the EM pump have the projected life and

reliability in terms of irradiation damage to the coils and the performance of the material insulating the coils.

2. The coastdown curves used in the ALMR Preliminary Safety Information Document (PSID) for the EM pump for all unprotected loss-of-flow (ULOF) events are calculated values. A database is needed to validate these coastdown curves. This would be the case for any other LMR design in which EM pumps are used.

- **Flow in the Upper Internal Structure (UIS)**. The control rod drive line thermal expansion feedback is a significant negative reactivity feedback that plays a major role in several of the DBAs (unprotected transient overpower (UTOP), unprotected loss of heat sink (ULOHS), and unprotected loss of flow (ULOF)). The flow paths in and around the UIS that flow past the control rod drive line are still an open issue. Data are needed to substantiate the flow rate and heat transfer to the control rod drive line during normal and off-normal conditions. The resulting thermal expansion of the drive line that inserts the control bundle into the core needs to be characterized as a function of position relative to the time in the fuel cycle.
- **The Ultimate Shutdown System**. The last revision of PRISM incorporated an alternate scram system called the "Ultimate Shutdown." The system is essentially a box of many small spherical boron carbide balls suspended above the hollow, central fuel assembly or channel of the core that has a small bypass flow through it. When activated by the operator, the bottom lid drops down and the boron carbide balls fall into the channel. This is supposed to function no matter what the geometry of the channel and have enough negative reactivity to terminate the fission power. The concept will need a proof-of-principle demonstration test and a database to provide estimates for its activation time and rate of reactivity insertion during an event.
- **Sodium and Water Representation (Two-Fluid)**. The steam generator (SG) tubes of a sodium-cooled fast reactor (SFR) are the boundary between the secondary sodium in the intermediate heat transport system (IHTS) and the higher-pressure steam system. If the steam generator has some tube failures, water and sodium may be found together in the IHTS. When the two fluids meet, an exothermic reaction would occur which could result in the failure of the intermediate pipe or, in the worst case, the failure of the intermediate heat exchanger (IHX). Because failing the IHX would open the primary containment (i.e., the reactor vessel), a model should be developed to predict the sodium/ water behavior under these conditions and the extent of IHX pressurization that develops from this and whether it provides adequate margin from damage. Similar considerations should also be given to lead-water reactions in the case of a lead-cooled reactor.
- **Leak Detection, etc**. One approach to dealing with the sodium-water reaction issue in the SG would be to base the design on the successful approach used at Experimental Breeder Reactor-II (EBR-II) and employ a double tube wall. EBR-II had no tube leaks in about 30 years of operation. The sodium would flow on the outside of the tubes, while the high-pressure steam would be on the inside. A small gap would be left between the two tubes, which could be filled with a porous wire mesh and helium. A leak-detection system would monitor both the gap and the shell-side sodium. Moisture in the helium would indicate an inner tube failure, while helium in the sodium would indicate outer tube failure. A sodium-water reaction could occur only if both tubes failed. A sodium dump system would actuate in such an event to alleviate this potentially damaging event. The Toshiba 4S design proposed for the village of Galena in Alaska has used this

approach.[8, 9] For more information on sodium reactions with air and water and leakage detection, etc., see the section on “Materials” under “Sodium.”

- **Two-Phase Sodium**. If sodium boiling is expected to occur during a transient, a model is needed that can track the boiling location and extent. This will impact the heat transfer within the assemblies and the local reactivity insertion caused by the void generation. To evaluate events with sodium boiling, a two-phase sodium boiling model needs to be represented in the code with the appropriate constitutive package for bubble size, interfacial shear, interfacial heat and mass transfer, and two-phase friction multipliers.
- **Multidimensional Upper Plenum**. The need for a two- or three-dimensional thermal hydraulics model for the upper plenum during an accident calculation is not clear at this time. If the UIS which supports the control rod drive lines and in-vessel refueling machine has a complex flow path to direct and wash the control rod drive lines with sodium from hot driver channels, a two-dimensional thermal hydraulics model would be needed in the upper plenum.
- **Reactor Vessel Auxiliary Cooling System (RVACS)**. Under normal operating conditions, the sodium level in the air jacket between the reactor vessel and the vessel liner is fairly low, so during normal operation only a small fraction of the reactor’s generated heat is transferred to the air jacket surrounding the reactor vessel. However, once a postulated loss-of-flow event begins and the pumps are tripped, the sodium level in the air jacket increases until it matches that in the upper plenum. If the normal paths to reject heat through the IHX are lost, the decay heat can be rejected to the outside air through the RVACS. Besides the increased conduction through the liquid metal, as the primary sodium heats up, its density decreases, causing the sodium to swell and flow over the vessel liner. This results in increased heat rejection to the RVACS by means of forced convection rather than just conduction through the sodium. On the air side, the increased heat rejection to the air increases its mass flow rate, which allows further increase in heat rejection.

  This is both a model component that needs to be developed for computer codes analyzing an LMR event and a phenomenon that needs a database to establish its performance.
- **Upper Plenum Sodium Level Tracking**. A model to track the sodium level in the upper plenum is required for simulation of LMRs that use the RVACS for passive cooling of the reactor vessel. During events in which the RVACS is required, the sodium level swell in the upper plenum determines when the liner spillover begins. The entry of sodium into the RVACS greatly increases the heat transfer (and thus heat rejection) in the RVACS. If this phenomenon is not properly modeled, the temperatures in the vessel and core would not be accurately known.
- **Auxiliary Cooling System (ACS).** The ACS is based on natural-circulation air-cooling of the steam generator. An air jacket surrounds the steam generators with a set of dampers at the inlet. During an event in which the water loop is lost but the primary and intermediate loop are available and continue to transfer heat to the SG, this heat can be rejected by air cooling with the ACS. While this may not be a safety-grade system and thus should be assumed not to function during many events, a model should be developed to analyze the system performance with it operating.

- **Metal Mass Temperature Model (Thermal Mass)**. The temperature and thermal expansion of several components during transient heat-up (such as those of the reactor vessel, control rod drive, above-core load pads, and lower core grid plate) are crucial and must be tracked in any calculation.
- **Natural Circulation Model**. The thermal hydraulics code must be able to calculate the flow rates associated with natural circulation in LMRs. The flows are driven by small density differences and are in the laminar regime. The Super System Code (see "Related NRC research" in the next section) has been assessed in this area for its ability to successfully make these calculations (Refs. 10 and 11).
- **Forced Circulation Model**. The thermal hydraulics code must have the appropriate models for heat transfer and friction factors in the higher Reynolds number regions associated with forced flow.
- **Balance-of-Plant (BOP) Model**. The BOP in an LMR is the tertiary loop in the system that contains the steam generator, feedwater pumps, piping to the turbine, and control system. This part of the system is not safety grade and is usually assumed not to be available during any accident. However, these models will be needed in a computer code to help understand how the system will respond as a whole during normal operation. In the Super System Code (SSC) series, the MINET code has models for these components and interfaces with SSC during any calculation when these models are activated (Ref. 12).
- **Fuel Assembly Heat Transfer**. A computer code would need models for the flow and heat transfer within a fuel bundle. Extensive work has been performed in this area for EBR-II, the Fast Flux Test Facility (FFTF), and the Clinch River Breeder Reactor (CRBR) (Ref. 13). The Super System Code has models to represent these heat transfer phenomena (Refs. 10 and 11).
- **Intermediate Heat Transport System (IHTS)**. This system containing non-radioactive sodium is between the primary (radioactive sodium) system and the water loop where steam is produced. The effect that the IHTS has on the core inlet temperature makes its modeling crucial for many transients.

Objectives and Planned Activities

Related NRC research

The Super System Code (SSC) series, comprising "SSC-L" for loop-type LMRs (Ref. 10) and "SSC-P" for pool-type LMRs (Ref. 11), was developed by Brookhaven National Laboratory (BNL) for the NRC in the late 1970s. This code series has many of the models required to evaluate LMRs. The code needs to be revisited to update its models and add new models (such as a two-fluid model in the IHTS, two-phase sodium model, multidimensional model for the upper plenum if needed, models for EM pump, RVACS, ACS, etc.). The Natural Convection Shutdown Heat Removal Facility (NSTF) at Argonne National Laboratory (ANL) has been used to demonstrate the concept of a passive decay heat removal system for LMRs and to validate code models.[14] Further experiments should be conducted to enhance our confidence in the performance of such passive cooling systems (Ref. 15). For the multidimensional upper plenum, water simulation tests with a 1/5-scale model using laser technology have been used for flow visualization. These tests and other work done in this area need to be reviewed.

Related international research

Identified research activities: NRC needs an independent capability for LMR T/H analyses that has been thoroughly assessed and peer reviewed. Whether international research effort will be focused on adding the necessary capability for LMR analysis to SSC is yet to be determined. Some work has been done in Korea on its version of SSC called SSC-K for simulation of their KALIMER SFR design."

Application of Research Results

This research will be applied to develop and demonstrate the ability to predict the behavior of the new LMR plant designs under normal and accident conditions. Results from the research activities described above will be applied to enable and support the staff's independent assessment of T/H issues associated with the respective advanced reactor designs.

As outlined in the preceding sections, the T/H research activities will result in developing the staff's technical insights in these areas and applying those insights toward establishing and qualifying independent analysis tools and capabilities. The development activities include the assessment of validation issues and modeling approximations, validation of success criteria, input into probabilistic risk assessment (PRA), and understanding of safety margins.

(1B) Nuclear Analysis

Background

One of the most significant design goals for the LMR would be to provide sufficient negative reactivity feedback to withstand "failure to scram" events without fuel damage. This would come under the passive safety system design of the LMR. For example, for a transient event that involves a failure of the reactor scram system together with a failure of the heat removal system (e.g., failure of the IHTS pumps, or a reduction in feedwater flow to, or steam from, the steam generator), the primary liquid metal system would heat up without insertion of negative reactivity by an active system. The resulting thermal expansions should tend to reduce core power as the coolant and core heat up. This type of response is sometimes referred to as "inherent reactor shutdown characteristics."

Purpose

Some of the research issues for the LMR nuclear analysis are highlighted below. The issues currently listed focus primarily on passive reactivity reduction, which is composed of several reactivity feedback properties.[4, 5]

- **Doppler Effect**. As the fuel temperature rises, the fuel captures more neutrons, which has the effect of removing active neutrons from the core and reducing reactivity. Fuel temperature rises as a result of power excursions, but Doppler feedback removes reactivity as the temperature rises and can thus help to limit the extent of power increase excursions. As fuel temperature drops with power reduction, the Doppler effect adds reactivity and tends to increase core fission power.
- **Positive Void Worth**. Should sodium boiling occur, the sodium thermally expands, creating voids where there are fewer sodium atoms. The dominant effect of this is to reduce the collisions between neutrons and sodium atoms, which increases the average neutron energy and results in a positive reactivity insertion. A smaller effect is that fewer neutrons are scattered back into the core. This increases leakage of neutrons around the core periphery, hence a small negative feedback effect. For a small LMR such as the EBR-II, the overall effect is a negative reactivity feedback caused by dominance of leakage effects, and is helpful. For larger LMR designs, this is a positive feedback. Should sodium boiling begin on a corewide basis under failure-to-scram conditions, the

feedback would be very large (approximately $5 positive reactivity for total core void for the PRISM LMR design, for example). The reactor under such circumstances would be likely to experience a severe power excursion and a potential hypothetical core-disruption accident (HCDA).

It turns out that in most safety analyses, these coolant voiding scenarios are beyond the design base (which in the United States means a probability of less than 10^{-6} per reactor year), but most fast reactor analysts believe that HCDAs will have to be analyzed anyway as part of a licensing process. There is, therefore, interest in reducing the void reactivity, and in developing passive means to mitigate the effects (Ref. 16).

- **Axial Fuel Expansion**. Axial expansion of fuel before failure will remove reactivity and turn a reactivity-insertion-driven overpower transient. The magnitude of the effect is quite different for oxide and metal fuels.[3] Metal fuel expands significantly when it heats up. Axial expansion within the clad increases the core size and decreases the effective density of the core materials. This increases the probability that neutrons will escape from the core, creating a significant negative reactivity feedback. Fuel axial expansion and the Doppler effect are the dominant negative feedbacks for metal fuel, with fuel axial expansion being slightly more negative at all power levels.[5] The magnitude and dynamics of fuel expansion over a range of conditions expected for design-basis and beyond-design-basis postulated initiating events (PIEs) must be investigated to support modeling and code validation. Some data are available from Integral Fast Reactor (IFR) Transient Reactor Test Facility (TREAT) experiments. Additional experiments are needed to extend the range of data and investigate the margins to failure.[3]
- **Radial Expansion**. The radial dimension of the core is determined largely by the assembly spacing, which in turn is determined by the grid plate below the core. When the structures heat up and expand, the core expands radially and the core density reduces, which increases neutron leakage and reduces the net reactivity.
- **Control Rod Drive Line Expansion**. The control rod drive lines, which are fixed in the upper internal structure (UIS), expand downward when they are heated. This inserts the control rods further into the core and adds negative reactivity. The component needs a model for the reactivity associated with this effect.
- **Reactor Vessel Expansion**. Because the control rod drive lines are attached to the top of the vessel and the reactor core attaches to a point much lower along the vessel wall, the expansion of the vessel wall as it heats up pulls the control rods out. This is a positive feedback, but it occurs much later because of the slowness of the entire vessel wall to expand, hence is not an early concern.
- **Gas Expansion Modules (GEMs)**. GEMs are simple devices, resembling large inverted test tubes, containing a trapped region of inert gas above the core under normal operating conditions. They are placed in the perimeter of the reactor to facilitate leakage of neutrons when needed. Under full-flow conditions, the gas in the tube is compressed so that sodium occupies a portion of the GEM that resides within the active core region and traps the gas in the GEMs above the core. When the pumps stop and the system dynamic pressure falls, the gas region expands into the core region, speeding the decrease in reactor power through increased leakage (escape) of neutrons from the core.

  The GEMs have been demonstrated in the FFTF (Refs. 5 and 17) and have been shown to passively insert negative reactivity whenever system pressure is lost relative to operating conditions. The worth of this device must be established from experiments and a database established for both its reactivity worth and its reliability.

- **Changes in Reactivity Caused by Burnup**. Changes in the reactivity effects of the fuel must be accounted for over the fuel cycle of an LMR.
- **Kinetics Model**. A core kinetics model is needed in a system code that evaluates the plant behavior during transients. The high-energy-spectrum LMR core could be represented by a point kinetics model as is done in the SSC series. Exceptions are when sodium voiding occurs; in such cases, spatial effects from the spectrum changes would require a multidimensional core model to be used.

Objectives and Planned Activities

For metallic fuel (for example, the ALMR fuel is U-27Pu-10Zr), the reactivity feedbacks over the life of the core have not been experimentally determined, nor have the ability of the feedbacks to transition the core to a lower power level by passive reactivity reduction been demonstrated. The developers of the ALMR have proposed criticality tests to be performed in the future on a prototype reactor to qualify the Doppler, axial expansion, radial expansion, and temperature feedbacks (from sodium and structure) of the core. These data are needed to predict the power response during anticipated operational occurrences, design-basis accidents (DBAs), and severe accidents (SAs). Integral transient tests involving passive reactivity have been performed in EBR-II for a small metallic core and in FFTF for a mixed oxide core.

The magnitude and dynamics of fuel axial expansion over a range of conditions expected for design-basis and beyond-design-basis postulated initiating events (PIEs) must be investigated to support modeling and code validation. Some data are available from IFR TREAT experiments. Additional experiments are needed to extend the range of data and investigate the margins to failure.[3]

Regarding the length effect on axial fuel expansion, most of the data have been collected in the EBR-II, which had a fueled region that was only about 0.343 m (13.5 inches) long and had an axial profile along the core that was basically flat. A database is needed to establish the behavior of fuel pins that are longer (e.g., for ALMR they are 1.346 m or 53 inches long) with a cosine power shape.

The reactivity feedback of GEMs in the LMR has not been fully evaluated. A few experiments were performed in the FFTF to provide proof of principle and integral feedback measurements from a reactor system during unprotected loss-of-flow (ULOF) events (Ref. 17). However, experiments are required to establish the worth of the GEMs. Data on the worth of the GEMs as a function of sodium level in the device are needed over a range of flow rates and temperatures over the life of the core, to enable analysis of ULOF events.

Related NRC research

Relevant past, ongoing, and associated NRC research efforts include the following: Development of the SSC series of codes and MINET under NRC sponsorship by BNL, and making code modifications to enable simulation of the special features and phenomena of LMRs (Refs. 10 and 12).

Related domestic and international cooperation

Opportunities for LMR-related domestic and international cooperation include the following:

To fill technology gaps above and beyond an applicant's responsibility, RES could:

Identified research activities

Listed below are the potential research and infrastructure developmental activities pertaining to the nuclear analysis issues described previously.

Application of Research Results

Fundamental to reactor safety analysis is the ability to predict the fission and decay heat sources that arise under credible normal and accident conditions. Results from the research activities described above will be applied to enable and support the staff's independent assessment of nuclear analysis issues associated with the respective advanced reactor designs.

As outlined in the preceding sections, the nuclear analysis research activities will result in developing the staff's technical insights in these areas and applying those insights toward establishing and qualifying independent analysis tools and capabilities.

(1C) Severe-Accident and Source-Term Analysis

Background

This section highlights (1) the issues and database needed related to the inherent behavior of the fuel during severe accidents, (2) the database needed to represent the fission product release and transport so that radiological assessments of severe accidents can be determined (other issues related to fuel behavior that are applicable to design-basis events and severe accidents are highlighted under "Fuels Analysis"), and (3) characterization of the safety margins and response of the LMR containment system.

Purpose

Some of the research issues for the LMR Severe Accident and Source Term analysis are highlighted below.

- **Retention Factors for Isotopes.** Data are needed on retention factors for isotopes in large sodium pools for the elements that are crucial to predicting the source term from the LMR core.
- **Fuel Failure Mechanisms.** Data are also needed on characterization of the failure mechanisms for the prototypical fuel under severe accident conditions.
- **Fuel Melt Dispersal Behavior.** If, during a severe accident, the fuel in a pin were to melt and fail the clad, the behavior of the melt in the fuel bundle must be determined. Likewise, the behavior of molten metal fuel ejected from a pin into an assembly during a severe accident should be analyzed. Data are needed on its migration through the system, freezing within an assembly, and expected locations of fuel debris concentrations. The fuel could refreeze in the bundle and cause other pins to fail from lack of adequate cooling or freeze elsewhere in the system. Because a sodium-cooled LMR is not designed to operate in its most reactive configuration, any fuel relocation may result in a supercritical mass (Ref. 4). Thus, locations of fuel debris could be important for recriticality issues. Hence, a database is needed on the behavior of metal fuel under melt conditions.
- **Core Melt Composition.** If in a severe accident the fuel melts and drains to the grid plate, what is the expected composition of the material and its attributes? A database

for this behavior must be compiled to review barriers that might protect the reactor vessel and to determine how vulnerable the core is to such behavior.

- **Fuel Failure Non-Propagation**. A database is needed to demonstrate that local fault-induced failures do not propagate in the LMR core. Intra-assembly failure propagation, induced by pin breach or flow blockage, must be established with a database. Furthermore, assembly-to-assembly propagation must also be evaluated.
- **In-Vessel Debris Coolability**. A database is needed to assess the mass of fuel/clad debris and its disposition in the primary coolant system following a postulated severe accident. Long-term coolability of debris is an important safety issue.
- **Extrusion**. Metal fuel during high power excursion is expected to have an inherent negative response called extrusion. Extrusion occurs when the fission gas pockets in the fuel pass the solidus temperature near the top of the fuel pin column and the column erupts and spews molten fuel up into the fission gas plenum of the fuel pin. The removal of fuel material from the core region is a strong negative feedback that would be relied on by a vendor to be an inherent mitigation response to severe accidents.

  Some of these data might have been experimentally determined from the IFR program. However, most of the required data have not been collected for prototypical fuel. TREAT experiments have been performed but not with prototypical fuel pins and high power ramp rates.
- **Fission Product Retention in Large Sodium Pools**. For cases during a severe accident in which the fuel has melted, a database is needed to determine what fraction of the available radionuclides will be released into the cover gas or atmosphere in the building. Essentially, experiments are required in order to identify the retention factors associated with metal fuel core covered by an extensive sodium pool.
- **Fission Products Generated from a Sodium Fire**. A database is needed to determine the retention of fission products in a burning sodium pool and the resulting radiological consequences for the surrounding area.
- **Containment Assessment**. A study should be conducted to determine the loads to the primary and secondary containments and determine the safety margins for their integrity under these loads. See further elaboration below.

Objectives and Associated Activities

A dominant feedback mechanism in an undercooling transient without scram will likely be axial and radial core expansion and bowing of subassemblies. Only very limited data on these phenomena are available, and analytical capabilities are also limited. Development of coupled thermal-hydraulic-structural analysis tools is needed, and innovative experiments might be needed to provide data for validation. However, the importance of various reactivity feedback mechanisms will depend on details of the reactor and system design, so it is not possible to identify definitive experiment needs at this time. Because of the complexity of the situation, experiments in a prototype reactor might be necessary to finally validate code predictions. This technology gap applies to both oxide- and metal-fueled systems.

In general, bounding events will produce core debris which must be cooled for the long term. If debris can relocate after an accident, it will be necessary to establish that the debris is coolable in its final location, such as in a debris bed in the inlet plenum. Questions of debris-bed coolability have been studied for both oxide and metal fuels, with much more extensive data available for oxide. Coolability of metal-fuel debris beds requires demonstration.

It is necessary to establish that individual fuel-element failures will not cause failures of adjacent fuel elements as a result of disruption of heat transfer and overheating of the adjacent fuel. There has been no evidence of pin-to-pin failure propagation during operation of metal-fueled systems. Modern analytical techniques, such as computational fluid dynamics (CFD) may be used to evaluate the question of short-term disruption of heat transfer. For the oxide-fueled system, there is experimental evidence from the EBR-II run-beyond-cladding-breach program that pin-to- pin failure propagation might occur under aggressive operating conditions if pins are run well beyond the time of detection of fuel release.

It is necessary to establish that chemical interaction between the coolant and fission products released on cladding failure is such that release can be detected, that cleanup can be performed, and that the potential for fission product release to the containment can be minimized. The chemical affinity of sodium coolant for fission products of high interest in this connection (i.e., iodine, cesium, and strontium) provides an important mechanism for mitigating fission product release (Ref. 3).

Containment assessment

Extremely low-probability combinations of events and non-mechanistically postulated situations may be used to test the containment design for licensing purposes and to assess residual risk. Generically, threats to containment could come from rapid internal energy release such as might be associated with an energetic recriticality, or from long-term pressurization of the containment from thermal or chemical interactions between fuel, sodium, and concrete producing non-condensable gases. Thus, it is necessary to establish the design requirements for the containment, including consideration of both long-term static pressure capability and the ability to withstand short-term dynamic loadings.[3]

Mechanisms for long-term containment pressurization include chemical interactions between core materials, sodium, and concrete or other containment materials. Areas for related R&D include: (1) evaluation of in-vessel coolability of core debris; (2) chemical interactions between core debris, sodium, and concrete; and (3) the consequences of sodium leaks and fires.

Related NRC and international research activities

To be reviewed.

Application of Research Results

To be reviewed.

2. Fuels Analysis

Background

Two fuel options exist for the SFR: (1) mixed oxide (MOX) and (2) mixed uranium-plutonium- zirconium metal alloy (metal). Both are highly developed as a result of many years of work in several national reactor development programs. Burnups in the range of 150 to 200 gigawatt-days per metric ton of heavy metal (GWd/tHM) have been experimentally demonstrated for both. Nevertheless, the experience and databases for oxide fuels are considerably more extensive than those for metal fuels (Ref. 18).

The current fuel cycle deployed in the United States and most other countries is a once-through cycle: nuclear fuel is fabricated from mined and enriched uranium, irradiated once in a reactor, and then eventually (planned) to be disposed of in a geologic repository. Open cycles have been deployed commercially for more than 30 years and have proven safe, environmentally sound, and economically attractive, but no geologic repositories have yet opened as the ultimate location for the disposal of wastes. Natural uranium contains 0.7 percent ^{235}U and 99.3 percent ^{238}U. It is enriched up to 5 percent ^{235}U for fresh light-water reactor fuel. Spent nuclear fuel contains about 95 percent uranium (mostly ^{238}U), more than 3 percent fission products, and less than 2 percent transuranics (neptunium, plutonium, americium, and curium). All actinides present in the spent fuel have potential value for energy generation.[19]

These open fuel cycles will not meet the long-term goals to sustain the world's increasing dependence on nuclear energy. This is because (i) they use only a small fraction (less than 1 percent) of the energy available in the original mined uranium; (ii) they discharge long-term radiotoxic elements (most importantly the transuranic isotopes) that must be contained for hundreds of thousands of years; (iii) the construction and licensing of geologic repositories for final disposal has been a difficult proposition.

These difficulties can be overcome by adopting a closed cycle, in which the irradiated fuel is reprocessed, and its constituent elements are separated into streams to be recycled into a reactor or in appropriate waste forms. The recycled fuel is then irradiated in a reactor, where certain long-lived fission products or transuranic isotopes are partially transmuted through neutron capture or fission into new isotopes. In fast reactors such as the SFR, fission is favored over capture, hence there is much more limited buildup of higher actinides. SFRs thus would use a closed fuel cycle to enable their advantageous actinide-management and fuel-use features (Ref. 19).

The options for fuel recycling are the advanced aqueous process (preferred for MOX fuel) and the pyroprocess (preferred for metal fuel). The technology base for the advanced aqueous process comes from the long and successful experience in several countries with plutonium and uranium recovery by extraction (PUREX) process technology. The advanced process proposed by Japan, for example, is simplified relative to PUREX and does not result in highly purified products. The technology base for fabrication of oxide fuel assemblies is substantial, yet further extension is needed to make the process remotely operable and maintainable. The high-level waste form from advanced aqueous processing is vitrified glass, for which the technology is well established.[18]

The pyroprocess has been under development since the inception of the Integral Fast Reactor program in the United States in 1984. When the program was cancelled in 1994, pyroprocess development continued in order to treat EBR-II spent fuel for disposal. In this latter application, plutonium and minor actinides were not recovered, and pyroprocess experience with these materials remains at laboratory scale. Important technology gaps are in the areas of (i) scale-up of the pyroprocess with demonstration of high minor actinide recovery, and (ii) development of oxide fuel fabrication technology with remote operation and maintenance (Ref. 18).

Mixed Oxide (MOX)

The development of mixed oxide fuel (PuO_2 , UO_2) was a cornerstone of liquid metal reactor programs around the world for over 20 years. Earlier, mixed oxide fuel testing was carried out in EBR-II, Rapsodie, Jōyō, and the Dounreay Fast Reactor (DFR). This was followed later by the demonstration of high-burnup mixed oxide fuel in the FFTF, Phénix, Monju, and Prototype Fast Reactor in the United States, France, Japan, and the United Kingdom respectively. The economic incentive for lower fuel cycle costs motivated a continuous improvement in the burnup capability of mixed-oxide fuel.

In the United States, three cladding materials have been employed with mixed oxide fuel: 20 percent cold-worked Type 316 stainless steel, a modified stainless steel alloy D9 with reduced irradiation swelling characteristics, and the very-low-swelling ferritic alloy HT-9. The latter exhibited no swelling caused by irradiation up to a fluence of 3×10^{23} N^0/cm^2. Similar alloys have been developed in Europe. Even with these improvements, the maximum fluence remains below the goal of some programs. The European Fast Reactor initiative, for example, sought a cladding fluence goal of 3.6×10^{23} N^0/cm^2.

There is similar pursuit of improved cladding materials in Japan, where the line of development centers around oxide dispersion-strengthened (ODS) ferritic steel. This is driven by the economic incentive of obtaining higher thermal efficiencies through higher coolant core-outlet temperatures. At core-outlet coolant temperatures of 530 to 550°C and cladding temperatures above 650°C, HT-9 has insufficient strength.

The response of mixed oxide fuel to off-normal events has been extensively examined in TREAT testing in the United States and in CABRI and SCARABEE in France. These tests provided data on fuel failure mechanisms, fuel motion during failure, and coolant channel blockage. The data were then used in developing and validating fuel behavior models, transient fuel performance codes, and integrated severe-accident codes Ref. 16).

There are few technical issues that impede deployment of mixed oxide fuel in sodium cooled fast reactors. The issue is optimization rather than feasibility. More transient tests with advanced mixed oxide fuel pins would be a technically welcome new addition (Ref. 16).

Metal

Metal fuel was the first fuel used in fast reactors. The simple fabrication of metal and metal alloys, the high thermal conductivity, and the relatively high fissile density all made metal fuel attractive to early reactor designers. The Experimental Breeder Reactors-I and –II (EBR-I and EBR-II), the DFR, and the Enrico Fermi Nuclear Generating Station Unit 1 (Fermi 1) all used metal fuel. The early metal fuel designs were not capable of achieving high burnup, nor were they capable of performing at high sodium-coolant outlet temperatures, both contemplated in the design of future fast reactors. Therefore, development of metal fuels was discontinued in the late 1960s in favor of ceramic fuels.

However, EBR-II continued to operate with metal fuel as its main or driver fuel, and this reactor was the test bed for all other fast reactor fuels and materials until FFTF became operational. As a consequence, a continual development of metal fuel occurred at Argonne National Laboratory. Over a number of years, design changes were developed that increased the maximum burnup of metal fuel. And during the same period, reactor coolant outlet temperatures were generally lowered. As a result metal fuel became a viable alternative to ceramic fuel.

The concept of an Integral Fast Reactor (IFR) using metal fuel, the pyroprocess, and a co-located fuel cycle facility was developed at ANL in the early 1980s. General Electric developed a similar concept using a co-located fuel cycle facility, the PRISM reactor system design. PRISM used metal fuel as the reference fuel design with mixed oxide fuel as a backup. A key aspect of these concepts was remote fabrication and electrochemical reprocessing (i.e., the pyroprocess), the goal of which was a simplified, inexpensive process and improved proliferation resistance.

Knowledge of metal fuel is sufficiently mature that a basis for design and licensing can be advanced for the alloy that was developed in the IFR program, namely, the U-Pu-Zr alloy. Because most of the metal fuel testing was performed with shorter fuel pins and binary fuel, it will be necessary to verify codes for longer fuel pins and ternary fuel. This would be part of the activity to gain an understanding of the extrapolation from EBR-II.

Where actinide transmutation is a design objective, the fabrication performance of the fuel with high minor actinide content, and with americium in particular, should be demonstrated with further testing.[16]

Nitride

The state of development of nitride fuel is modest when compared to either the mixed oxide or the metal alloy. Nitride fuel is attractive for two reasons. It exhibits many of the same desirable characteristics of metal fuel, e.g., high heavy metal density and good thermal conductivity. Further, it has excellent compatibility with sodium (and lead). But the amount of testing to date is very small.[16]

Purpose

Some of the research issues for the LMR fuels analysis are highlighted below.

- **Thermal Conductivity**. The metal fuel proposed for the ALMR was originally estimated to have a thermal conductivity and a correspondingly low peak and average fuel centerline temperature. However, data coming out of the IFR program at ANL indicated that the thermal conductivity might be lower than expected because of isotope migration (i.e., by Zr and Pu), porosity development, and fission-product accumulation. A database for the prototypical fuel is needed to determine the thermal conductivity in the equilibrium cycle (i.e., with reprocessed material) as well as with fresh fuel. The data are needed to accurately predict peak fuel temperatures for all the DBAs.
- **Isotope Migration and Liquidus / Solidus Temperatures**. A database is needed to determine the effects of isotope migration and to qualify the liquidus and solidus temperatures of the metal fuel when isotopic migration is considered in both fresh and recycled fuel.
- **Eutectic Penetrations**. Eutectic penetrations of the clad occur when the iron in the clad interacts with isotopes in the fuel to cause a low-temperature melt to form at the clad/fuel interface which dissolves the clad away. Because of the internal pressure from fission-product gases, the clad will then burst open. This is one area where ANL has developed a substantial database, but not for prototypical fuel. The database must be extended to incorporate fuel with high burnup because lanthanide attack could form a eutectic at a low temperature. The database is needed to qualify fuel pin failures during off nominal conditions.
- **Fuel Performance**. Demonstration of recycled metal fuel alloys up to the 150,000-MWd/t burnup level is needed. The effects of lanthanides and actinides on fuel performance should be investigated.
- **Fuel Swelling**. A database is needed to determine the effects of temperature, neutron flux, Pu enrichment, actinide and lanthanide concentrations, and burnup on irradiation swelling of fuel. The data are needed over the full range of normal and off-normal conditions in an LMR.
- **In-Pin Molten Behavior**. If a section of the fuel melts, data must be developed to determine the behavior of the melt within the pin and what mechanism would cause fuel clad failure. It would be of interest to know whether the melt moves toward the clad to cause a eutectic penetration failure or stays within a small zone within the fuel.
- **Fuel/Clad Mechanical Interaction**. The mechanical pressure the prototypical fuel exerts on the prototypical clad once it swells out to the clad must be considered and a database developed. The bounding material between the fuel and clad is sodium.

Minor actinide-bearing fuels require further property assessment work for both MOX and metal fuels (Ref. 18).

- **Fuel/Clad Chemical Interaction**. Once the fuel swells out and contacts the clad, the chemical constituents can pass between the fuel and the clad and form an alloy that melts at low temperatures. This is a limiting condition for the metallic fuel. The iron and uranium form a eutectic at a temperature of about 704 degrees C (1299 degrees F) and will fail the clad if the process continues. Also, the fuel might be using recycled fuel material from previous cycles, so the effects of the actinides and lanthanides must also be taken into consideration when developing the database.
- **Clad Eutectic Penetration Rates**. All constituents that have a potential to form a low-temperature alloy when the clad and fuel are in physical contact must be determined. The most limiting cases must be used for the design of the clad and penetration rates so that clad wastage can be determined in each of the DBEs. This will clarify the fuel failures expected for each category of events.
- **Fuel Axial Conduction Model**. One of the major features of metallic fuel is its expected high thermal conductivity. Most thermal hydraulic codes (including SSC) were developed for low-thermal-conductivity oxide fuel. Hence, neglecting axial heat conduction was not a concern. However, for metallic fuel with high thermal conductivity, an axial conduction model would be important because it would reduce peak local temperatures and might even change the outcome of several severe-accident events. In other words, the difference could be between the code predicting melting and the code predicting no melting. A model for axial conduction in the fuel in a thermal hydraulics code could significantly reduce the peak clad and fuel temperatures and also give more accurate predictions of these temperatures.
- **Fuel Length Effects**. A database is needed to determine the effects of pin length on axial fuel swelling. Most of the data has come from EBR-II, which has a flat power profile and pin lengths of 33 cm, while the ALMR, for example, would be 134 cm long and have a more pronounced cosine axial power shape.

  Basic property needs include data on fuel performance for SFR fuels that contain minor actinides. The impacts of minor actinides on thermophysical properties must be assessed. The systems based on MOX fuel are primarily under development in Japan, and their preferred recycling option is an advanced aqueous process. Metal-fueled reactor systems under development in the United States use a pyroprocessing recycling process as the preferred fuel cycle option.[5]
- **Performance Data for Recycled Fuel**. A significant technology gap for systems using recycled fuel is a need for performance data and transient safety testing of fuel that has been recycled using prototypic processes.
- **Performance of Remotely Fabricated Fuel**. For either fuel option, the fuel will contain a relatively small fraction of minor actinides and, with the low-decontamination fuel cycle processes contemplated, also a small amount of fission products. The presence of the minor actinides and fission products dictates that fuel fabrication be performed remotely. This creates the need to verify that this remotely fabricated fuel will perform adequately in the reactor (Ref. 18).

Objectives and Associated Activities

It is necessary to establish that fuel-element failure propagation will not occur by chemical mechanisms. For oxide fuels, it is known that interaction between the fuel and coolant will cause swelling of the fuel and a potential failure propagation mechanism. More accurate prediction of cladding failure is needed to advance the fuel design. For metal fuels, the fuel/coolant chemistry is generally well known and is benign. Addition of minor actinides to the fuel mixture will introduce an uncertainty. More data on high-burnup fuel/clad chemical interaction of recycled fuel is needed. This is a fuel development issue.[3]

Related NRC research

These need to be reviewed.

Related domestic and international cooperation

These need to be reviewed.

Related NRC and international research activities

These need to be reviewed.

Application of Research Results

To be reviewed.

3. <u>Materials Analysis</u>

Background

A key research area important to safety is the behavior of sodium coolant and the materials performing the structural, barrier, and retention functions under normal and off-normal conditions expected in LMRs.

Purpose

Some of the issues for LMR materials analysis are highlighted below.

<u>Sodium</u>. Sodium increases the reliability and operating life of components, partly because it does not corrode common structural materials, such as stainless steel. The experience in decommissioning EBR-II showed that materials and components in the core operated in liquid sodium without significant damage or corrosion. When components were removed from the sodium pool after 30 years they were found to be just as shiny as the day they went in. The original marks that welders and other craftsmen had made 30 years earlier when they created the component could be seen (Ref. 2).

Other sodium properties also enhance reactor safety and reliability. For example, sodium is chemically compatible with the metal fuel. This makes small failures in the cladding, the stainless-steel tubes that encase the fuel, far less likely to grow. In addition, sodium tends to bind chemically with several important radioactive fission products, which reduces radioactive releases if fuel fails (Ref. 2).

Of course, because of its high melting point, the use of sodium as a coolant imposes a requirement to have trace heaters around components and piping to preheat the system before sodium charging and to keep sodium in liquid state under all reactor conditions, including maintenance (Ref. 20). Other challenges include sodium's high density and its interactions with air and water.

The system has to be designed to be leak-tight with provision of inert cover gas over free sodium surfaces in components in order to avoid any ingress of air and to accommodate sodium volume changes with temperature. The volumetric expansion of sodium (about 2.7 percent) on melting requires an expansion tank be located at the highest point to allow free expansion of the metal to take place without any incident. This property of sodium is also very conducive to the establishment of natural circulation of coolant by modest temperature differences during decay heat removal.[20]

Liquid sodium reacts readily with air and oxidation reaction can occur in a runaway manner leading to sodium fire. The ignition temperature for sodium in air is 200 degrees C and as low as 120 degrees C in a stirred liquid pool. Hence, the piping and components are to be equipped with leak-detection devices to detect any leakage early in order to limit the effects of the fire.

Furthermore, provisions should be made to collect the leaking sodium and to avoid any reaction of sodium with structural concrete. Sodium reacts readily with water or steam to form sodium hydroxide and hydrogen. This reaction is highly exothermic; hence, these reactions have major implications in the design, material selection, and protection system for sodium-heated steam generators.[20]

High-Temperature Materials. A paper on NRC and Advisory Committee on Reactor Safeguards (ACRS) Technical Issues Relating to the Clinch River Breeder Reactor (CRBR) presented at the June 2006 ANS meeting's technical session on "High Temperature Design, Methodology and Regulatory Issues" focused on technical issues and safety concerns with high-temperature metals for fast LMRs (Ref. 21). Those issues include:

i) Recognizing that creep strains concentrate in grain boundaries, how do limits on strains in the equivalent homogeneous material prevent excessive grain boundary strains and grain boundary cracking?

ii) Because material properties at elevated temperatures are measured in uniaxial test specimens, how do we account for the lower ductility and creep rupture strength under biaxial conditions?

iii) How can we reliably predict and prevent long-term creep cracking behavior?

iv) Because base metal, weld material, and the heat-affected zone (HAZ) of weldments have different creep properties, how are we accounting for the resulting strain concentration effects at the weldments?

v) How are we accounting for long-term environmental and irradiation effects?

vi) Because Linear Elastic Fracture Mechanics (LEFM) and Elastic Plastic Fracture Mechanics (EPFM) are not applicable in the creep regime, how is very-high-temperature crack growth being analyzed?

vii) How is the aging effect on material being taken into account from a safety point of view?

viii) Have lessons learned from vessel failures at elevated temperatures in the commercial and industrial world been considered in the design criteria?

ix) Is inspection technology available for measuring creep swelling, creep rupture damage, and creep cracking?

x) Are flow tolerance technologies available for very-high-temperature safety-related reactor components or do they need to be developed?

xi) Have the effects of material imperfections been considered in the safety analyses?

xii) Safety which depends entirely on "black box" finite-element cyclic creep analyses is not sufficiently reliable for licensing purposes. An independent simplified method of verifying the cyclic creep response is needed to provide the necessary assurance of reliability.

Other Material Issues. Materials issues include (1) fuel-cladding constituent interdiffusion behavior for minor actinide (MA) bearing fuels, (2) development of high-strength steels for use in structures and piping to improve safety and economics, and (3) improved materials for recycling systems (Ref. 7).

Structural Materials. Chrome ferritic steels, instead of austenitic steels, are viewed as promising structural materials for future plant components, because of their superior strength and thermal properties at elevated temperatures, including high thermal conductivity and low thermal expansion coefficient. With these materials, more compact structural designs are foreseen. On the other hand, some drawbacks have to be overcome with these materials. They include degradation of ductility and toughness during high-temperature service. Weldability is also a concern. An elevated-temperature material-strength database should be established for design-by-analysis purposes.[3]

SUPPORTING AREAS

4. Regulatory Framework

Background

Knowledge Management. The U.S. Nuclear Regulatory Commission (NRC) has recognized the importance of Knowledge Management (KM) as a discipline and as a tool for capturing and transferring knowledge as part of its human capital management process. KM programs and activities will support agency objectives to maintain core competencies and meet the future needs of program and regional offices. As NRC adds new staff in anticipation of increased workloads resulting from announcements by licensees to submit some 11 combined operating license applications for as many as 17 new commercial nuclear power plants, KM programs can support the transfer of knowledge from staff who have many years of licensing and regulatory experience to new staff to not only assist in the licensing of new plants but also in the continued oversight of the safe operation of existing plants.

KM can be succinctly defined as to include both the active creation, transfer, application and reuse of (tacit) individual knowledge and codified (explicit) collective knowledge, supported by new approaches, relationships and technologies, to increase the speed of innovation, decisionmaking and responsiveness to organizational objectives and priorities.

NRC's Office of Nuclear Regulatory Research (RES) has a pilot KM project underway focusing on high-temperature gas-cooled reactors (HTGRs). There is the need to initiate a comparable effort for liquid-metal-cooled reactors (LMRs) in view of the Department of Energy's recently announced Global Nuclear Energy Partnership (GNEP) initiative that includes the development of sodium-cooled fast reactors.

GNEP represents an overall strategy to expand the use of nuclear power, develop and deploy new technologies for recycling nuclear fuel, minimize waste, and develop enhanced nuclear safeguard approaches for proliferation-resistant fuel cycle technologies. Actinide burning (i.e., management of wastes) is an example of a technology breakout for LMRs that goes beyond their original design basis.

A key element of the GNEP initiative is the development and demonstration of an Advanced Burner Reactor (ABR). The ABR is a fast-spectrum reactor designed to "consume" or transmute the transuranic elements (plutonium and other long-lived radioactive material) in spent nuclear fuel from existing light-water reactor (LWR) fuel into shorter-lived isotopes. The approach calls for the "sequential development of two reactors: (1) an Advanced Burner Test Reactor (ABTR)—a relatively small-sized test reactor—and (2) a prototype commercial-scale

ABR with an integrated fuel cycle plant. Given the previous work done on fast reactors using sodium as the coolant in the United States (EBR-II, FFTF, and LMFBR/CRBR) and internationally (Russia's BN-600 and BOR-60, France's Phénix, and Japan's Jōyō), sodium was selected as the coolant for the ABTR. Note that sodium-cooled LMR technology development programs have recently been started in both Korea and China, with the China Experimental Fast Reactor (CEFR) achieving first criticality in 2010. International cooperation is being touted as a key in the development of the ABTR and ABR (Refs. 22 and 23).

One of the key elements of GNEP includes the development of small-scale reactors with exceptional safety, reliability, safeguards, and proliferation resistance for deployment in developing countries that have rapidly growing energy demand but limited grid capacity and nuclear support infrastructure. The consensus of both U.S. and international studies is that small and medium-sized reactors meet these user needs and proliferation concerns better that the current generation of commercial nuclear power plants. The power range for an SMR as identified by the International Atomic Energy Agency (IAEA) is up to 300 MWe for small and up to 700 MWe for medium-sized reactors. The stated range of interest for GNEP applications has been initially targeted at the 50 to 350 MW(e) size (Ref. 22).

Purpose

Need for KM for LMRs. The NRC has over 40 years of experience with licensing and regulating commercial nuclear power plants. However, most of this experience has been focused on LWRs and will likely have limited applicability for LMRs. The safety and operational issues of LMRs will also be considerably different. Given the large-scale planning underway for fiscal year (FY) 2006 and aggressive startup for FY 2007 (funding level projected at $250 million overall) of the GNEP program, NRC should give consideration to initiating a KM project for sodium-cooled LMRs in FY 2006, comparable to the HTGR KM activity. The LMR KM activity can evolve as DOE finalizes plans for the ABTR/ABR and SMR development in FY 2006 so that tasks can be initiated to identify, evaluate, and categorize appropriate information to support future licensing activities. The ABTR/ABR project will be moving quickly to identify not only key technical documents, test results, and data from work conducted over 30 to 40 years ago in the United States on LMRs, but also key individuals who possess firsthand knowledge in the design and operation of these reactor systems. This human technology base is disappearing because of the attrition of these experts. It will also be important to understand and capture relevant information in terms of documents, results, and experts from the international community for those countries who have currently operating LMR power and/or test reactors or have operated LMRs in the past.

Not only is the information of value technically, but if DOE uses information from international sources in some way to support the licensing basis for the design certification for the commercial ABR, the NRC will need to understand the content and context of such information. Furthermore, should the next-generation SMR happen to be an LMR, such as Toshiba's 4S design, the collection, evaluation, and transfer of the LMR knowledge will be important as well for any potential licensing activities for the 4S.

Thus, early initiation of an LMR-based KM activity would position NRC to start preparation for the eventual preapplication and full design certification phases for ABRs and likely leverage DOE activities as far as identification of important documents and experts, both domestic and international.[22]

Major Regulatory Issues. In the early 1990s the NRC conducted preapplication reviews of several advanced reactors. Of particular interest are the preapplication reviews of the PRISM and SAFR liquid-metal reactors (LMRs) reported in NUREG-1368 and NUREG-1365, respectively. Following the completion of these reviews, a staff policy-issues paper (SECY-93-092) to the Commission identified ten generic advanced reactor issues, eight of which apply to LMRs and depart from the current regulatory requirements. These eight policy issues pertain to LMR designs of power range between 350 and 465 MWe.[24]

A number of possible challenges would be involved in licensing an LMR in the United States. Some of the regulatory issues would be: residual heat removal, accident evaluation, seismic isolation, fuel performance, new materials, in-service inspection if the design involves a long (e.g., 30-year) operation, emergency planning, and quality assurance. Some of these issues would be expected to involve Commission policy considerations.[8, 24]

SUMMARY OBJECTIVES AND FUTURE WORK

The primary R&D objectives of the liquid-metal-cooled reactor (LMR) program are to (a) develop independent accident and transient analysis tools for safety and licensing reviews, (b) in the near-term, continue to implement a knowledge-management program for LMRs, given the dearth of NRC staff with extensive experience in this technology, and (c) continue ongoing interaction with the DOE on GNEP activities related to liquid metal-cooled advanced burner reactors, as well as interaction with Toshiba and their partners in their presentations to the NRC related to preapplication review of the Toshiba 4S reactor design.

If LMR applications appear likely,

1. An in-depth assessment of infrastructure, in concert with a Phenomena Identification and Ranking Table (PIRT), should first be conducted to more fully identify potential materials issues.
2. A more detailed research plan should then be developed.
3. Rigorous prioritization and identification of necessary confirmatory research should then be pursued. For example, the staff might need to identify those issues that will require independent research to evaluate the technical basis that might be developed by prospective applicants.

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11. ABSTRACT (200 words or less)

The Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research (RES) has been actively engaged in an effort to develop and compile information on liquid-metal-cooled reactors (LMRs), particularly sodium-cooled fast reactors (SFRs), as part of a concerted knowledge management (KM) program for LMRs. The objective of this program is to apply KM principles to capture and retain technical knowledge related to LMRs that NRC staff might need to support evaluations as part of licensing activities as related to future LMR applications for design certification (DC) and combined operating licenses (COL). In support of this objective, efforts have focused on documentation of previous NRC licensing activities associated with Fermi-1, Power Reactor Inherently Safe Module (PRISM), and the Clinch River Breeder Reactor (CRBR), as well as international research and development, safety analyses, and licensing activities associated with LMRs. This includes information and documentation on LMR severe accidents, operational issues, and analysis tools and codes that would be relevant for licensing purposes. In addition to capturing historical information and discussing recent and/or current activities, a second objective was to develop informational tools to facilitate the compilation and access to this information such as an LMR Desk Reference and an SFR Technology Course that are described in this report. Much of the information compiled and collected for LMRs has been added to NRC's Knowledge Center, which is one of NRC's key information technology applications for capturing and sharing knowledge.

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