SANDIA REPORT

SAND2012-4260
Unlimited Release
Printed May 2012

Sodium Fast Reactor Safety and Licensing Research Plan – Volume I

M. Denman, J. LaChance, T. Sofu, G. Flanagan, R. Wigeland, and R. Bari

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

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Sandia National Laboratories

Issued by Sandia National Laboratories, operated for the United States Department of Energy by Sandia Corporation.

Printed in the United States of America. This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from
U.S. Department of Energy
Office of Scientific and Technical Information
P.O. Box 62
Oak Ridge, TN 37831

Telephone:	(865) 576-8401
Facsimile:	(865) 576-5728
E-Mail:	reports@adonis.osti.gov
Online ordering:	http://www.osti.gov/bridge

Available to the public from
U.S. Department of Commerce
National Technical Information Service
5285 Port Royal Rd.
Springfield, VA 22161

Telephone:	(800) 553-6847
Facsimile:	(703) 605-6900
E-Mail:	orders@ntis.fedworld.gov
Online order:	http://www.ntis.gov/help/ordermethods.asp?loc=7-4-0#online

SAND2012-4260
Unlimited Release
Printed May 2012

Sodium Fast Reactor Safety and Licensing Research Plan – Volume I

Matthew Denman
Sandia National Laboratories
Albuquerque, NM 87185-0748

Tanju Sofu
Argonne National Laboratory
Argonne, IL 60439

Roald Wigeland
Idaho National Laboratory
Idaho Falls, ID 83415

Robert Bari
Brookhaven National Laboratory
Upton, NY 11973

George Flanagan
Oak Ridge National Laboratory
Oak Ridge, TN 37831

Jeffrey LaChance
Sandia National Laboratories
Albuquerque, NM 87185

ABSTRACT

This report proposes potential research priorities for the Department of Energy (DOE) with the intent of improving the licensability of the Sodium Fast Reactor (SFR). In support of this project, five panels were tasked with identifying potential safety-related gaps in available information, data, and models needed to support the licensing of a SFR. The areas examined were sodium technology, accident sequences and initiators, source term characterization, codes and methods, and fuels and materials.

It is the intent of this report to utilize a structured and transparent process that incorporates feedback from all interested stakeholders to suggest future funding priorities for the SFR research and development. While numerous gaps were identified, two cross-cutting gaps related to knowledge preservation were agreed upon by all panels and should be addressed in the near future. The first gap is a need to re-evaluate the current procedures for removing the Applied Technology designation from old documents. The second cross-cutting gap is the need for a robust Knowledge Management and Preservation system in all SFR research areas. Closure of these and the other identified gaps will require both a reprioritization of funding within DOE as well as a re-evaluation of existing bureaucratic procedures within the DOE associated with Applied Technology and Knowledge Management.

ACKNOWLEDGMENTS

The authors would like to express their appreciation to the Advanced Reactor Concepts (ARC) Program within DOE-NE for funding this multi-year effort. This report was finished under the DOE-ARC work packages # AR-12SN040205, AR-12IN040205, AR-12OR040205, and AR-12BN040205.

The authors would also like to thank the chairs and members of the supporting gap analysis reports for their continued assistance and recommendations for resolving the gaps their panels identified. During this report, Thomas Wei (Argonne National Laboratory [ANL]), Richard Vilim (ANL), Richard Wood (Oak Ridge National Laboratory [ORNL]), and David Holcomb (ORNL) were tasked with identifying gaps in instrumentation and control for sodium reactor technology. Their assistance in this regard was invaluable. The expertise of Chris Grandy (ANL), Ken Natesan (ANL), and Abdellatif Yacout (ANL), was vital in helping compile the ranking tables in this report. Gratitude is also extended to Jesse Philips (SNL), John Sackett (INL), and Art Wright (ANL), for their assistance in editing the report.

The authors would like to thank TerraPower, Westinghouse, and AREVA, for their feedback regarding the results of the five gaps analysis reports. Their industrial insight was greatly appreciated.

CONTENTS

FIGURE

TABLES

ACRONYMS

4S	Super-Safe, Small, Simple
ACRR	Annular Core Research Reactor
AEC	Atomic Energy Commission
AFC	Advanced Fuel Campaign
ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARC	Advanced Reactor Concepts
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
AT	Applied Technology
ATR	Advanced Test Reactor
BDBE	Beyond Design Basis Event
BOP	Balance of Plant
CDA	Core Disruptive Accident
CEA	Commissariat à L'energie Atomique
CEFR	Chinese Experimental Fast Reactor
CO_2	carbon dioxide
CRBR	Clinch River Breeder Reactor
DBE	Design Basis Event
DBT	Ductile to Brittle Transition
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor-II
ECI	Export Controlled Information
EOL	End of Life
FBTA	Fuel Behavior Testing Apparatus
FCCI	Fuel Cladding Chemical Interactions
FCCT	Fuel Cladding Transient Tester
FCMI	Fuel Cladding Mechanical Interactions
FCT	Fuel Cycle Technology
FFTF	Fast Flux Test Facility
FM	Ferritic/Martensitic
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GIF	Generation-IV International Forum
HFIR	High Flux Isotope Reactor
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency

IEEE	Institute of Electrical and Electronics Engineers
IFR	Integral Fast Reactor
IGR	Impulse Graphite Reactor
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
JAEA	Japan Atomic Energy Agency
LDRD	Laboratory Directed Research and Development
LMFBR	Liquid-Metal Fast-Breeder Reactor
LWR	Light Water Reactor
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NEET	Nuclear Energy Enabling Technologies
NQA	Nuclear Quality Assurance
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
OSTI	Office of Scientific and Technical Information
PFBR	Prototype Fast Breeder Reactor
PICT	Plant Inherent Control Tests
PIE	Post-Irradiation Experiments
PNNL	Pacific Northwest National Laboratory
PRA	Probabilistic Risk Assessment
PRISM	Power Reactor Innovative Small Modular
PSER	Preapplication Safety Evaluation Report
PSID	Preliminary Safety Information Document
R&D	Research and Development
RDT	Reactor Development and Technology
ROSATOM	Rosatom Russian Nuclear Energy State Corporation
SA	Severe Accident
S-CO_2	Supercritical-CO_2
SDO	Standard Development Organization
SFR	Sodium Fast Reactor
SHRT	Shutdown Heat Removal Tests
SMR	Small Modular Reactor
SNL	Sandia National Laboratories
SFR	Sodium Fast Reactor
SSC	System, Structures, and Components
SQE	Software Quality Engineering
TREAT	Transient Reactor Test facility
U.S. NRC	U.S. Nuclear Regulatory Commission
WPF	Whole-Pin Furnace

EXECUTIVE SUMMARY

While the Department of Energy Nuclear Energy (DOE-NE) has historically sought to work cooperatively with nuclear power vendors to license a Sodium Fast Reactor (SFR) domestically, recent support for these efforts has been intermittent. As a result, the ability for the DOE-NE to successfully support the safety-related aspects of a SFR license application has become uncertain. To better understand DOE-NE's current capabilities and provide insight into potential future research programs, a series of five safety-related gap analysis panels were formed in the following areas: Accident Sequences and Initiators, Sodium Technology, Fuels and Materials, Source Term Characterization, and Codes and Methods. These panels were comprised of representatives from across the DOE lab complex, academia, industry and international bodies and identified key gaps relating to existing experimental databases and capabilities, computational abilities, human capital, and the knowledge-base. The resulting five gap analysis reports are compiled in Volume II of this report.

Volume I of this report is focused on consolidating the safety-related gaps identified by the five expert panels and evaluating these gaps to inform future decision-makers. Historical operating experience, licensing efforts, and proposed industry consensus standards were examined for insights concerning the relative importance of identified gaps to a safety case. Existing domestic and international facilities were cataloged to identify which safety gaps could be closed without new investment, for large, capital-intensive facilities. Experts were consulted to estimate costs associated with addressing identified gaps to help inform future decisions made with limited resources. Finally, existing research programs which were already addressing safety-related research needs were highlighted.

Once the safety-related gaps were evaluated, they were divided into six prioritization categories including: gaps which are estimated to cost less than a million dollars, time sensitive gaps which should be addressed in the near term, gaps with the potential for international cooperation, gaps related by precursors (other gaps or capabilities), gaps significant to normal operation of a SFR, and gaps which would be addressed with a fully funded SFR program. These categories were then examined for similarities under the assumption that resolving gaps which appeared under numerous prioritizations would be beneficial to the SFR community. The following general areas emerged as high level research needs which the DOE-NE should address in the foreseeable future: knowledge preservation and management efforts, source term and sodium fire modeling capability, modernization of codes which support licensing, and improved abilities to model accident phenomena in a post-Fukushima world.

A sufficiently funded and coordinated knowledge preservation and management emerged as the most pressing need within the SFR community. Documentation for some codes significant to licensing have been lost (e.g., NUBOW), and other codes are not currently maintained (e.g., LIFE-Metal). Many documents and test data have been saved recently using funds from the Advanced Reactor Concepts program, but much of this information still needs to be sorted, interpreted, and transferred into retrievable storage. While patchwork efforts have ensured that information has not been lost, these efforts should be coordinated to make certain that the DOE-NE knows what it has preserved and what still needs to be determined. A key component to knowledge preservation and management is to ensure that the codes which are needed to support a safety case have the appropriate level of stewardship and user-base within the DOE labs and academia. While the user-base for even the highest profile SFR safety-related codes (e.g., SAS4a) needs improvement, the fuel performance code LIFE-Metal has only a few, nearly emeritus, stewards. Should they retire before the code has been transferred to the next generation of users, the DOE's capability to support and defend a fast reactor fuels qualification case may be lost. Finally, the current process for handling Applied Technology (AT) documentation was determined to occasionally be counterproductive to knowledge preservation efforts. DOE-NE should consider streamlining the AT review process, by allowing qualified researchers speedier access to AT documents, by allowing referencing of AT documents, and timely removal of the AT designation when no longer needed while still retaining the preservation of the technology for possible international exchange.

Due to the SFR's reliance on inherent and passive safety, little effort has been made to maintain domestic capabilities to characterize source terms and sodium fires. After Fukushima, these capabilities will likely receive greater attention in the SFR's safety case. The NRC currently uses MELCOR to examine severe accidents of Light Water Reactors. MELCOR has many similarities to the internationally supported SFR containment performance code Contain-LMR, making MELCOR an ideal code to absorb Contain-LMR's capabilities to become the domestically supported source term and sodium phenomenology code. Additionally, some U.S. facilities, such as Sandia National Labs' (SNL) sodium fire testing vessel Surtsey, are currently under-utilized and are capable of addressing many high-priority sodium phenomenology gaps. These facilities, if properly utilized, can help close the remaining sodium-related safety gaps in a sodium version of MELCOR.

In addition to the need for expanded user-base for SFR safety codes, the codes themselves need to be updated to take advantage of modern computing practices and infrastructure, and to improve their capabilities. SAS4A needs enhancements to improve modeling accuracy, functionality, and usability as well as to take advantage of the multi-processor capabilities which are the current trend in modern computing. Models within SAS4A also need to be improved, especially concerning transitions from full power to natural circulation if passive safety is to be an integral part of the SFR's safety case. In regards to LIFE-Metal, data from the new fuel tests should be incorporated into the code's empirical correlations, and the user manual needs to be updated to incorporate the new changes.

Another implication of Fukushima will likely be increased importance of seismic isolation and modeling. New experimental data concerning the response of SFR core materials and structures, systems, and components to earthquakes and other external events is needed to improve the current predictive modeling capability. Additionally, no validation experiments yet exist to gauge the model's predictive capability.

Finally, external stakeholders were given the opportunity to comment on this report's recommendations before final publication. While various stakeholders obviously would like DOE research to more closely align with their specific design needs (e.g., very high burnup fuel or a stronger emphasis on severe accident research), commonalities in their comments were also identified. A general agreement was reached concerning the importance of seismic-related modeling and validation efforts, as well as the need to preserve licensing codes.

1 INTRODUCTION

1.1 Motivation

The U.S. Department of Energy (DOE) is currently funding various research initiatives to support future fast reactor deployment. In addition to accomplishing new technical achievements, the ability to support a U.S. industrial partner's licensing effort may potentially influence future DOE funded research areas. Sodium fast reactors (SFRs) have long been both studied and operated by the DOE and its predecessor, the Atomic Energy Commission (AEC). By the early 1990s, the DOE complex operated a wide range of fast reactor related experimental facilities. The DOE national laboratories supported fast reactor research with both experimental facilities and predictive model development across all relevant areas pertaining to SFR deployment until the program was suddenly closed in 1994.

In 1994, policy changes in the U.S. Government abruptly canceled nearly all of the fast reactor research in the United States. The abrupt end to fast reactor research resulted in many programs ending prematurely, with no funding available to properly preserve intermediate or final project results. Currently raw data tapes and paper records are stored in uncontrolled environments where they are not protected from deterioration or unintentional destruction. Even if the data tapes are recovered, experts from the early 1990s would need to be consulted in order to decipher them. These experts are quickly retiring from the DOE, ensuring that the likelihood of significant additional capability loss increases dramatically with every passing year.

1.2 Determination of the State of Knowledge of SFR safety

Between 2009 and 2011, the Advanced Reactor Concepts (ARC) program within DOE has funded a series of five gap analysis panels tasked with identifying safety-related gaps that still remain in the knowledge needed for making the safety case for licensing a SFR. These five gap analysis reports have been combined in Volume II of this report. Both burner and breeder reactors were considered in this expert elicitation, although their respective needs were not delineated in the report's recommendations. Filling these gaps would be essential in order to license a future SFR. It was expected that, because of the relatively mature state of SFR technology, many of the identified gaps would be related to design options that were near maturity at the end of the Integral Fast Reactor (IFR) program, i.e., metallic fuel qualification, evolution of the licensing structure (e.g., increased use of Probabilistic Risk Assessment [PRA]), or because of loss of institutional knowledge (e.g., abandoned computer codes).

Expert panel elicitation was used in order to identify the regulatory significant gaps in each of the five topical areas. Experts were asked to rank each research area on the technical adequacy of existing knowledge in an area of interest and the importance of the research area to the licensing process.

The five topic areas examined for safety-related gaps included:

- **Accident Sequences and Initiators –** How well known are the accidents and associated phenomena that are important in establishing the safety case for licensing a SFR? (Sackett et al., 2010)
- **Sodium Technology –** How well can a designer accommodate and model potential energetics associated with sodium fires? (Corradini et al., 2010)

- **Fuels and Materials –** How well does the existing experimental database allow for fuel qualification and use of advanced structural materials, and what is the status of fuel performance computer codes? (Walters et al., 2011)
- **Source Term Characterization –** How well can we model the source term for a SFR to support emergency planning and other regulatory issues? (Powers et al., 2010)
- **Codes and Methods –** What are the status and capabilities of existing computer codes and models for SFR accident analysis? (Schmidt et al., 2011)

1.3 Objective of this Report

This report is intended to both make research recommendations based on five previously conducted safety-related gap analysis reports and determine cross-cutting needs that exist throughout the SFR-related DOE complex. While the eventual funding decisions will be privy to the changing needs and budget priorities of external decision-makers, the identification of cross-cutting gaps, i.e., a coordinated Knowledge Management and Preservation effort, may be the most important information highlighted by this report.

In general, this report assumes that a U.S. SFR design will use metallic fuel, either binary (U-10%Zr) or ternary (U-x%Pu-10%Zr). While some of the gap reports considered other fuel types, such as oxide fuel, for simplicity these gaps are not included in this cumulative report due to the existing U.S. research direction for SFRs. Additionally, it is assumed that gaps relating to design alternatives, e.g., qualifying extremely high-burnup fuel, deploying a loop or pool design, or developing a Supercritical-CO_2 (S-CO_2) power conversion cycle, will have a lower priority associated with resolving outstanding licensing and safety issues than gaps that cross-cut almost any SFR design. Unlike gaps relating to oxide fuel, optional gaps are not discarded from this report's analysis.

In an attempt to ensure that the research plan identified in this report addresses the needs of all potential U.S. SFR designers, a wide range of stakeholders will be consulted to provide input on the recommendations of this report. A series of case studies were used to assist the decision maker in consideration of a range of varying funding priorities, i.e., Low Cost or Time Sensitive Gaps.

A draft of this report was released to DOE in February of 2012 (without external input). The final version of this document was released in June of 2012. This final version will include all external feedback acquired between the release of the draft and final document.

1.4 Structure of the Report

This report is intended to highlight current safety-related research needs for the SFR program. In order to provide insights from the identified safety-related gaps, a SFR decision maker needs to understand the history of SFR research program, ongoing SFR related research activities, and the most likely future licensing pathways facing the SFR. Thus, the report is structured as follows:

1. **Introduction** – Provides an overview of the motivations and objectives of this project.
2. **History and Future of SFR Development** – Discusses historical SFR research and facilities, open items from licensing efforts between the 1970s and 1990s, the role of beyond design basis accidents, and updates to the SFR licensing standard ANS/ANSI 54.1.

3. **Methodology –** Discusses the initial selection of the five topical reports, the general selection criteria of the expert panels, the procedure for identifying and ranking safety-related gaps used in the five gap analysis reports, the methodology for evaluating the safety-related gaps, and a discussion of safety-related topical areas which were not selected for a formal gap analysis report.
4. **Review of the Gap Analysis Reports –** Reviews the recommendations from each gap report, summarizes the gaps into generic gap categories for easier prioritization and evaluates the gap categories using expert opinion elicitation.
5. **Potential Resolution of Gaps –** Overviews both national and international experimental facilities, national programs which are already funding the resolution of safety-related gaps, and documents the need for a systematic knowledge management and preservation program.
6. **Potential Roadmaps –** Groups safety-related gap categories into six potential funding prioritization groups and highlights commonalities from these prioritizations.
7. **Observations and Recommendations –** Reviews the insights gained from the study and summarizes external feedback received before publication of the report.
8. **References –** Lists the references used to support this report.
9. **Bibliography –** Lists the references used to support the five gap analysis reports.

2 HISTORY AND FUTURE OF SFR DEVELOPMENT

2.1 History of SFR Deployment

Next-generation nuclear energy systems currently under consideration aim for significant advances over existing and evolutionary light water reactors (LWRs) in the areas of sustainability, economics, safety, reliability, and nonproliferation. Development of these systems is an international effort, involving collaborations under the framework of the Organization for Economic Cooperation and Development's (OECD's) Generation-IV International Forum (GIF) and the International Atomic Energy Agency's (IAEA's) International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

Recent studies under these programs highlight the importance of closed fuel cycle systems using fast-neutron reactors to meet the sustainability goals through efficient resource utilization. In comparison to LWRs, fast reactors can extract about two orders of magnitude more energy from the same amount of fuel. Moreover, nearly all long-lived heavy elements (transuranic waste that remains radioactive for a long time) can be consumed in a fast reactor with a closed fuel cycle, greatly reducing the amount of repository space needed for waste isolation.

Although reconsidered as part of the next generation of nuclear reactors, the fast-spectrum systems, particularly the liquid sodium cooled reactors (SFRs), are not new concepts. Since the 1950s, SFR technologies have been pursued and demonstrated worldwide, leading to the construction and operation of several experimental and prototype fast reactors in the United States (Experimental Breeder Reactor [EBR]-I and -II, FERMI , Southwest Experimental Fast Oxide Reactor (SEFOR), and the Fast Flux Testing Facility [FFTF]), Soviet Union (BR-10, BOR-60, BN-350 and -600), United Kingdom (DFR and PFR), France (RAPSODIE, Phénix, and Superphénix), Germany (KNK and SNR-300), and Japan (JOYO and Monju). These fast reactors have achieved well over 300 reactor-years of accumulated operation experience.

Although interest in SFRs has declined during the past several decades, the restart of Monju in Japan, completion of the Chinese Experimental Fast Reactor (CEFR), and ongoing construction of the Prototype Fast Breeder Reactor (PFBR) in India and BN-800 in Russia have demonstrated a renewed interest in SFR development. Recent commercial interests in building SFRs within the United States have been shown by GE (Power Reactor Innovative Small Modular [PRISM]), Toshiba (Super-Safe, Small, Simple [4S]), and TerraPower (TP-1).

2.2 Open Items From the Previous SFR Licensing Efforts

The current U.S. SFR licensing experience has come about from the Clinch River Breeder Reactor (CRBR) and the Advanced Liquid Metal Reactor (ALMR) program interactions with the U.S. Nuclear Regulatory Commission (U.S. NRC). In the 1970s and early 1980s, the DOE attempted to license CRBR, but Congress cut funding before the project was complete. While core disruptive accidents (CDAs) were not considered as part of the design basis for CRBR, accidents that could lead to CDAs, including unprotected accidents and large break loss of coolant accidents, received a large amount of regulatory attention, which prolonged the licensing process. The U.S. NRC Atomic Safety and Licensing Board (ASLB) eventually excluded CDAs from the licensing basis, with the U.S. NRC staff stating:

It is our current position that the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum.

The CRBR licensing process resulted in a U.S. NRC Safety Evaluation Report in 1983, NUREG-0968.

After CRBR was canceled in 1983, the DOE embarked on the Advanced Liquid Metal Reactor (ALMR) program. This program emphasized a pool-type reactor concept and metal fuel to avoid severe accident related regulatory issues that impeded CRBR's licensing. After an initial design competition between the PRISM (GE) and Sodium Advanced Fast Reactor (SAFR) (Rockwell/Westinghouse) reactor concepts, with both designs submitting a Preliminary Safety Information Document (PSID) to the U.S. NRC in 1986, the GE-led PRISM reactor became the focus of the ALMR program in 1988. Coupling the ALMR fuel cycle with GE's PRISM reactor concept later led to establishment of the Integral Fast Reactor (IFR) program.

The resulting Preapplication Safety Evaluation Report (PSER) highlighted key regulatory issues for PRISM (U.S. NRC, 1991, 1994). The major non-design-specific items highlighted by the U.S. NRC Staff in the PRISM PSER include:

- limited performance and reliability data for passive safety feature,
- unverified analytical tools used to predict plant response,
- limited supporting technology and research,
- limited construction and operating experience, and
- incomplete information on the proposed metallic fuel.

While IFR program continued to address these identified issues, some of them were never fully resolved due to the abrupt closure of the IFR program in 1994. While the PSID was analyzed by a relatively small group within the U.S. NRC and thus cannot be taken as encompassing all potential regulatory issues, the PSER provides the best indication of potential regulatory concerns for the sodium reactor. As will be seen in Section 4, many of these issues were also captured by the five supporting gap analyses. Only the limited construction and operating experience was neglected from our findings because this gap can only be addressed through the act of building more reactors, not sponsoring more research.

2.3 Consideration of Beyond Design Basis Accidents

The relative importance of safety-related gaps for the sodium reactor depends upon the safety approach taken by both the design and the potential regulatory. These approaches have evolved over time, shifting the relative importance of research areas along with the evolving philosophies.

The safety of reactors is traditionally based on deterministic approaches. Deterministic approaches consider a set of challenges to safety and determine how those challenges should be mitigated. A probabilistic approach to safety enhances and extends this traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, by providing a logical means for prioritizing these challenges based on risk significance, and by allowing consideration of a broader set of resources to defend against these challenges. For both the traditional deterministic approach and a deterministic approach that is enhanced by PRA, the resulting safety framework for a reactor design is essential to the licensing application that is presented to the regulator.

The licensing basis for the reactor is constructed around the definition of events to be considered within the safety framework of the reactor. Those that are beyond the safety framework are subject to much less rigorous evaluation. A PRA, because it considers both likelihoods and consequences of events, can very readily be used to identify and study those events that would fall within a licensing basis. Events that have large consequences, beyond the capability of the reactor plant to safely mitigate, are reduced in risk significance by requiring that their likelihoods are acceptably small such that they do not cause the risk profile of the plant to be beyond an accepted norm. For advanced plants, such as the SFRs, even higher aspirations for safety are expected because the advanced plants incorporate improved design features and may benefit from operational experience with the current fleet of LWRs (U.S. NRC., 2008)

Fukushima has refocused regulatory attention on severe accidents as well as the low probability range of PRA results. In post-Fukushima work, the regulatory importance of containment response codes, e.g., Contain-LMR or MELCOR, will be increased even with the enhanced passive safety of current SFR designs. Seismic risks will take a more prominent role in advanced reactor risk assessment, possibly even driving containment design or the adoption of three-dimensional seismic isolation. The increased priorities associated with addressing gaps related to containment response and seismic related modeling has been incorporated to the following analysis.

Furthermore, determining how passive safety relates to the required redundancy and diversity of defense-in-depth will be extremely important. Passive systems do not fit in the binary event or fault trees within a PRA. This challenge may cause the potential for a designer to overly rely on passive systems to ensure safety.

2.4 Proposed Updates to Fast Reactor Industry Consensus Standard ANSI/ANS 54.1

If a reactor designer approached the U.S. NRC to license a SFR, the designer would have to demonstrate that they meet all appropriate codes and standards to ensure the safety of the public. Unfortunately, while some standards were published (see Section 3.5.3), most previous attempts to develop fast reactor codes and standards, including a licensing standard, were abandoned when the research programs were terminated. These standards need to be resurrected or completed before it can be determined how much supporting research is required for licensing.

Currently, the ANS and ANSI are working on an update of the 1989 SFR industry consensus standard ANSI/ANS 54.1 - 1989, *General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant* (ANSI/ANS 54.1, 1989). The 1989 standard, which was eventually withdrawn in 1999 because it was not being maintained by the ANS, was intended to provide the SFR equivalent of the LWR-specific general design criteria (GDC) that are promulgated in the U.S. NRC regulation 10 CFR 50 Appendix A. This current updated draft standard recognizes changes in SFR technology, safety, and the use of PRA in achieving a risk-informed and performance-based approach to the safe design of a reactor.

The purpose of this draft standard is to define criteria to be satisfied for providing assurance that SFRs are designed so that they can be operated with acceptable risk to public health and safety and to the environment. This purpose is achieved through the identification of applicable safety requirements from the U.S. NRC, industrial codes and standards, and other published guidance and professional engineering practices. It is also the purpose of the draft standard to define requirements for the acceptable use of probabilistic risk information in support of design decisions (i.e., risk-informed design criteria).

The update to ANSI/ANS 54.1 builds on the body of knowledge on sodium-cooled reactor technology and information that has been developed over the past half-century. Previous standards, guidelines, safety reviews, design, operating, and research experience have been recognized in the development of this draft standard. It is intended to be implemented on both a design- and site-specific basis and to be compatible with regulatory dose limits related to the protection of public health and safety and the environment.

While this report summarizes the current safety-related gaps associated with SFRs, the degree of required research in various areas cannot be determined until standards, such as ANSI/ANS 54.1, are complete.

3 METHODOLOGY

3.1 Selection of the Five Topical Areas

Since SFR design studies are at an early stage and currently include many system design and fuel options, the safety-related gap evaluation must address the range of relevant options needed to fully assess the phenomena that must be considered in the safety evaluation. In defining the major safety-related gap topic areas, it is recognized that the topic areas and phenomena are driven by the potential accident sequences (the first topic area listed below). The accident sequences are in turn significantly affected by the SFR design. However, some aspects of SFR safety, such as sodium phenomena, are less design-specific. A preliminary list of gap topic areas was identified based on general knowledge of SFR technology and requirements for evaluating the SFR safety case (Pickard et al., 2008). The topic areas (corresponding to panels) for the SFR safety-related gap evaluation project are:

- **Accident Initiators/Sequences –** The accident scenario panel addressed the broadest scope of safety-relevant phenomena. This panel utilized available information and expert opinion to define classes of events with safety significance and the systems or subsystems that are affected; identified the phenomena that are active in those events; assessed the importance of these phenomena against safety criteria, and assessed the state of knowledge for analyzing the safety significance of these phenomena. The broad spectrum of phenomena addressed in this panel resulted in overlap with the other phenomenological panels. The scope of this panel extended to secondary systems and balance of plant interactions with accident events where appropriate.

- **Sodium Phenomena –**Sodium coolants add an additional dimension of safety relevant phenomena that must be considered in the overall evaluation of SFR safety. The probability and location of a sodium fire is design-dependent, but the phenomena associated with the range of sodium fires that could result from a leak in the primary or secondary system can be assessed at an early stage. There has been considerable research on sodium fires and sodium phenomena, such as sodium concrete interactions, in the United States and other countries as part of previous fast reactor development programs. Pool fire phenomena that were considered by this panel include radiation heat fluxes between the pool surface and environment, aerosol generation, convection at the surface, development of the oxide crust, and sodium flow (spreading) issue. Spray fire phenomena include modeling of the plume and spray dynamics (thermal-fluid dynamics), spray characteristics, including droplet size and velocity distributions, chemical combustion kinetics, and agglomeration phenomena. Sodium interactions with concrete may also result in hydrogen production and aerosol generation.

- **Fuels and Materials –** Understanding the characteristics of fuels and key materials under accident conditions is essential to reactor safety analysis. Advanced fuel characterization under both normal and accident conditions is needed to assess fission product release/retention, fuel-coolant interactions, fuel-clad interactions, fuel swelling, and fuel motion mechanisms under accident conditions. Both metal and oxide fuel types are being considered for SFRs and the implications of both types must be evaluated. The anticipated high burnups and long service lifetimes also pose issues for key non-fuel materials, particularly cladding materials. The very high burnup and the resulting high fast neutron fluences require that clad properties (embrittlement, swelling, etc.) be understood and included in accident analysis. The fuels and materials panel assessed fuel and material phenomenology important to safety and identified the information needed to support the overall safety and licensing approach. Fuel types and materials of construction, and the associated conditions of safety-relevant service, would be design- and scenario-dependent, and therefore this panel was convened after the accident initiator panel.

- **Source Term –** The source term of primary interest is the release of radionuclides to the site and beyond the site boundary. To assess the defense-in-depth of a plant design, the regulatory process also will define a source released to the containment and evaluate the leakage of radionuclides to the environment. The concentrations of radionuclides suspended in the containment atmosphere as functions of time are of crucial importance. With the exception of noble gas releases and some small fraction of the radioactive iodine release, the radionuclides are suspended in the reactor containment as aerosol particles. Aerosol sources to the containment arise directly from fuel in the case of fuel handling accidents. Otherwise, important aerosol releases to the containment come from the sodium coolant. The least intense type of radionuclide release from sodium comes from quiescent sodium pools retained in the reactor coolant system. Contamination of the sodium coolant with radionuclides is most important when fuel rods have ruptured and the fuel is exposed to the coolant.
- **Computer Codes and Models –** Based on the range of scenarios and phenomenology identified from the SFR safety evaluation performed in the other safety-related gap topics, the codes and models panel addressed the analytical capabilities and data required to adequately assess the safety implications of SFR scenarios and phenomena. The scope of this panel included the assessment of thermal-hydraulics (TH), heat transfer, and structural and neutronics modeling capabilities, as well as the evaluation of the validation basis. Of particular interest is the evaluation of accident analysis tools that are generally unique to nuclear reactor safety. In addition, this panel addresses the potential for modern advanced modeling and simulation techniques to improve nuclear safety analysis approaches using higher-fidelity, integrated multi-process tools. This activity was closely coupled to the Accident Initiators/Sequence panel scope.

3.2 Selection of Expert Panels

While each of the five gap analyses used slightly different criteria to select their expert panels, the following three guidelines were generally followed:

- The panel should be chaired by an authority in the topical area of interest,
- The panel should include at least one expert in every topical area analyzed, and
- The panel should be representative of the DOE complex and, if possible, the international community.

The panel size varied for each topic ranging from 5 to 12 panelists.

3.3 Gap Identification and Ranking Process

The individual panel evaluations identified safety relevant features and components that are involved in the range of accident sequences relevant to that panel, and then assessed the phenomena active in those scenarios. The panels assessed the importance of those phenomena to the safety case for a SFR and the knowledge level currently available to address these issues for licensing. Gaps or areas of inadequate understanding were identified to define safety-related R&D needs. Figure 1 shows a high-level description of how the gap analyses were conducted.

The importance of the issues identified by each panel was ranked qualitatively by the panel members as either: High (H), Medium (M), or Low (L) importance. The general descriptions of these importance ranking levels are:

- High (H) – phenomena is of first order (fundamental) importance based on evaluation criteria.
- Medium (M) – phenomena is of secondary (contributing) importance based on evaluation criteria.
- Low (L) – phenomena not important for the scenario and evaluation criteria being considered.

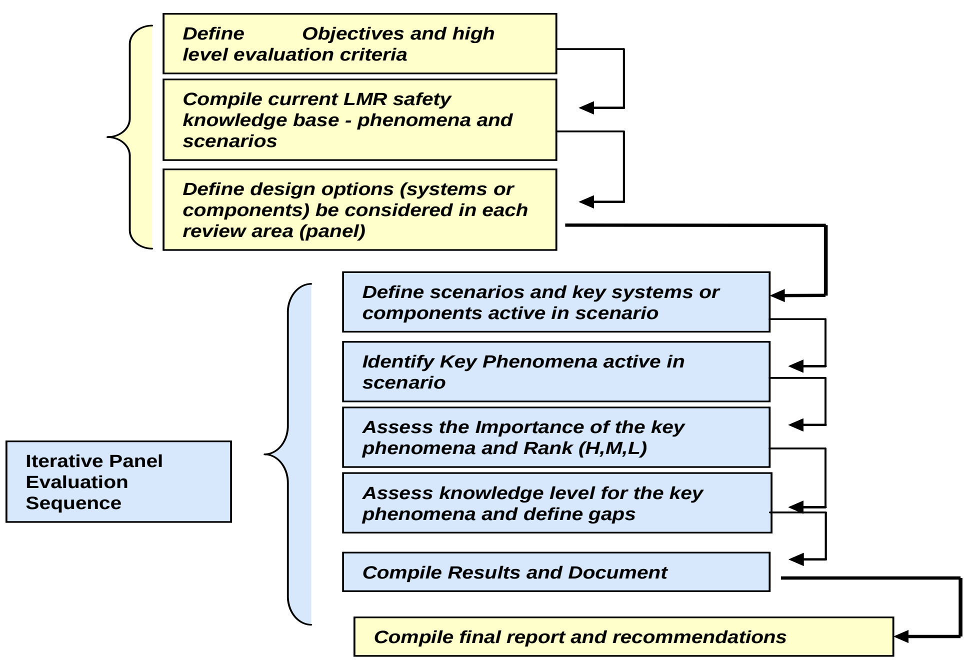

Figure 1. Sequence of gap analysis activities and panel process.

Evaluating the state of knowledge of a phenomenon generally involves the assessment of both the modeling capabilities and the database to validate the model(s). The panels discussed each phenomenon extensively during the evaluation, with the general criteria for ranking the state of knowledge defined as:

High (H)

- A physics-based or correlation-based model is available that adequately represents the phenomenon over the parameter space of interest.
- A database exists adequate to validate relevant models or to make an assessment.

Medium (M)

- A candidate model or correlation is available that addresses most of the phenomenon over a considerable portion of the parameter space.
- Data are available but are not necessarily complete or of high fidelity, allowing only moderately reliable assessments.

Low (L)

- No model exists, or model applicability is uncertain or speculative.
- No database exists; assessments cannot be made reliably.

The gap analysis knowledge results are also provided in the summary table, which includes comments for each ranking. In that same section of the report, we provide details of the rationale or justification for the panel knowledge ranking in our discussion.

3.4 Gap Closure Evaluation

Once all of the gaps have been identified, descriptive ranking criteria are needed to evaluate and prioritize the closure of these gaps. These criteria were first evaluated by chairs of the identifying panel and then updated by other subject matter experts. The ranking criteria which were used are described below. It should be noted that the expected responses were left broad to acknowledge the inherent uncertainty of predicting many of these variables. The evaluation criteria are explained in Table 1.

Table 1. Gap Evaluation Criteria.

Criteria	Expected Response
What is estimated cost range?	How much will it cost to fill this gap if the requisite facilities were available? $100K–1M, $1M–10M, $10M–100M, $100M+.
Are there related DOE funding programs other than ARC?	List other U.S. programs which may be interested in helping to finance this gap.
Are there related international funding programs?	List other international programs that may be interested in financing this gap.
Is there a time sensitivity associated with resolving this gap?	If the gap is not filled by the following time window, significant capabilities would be lost. < 5 years, < 10 years, < 15 years, 15+ years.
How much time would be required to fill the gap?	How long will it take to fill the gap? < 5 years, < 10 years, < 15 years, 15+ years
Are there precursors?	List activities that much first be completed before the gap can be filled.

Table 1. Gap Evaluation Criteria (cont.).

Criteria	Expected Response
US Facilities	List any U.S. facilities that could be utilized to fill this gap.
International Facilities	List any existing international facilities that could be utilized to fill this gap.
Importance	Two values are listed: • Was the gap rated high, medium, or low regulatory significance? • Was the current state of knowledge associated with the gap rated low, medium, or high? It should be noted that this evaluation is made assuming that the capability related to the gap would be included in a licensing effort. It is left to the decision maker to interpret the relative priority of design optional gaps.
Event Category	Does this gap fill an Anticipated Operational Occurrence (AOO), Design Basis Event (DBE), Beyond DBE (BDBE) or Severe Accident (SA) need?
Optional Design Feature	Is this gap associated with a fundamental design feature of a sodium reactor or with a beneficial, yet optional, design feature?

3.5 Gaps in the Reports

While an effort was made to form expert panels focused on the major areas affecting reactor safety, some subject areas fell outside the primary scope of this project. A cursory review of the safety-related subject area was conducted where the authors were able to identify potentially neglected subject areas. These potential gaps are addressed in Sections 3.5.1 through 3.5.3. The authors acknowledge that some subject areas may have been omitted.

3.5.1 Instrumentation and Control

Instrumentation and Control (I&C) is important to reactor safety and is addressed in Chapter 7 of a reactor's Final Safety Analysis Report (FSAR). Although no official gap panel was assembled to identify potential safety-related gaps for the SFR, I&C experts from Argonne National Laboratory (ANL) and Oak Ridge National Laboratory (ORNL) were consulted to conduct a cursory review of safety-related gaps in this area. These experts identified safety-related gaps concerning key sensors that proved unreliable in EBR-II and FFTF, the need for a well-funded Knowledge Preservation and Management program, the development of a SFR Surveillance Diagnostics and Prognostics system, and numerous technological gaps that, if narrowed, could improve the overall economics of a SFR.

The top two high-priority gaps that were identified were related to the reliability of high-temperature sensors in a sodium environment and an improved Knowledge Preservation and Management program. The need for improved sensors was demonstrated by the performance of the under-sodium sensors in EBR-II. Safety-related flow and pressure sensors slowly failed over the life of EBR-II, with barely enough remaining operational when EBR-II was shut down. Ensuring that an adequate number of safety-related sensors are functional over the plant's lifetime can be addressed through a combination of improved design (the core should be designed to allow for replacement of these sensors) and improved reliability both at high temperatures and in a sodium environment. The Nuclear Energy Enabling Technologies (NEET) program is currently developing high-temperature sensors for reactor concepts, but there exists a need for the program to focus on sodium environment effects on these new sensors.

The other high-priority gap that was identified was related to the need for a Knowledge Preservation and Management program. While a Knowledge Preservation and Management program is currently in place for fast reactor I&C, this effort is not comprehensive enough to prevent a loss of existing capability. This gap cross-cuts almost every area of SFR technology.

An additional high-priority safety-related gap was the need for Surveillance Diagnostics system to confirm safety-related passive feedbacks. This system couples online sensor measurements with computer models and uncertainty propagation to continuously verify that the passive feedbacks, which are relied upon to avoid core damage during unprotected accidents, behave as expected. Because these feedbacks can change over a 60-year reactor life, an online verification system will be extremely beneficial to ensure the regulator of continued reactor safety. This underlying capability is currently being developed in the Small Modular Reactor and Light Water Reactor Sustainably programs, but a SFR focus will be needed to appropriately account for fast reactor specific phenomena, such as core expansion.

The final gaps identified would also deal with the economic competitiveness of the SFR. Improvements in acoustic instruments to measure sodium flow in the primary pool, steam generator leak detection, under-sodium viewing, subassembly blockage detection and mitigation (primarily a safety issue), and development of under-sodium maintenance robotics will all improve the economics of the SFR as well as safety, but will not jeopardize the licensability of the SFR if they are not addressed quickly.

3.5.2 In-service Inspection and Under-Sodium Viewing

Development of in-service inspection capabilities to facilitate periodic examination and repair of the plant components is primarily a design challenge. The design considerations include improvements of the primary and secondary systems in order to reduce the number of structures and components to be surveyed, to locate sensitive zones in accessible areas from either the inside or outside, and to reduce the need for welding. The efforts include adaptation of the design for in-service inspections and repair requirements, development of ultrasonic transducers for under-sodium viewing, non-destructive examination techniques, repair processes, and associated robotic equipment.

In addition to improving the plant’s economics and providing an investment protection, in-service inspection and repair is also fundamentally related to reactor safety (for detection of failures) and physical protection (assuring the security of the nuclear material and equipment during the reactor’s life span). Therefore, establishment and validation of the techniques to improve inspection, maintenance, availability, and decommissioning have broader licensing implications.

One of the major difficulties for SFR inspection and repair is the high-temperature sodium coolant, which is opaque and reactive with air and water. Some of the reactor components can be inspected and repaired by removing them from the reactor, but majority of other parts of the reactor block need specific means for in situ inspection. Since the reactor inspections and maintenance operations for these parts have to be performed under sodium, instead of optical devices, the ultrasonic devices that are tolerant of high temperatures are needed.

In-service inspection for SFR is usually aimed at assuring the main safety functions such as reactivity monitoring and control, decay heat removal, and containment of hazardous products. It is typically implemented in various stages of plant design and operation by:

1. Reducing inspection and repair needs at design stage by design optimizations,
2. Continuous monitoring of the structures and components during the reactor operation against reactivity anomalies, unexpected temperature spikes, leaks, mechanical deformation, vibrations, and anomalous acoustic signals,
3. Performing required periodic tests and examinations to confirm the operational status observed during continuous monitoring, and looking for undetected damages like cracks and corrosion,
4. Answering to any abnormal situation by allowing intervention (including core unloading, draining, and examination), and
5. Repairing and replacing the components that are found to be defective.

Since a regulatory oversight would be required for any of these in-service inspections and especially the repair stages described above, consideration of the gaps will also be important. However, although these objectives require technological readiness and progress particularly in the area of under-sodium viewing, non-destructive examination, and robotics for basic repairs, no major specific safety-related gaps have been perceived at this time.

3.5.3 Codes and Standards

In designing any nuclear reactor or fuel-cycle facility, designers have traditionally relied on a large number of codes and standards, many of which are consensus documents arrived at using the consensus process of the American National Standards Institute (ANSI) and other standards development organizations. These codes and standards have been developed and adopted by governmental agencies, either regulatory or developmental agencies like the U.S. NRC or DOE.

For U.S. sodium fast reactors, there already exists a huge body of such standards, which fall into two broad categories:

1. The AEC and its successor, the DOE, developed and adopted many "Reactor Development and Technology (RDT) Standards," starting in the 1960s and continuing into the early 1980s; and
2. Several standards development organizations (including the American Nuclear Society [ANS], American Society of Mechanical Engineers [ASME], Institute of Electrical and Electronics Engineers [IEEE], and a few others) developed a large number of consensus standards. Taken all together, these codes and standards number in the few hundreds, albeit the most important of these number in the several dozens.

Unfortunately, the major standards development activities for sodium fast reactor technology mostly came to a halt in the 1980s. Since that time, these standards have largely been dormant. To revive them and make them applicable to today's needs, it is necessary to review them individually to determine which are still applicable, which could be easily updated, and which are so outdated that it would be necessary to rewrite the standards.

Revising a consensus code or standard is a time consuming effort – it takes two to three years minimum, and sometimes longer, depending on the issue. A five year period is not atypical if a new technical approach is being introduced and consensus about it must be reached within the very broad technical community. Yet some of the standards development activity is now, or will soon be, on the "critical path," in the sense that it will impede the specific technical work that facility designers will need to do. Both the design work and the U.S. NRC regulatory review process are at risk. The standards development work is intellectually challenging and also organizationally challenging in that this work requires bringing industry, DOE, the DOE labs, U.S. NRC, and others together into a small set of new ANS, ASME, and IEEE standards committees.

The standards revision efforts are also not necessarily expensive – each new standards committee needs only $50K to $100K of outside funding to do its work. The bulk of the labor is almost always donated by the participating experts' institutions. However in the case of SFRs where most of the expertise resides at national laboratories additional DOE funding may be needed to support these subject matter experts time spent on standard development. But funding is needed to pay travel for those few (retirees, consultants) who cannot get an employer to pay travel, and also for staff time at the standard development organizations (SDOs).

The initial work must identify the codes and standards required to design, construct, and license the SFR, after which a dialogue must be established with the pertinent SDOs to begin a timely process for developing or modifying the necessary codes and standards.

Specifically, a survey is needed of codes and standards required for design, construction, and licensing of SFRs. This survey would identify existing codes and standards that are applicable, in whole or in part, to SFRs, and identify additional codes and standards that are required for the SFR. The status and viability of existing codes and standards must be assessed considering important factors including the spectrum of possible SFR designs and materials. Based on this evaluation, the codes that are on the critical path for the design and licensing of a SFR would be identified. Then proposals would need to be developed and presented to the appropriate SDOs.

4 REVIEW OF THE GAP ANALYSIS REPORTS

Five gap analysis reports were written by expert panels to highlight the open safety hurdles still facing the sodium reactor. This section reviews the conclusions from each report, summarizes the highlighted gaps, and provides additional expert opinion concerning the level of effort needed to close each gap.

4.1 Accident Initiator/Sequences Gap Analysis

The Accident Initiator/Sequence gap analysis examined the regulatory significance and state of knowledge for various phenomena that can affect the SFR's response to anticipated operational occurrences, design basis accidents, and beyond design basis accidents.

4.1.1 Summary of Findings

The following general conclusions were reached by the expert panel:

- **There are no major technology gaps that would prevent the design and the development of a licensing case for a sodium-cooled fast reactor as long as one stays with known technology.** Additionally, there were no identified differences in knowledge between oxide and metallic fuel, or between pool and loop designs. New transient testing may be needed to verify margin to failure for planned reactor fuels and to complete severe accident testing to support safety analyses and U.S. NRC licensing discussions.
- **There are technology knowledge gaps for fuel with minor actinide content significantly higher than known fuels**, likely requiring a fuel qualification program sufficient to understand the extent of the differences. Depending on the outcome of the comparison, new transient testing may be required to quantify margin to failure and identify post-failure phenomena.
- **Passive, self-protecting features in the plant design can be an effective and important part of the safety case, potentially reducing the importance of phenomena that historically have had higher uncertainties**. Verification of predicted reactor system response to upsets as part of plant qualification testing is recommended to reduce uncertainties in expected reactor response arising from modeling uncertainties. Continued development of analysis tools is recommended to improve simulation capability and reduce prediction uncertainties.
- **Availability and accessibility of known technology is required to avoid repeating past R&D**. A comprehensive knowledge management effort is recommended to achieve this. Although the team did not identify any significant knowledge gaps, the data supporting the modeling may not have been collected and reviewed/evaluated with the rigor needed to support licensing of an SFR. There is a real possibility that this firsthand knowledge of the data and its interpretation may be lost if a knowledge management program is not implemented. This would include collecting and cataloging information that exists in log and data books, especially from facilities that are destined for deactivation and decommissioning.
- **A plan needs to be developed to address the lack of experiments and tools qualified for use in a licensing environment, either by qualifying the existing experimental**

data and analysis tools, and/or by performing new experiments and developing new analysis tools.

There are important “stretch technologies” that have been identified and could be studied or developed to determine if they offer opportunities to improve the economics, safety, and security of a SFR. Although not needed to proceed with an advanced SFR, as such they may be considered as “gaps” for further advances in development of these specific technologies. These are:

- Advanced simulation of coupled neutronic/fluid flow dynamics.
- Supercritical CO_2 power conversion.
- High minor-actinide content fuel.

While no significant gaps were identified by the report, the rigor of the licensing justification for backup decay heat removal and core restraint systems were all areas of concern.

While emergency decay heat removal systems were subjected to testing before they could be employed in EBR-II and FFTF, it is unknown if the test documentation still exists and if these tests would meet current Nuclear Quality Assurance (NQA) standards. Additional questions about decay heat removal include:

- Were tests conducted under all worst conditions?
- Are there international data available?
- Could some of the system evaluation be conducted by codes (best estimate plus uncertainty)?

An additional area of concern when attempting to license a reactor with SAS4A/SASSYS-1 is that this code system was developed as a research tool, not a licensing code. Thus, SAS4A/SASSYS-1 may not meet NQA standards required by the U.S. NRC. Additionally, much of the supporting information for the code is Applied Technology (AT) and thus cannot be handled by the U.S. NRC. A program would have to be initiated to remove unnecessary AT designations from important documentation to create a licensing code from SAS4A or SASSYS.

The final area of concern resides in the area of core restraint modeling. The current code, NUBOW, couples stress and creep calculations with a point kinetics model of the reactor. Point kinetics may not be sufficient to model this behavior; thus more research may be needed in this field. Additionally, the state of the supporting experimental database for NUBOW is currently unknown. This gap is also associated with sixth gap category identified in the Fuels and Materials gap report.

4.1.2 Identified Gaps

The accident sequence and initiator gap report identified 10 gaps of varying degrees of importance. To facilitate the prioritization process with gaps from the other four reports, these gaps were consolidated into the following seven topical areas listed in Table 2. A detailed explanation of each gap was not provided in the Accident Initiator/Sequence gap report.

Table 2. List of Research Gaps Associated with High Level Gap Topical Areas for Accident Initiator/Sequences.

Gap	Experimental Database	Ability to Model
Steady State Intact Fuel and Fuel Changes (AIS01)		
End-of-Life (EOL) fuel composition.	**M**	**M**
EOL prediction of reactivity feedback.	**M**	**M**
Transition to Natural Convective Cooling (AIS02)		
Sodium stratification.	**H**	**M**
Thermal Response of Structures (AIS03)		
Thermal striping.	**M**	**H**
Decay Heat Rejection (AIS04)		
Radiation heat transfer from vessels.	**M**	**H**
Power Conversion Cycle (AIS05)		
CO_2-sodium chemical interaction.	**L**	**L**
CO_2 release and impact.	**L**	**L**
Fuel Transient Behavior (AIS06)		
Length effects on fuel performance during transient for metallic fuel.	**L**	**H**
High-minor-actinide content fuel performance.	**L**	**L**
Severe Core Damage (AIS07)		
Fuel motion, dispersal, and morphology for metallic fuel.	**M**	**M**

Table 3 is a consolidated list of the highest-priority gaps in the area of Accident Sequences and Initiators. While detailed gaps were identified in the Accident Sequences and Initiators report that can guide the details of further research in this area, Table 3 is intended to provide an overview of the types of programs that need to be funded to close the remaining gaps. The colors in the table are provided to highlight differences in the classification of gaps.

Importance to safety H/M/L designations reflects the expert panel's belief of the relative importance of each gap in defending a SFR safety case. The gap report assigns values for importance to safety to each of the gaps listed in Table 2. The importance to safety within category column in Table 3 is an aggregated representation of the rankings associated with the individual gaps. The overall lowest knowledge state is the lowest value assigned to either ability to model gap or experimental database. Because not all gap reports listed both modeling and experimental states of knowledge, the overall lowest knowledge state is needed for comparison across gap reports.

Table 3. Accident Sequence and Initiator Gap Topical Areas.

Gap ID	Name of Gap Topical Areas	Importance to Safety Within Category	Ability to Model Gap	Experimental Database	Overall Lowest Knowledge State
AIS01	Steady State Intact Fuel and Fuel Changes	H	M	M	M
AIS02	Transition to Natural Convective Cooling, Sodium Stratification	H	M	H	M
AIS03	Thermal Response of Structures, Thermal Striping	H	H	M	M
AIS04	Decay Heat Rejection, Radiation Heat Transfer from Vessels	H	H	M	M
AIS05	Power Conversion Cycle, S-CO_2 Accident Analysis	H	L	L	L
AIS06	Fuel Transient Behavior	M	H	L	L
AIS07	Severe Core Damage, Metal Fuel Motion, Dispersal and Morphology	H	M	M	M

4.1.3 Gap Closure Criteria

Table 4 summarizes expert opinions on topics that may be used to help prioritize funding of the gaps identified in Table 3. The colors in the table are provided to highlight differences in the range of the evaluation parameters.

Table 4. Accident Sequence Gap Closure Estimates.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
AIS01	10M-100M*	AFC	JAEA / CEA / ROSATOM	10 – 15*	5 – 10	Fuel Testing Facility	ATR and/or HFIR	JOYO	DBA, BDBA	H/M	No
AIS02	1M-10M*	-	Most GIF Members	10 – 15*	< 5	Sodium Component Test Facility	-	PLANDTL, CCTL	DBA, BDBA	H/M	No
AIS03	1M-10M*	-	-	10 – 15*	< 5	Sodium Component Test Facility	MELT Facility once Built	PLANDTL, CCTL	DBA, BDBA	M/M	No
AIS04	1M-10M	-	JAEA / CEA / ROSATOM	10 – 15*	< 5	-	NSTF	ATHENA	DBA, BDBA	H/L	Yes
AIS05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	10 – 15*	< 5	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed / Scaled up version of SNAKE	-	DBA, BDBA	M/M	Yes
AIS06	10M-100M*	AFC	-	10 – 15	< 5	Irradiated Fuel and (TREAT or ACRR)	TREAT or ACRR	-	DBA, BDBA	M/L	No
AIS07	10M-100M*	AFC	-	10 – 15	< 5	Irradiated Fuel and either (TREAT or ACRR)	TREAT or ACRR, Possibly CAMEL	MELT-II	SA	H/L	No
AIS08	1M-10M	-	JAEA	10 – 15	5 – 10	-	-	-	DBA BDBA	H/L	Yes

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

4.2 Sodium Technology

The sodium technology gap analysis focused on the current ability to model sodium fires, gas production, and sodium interactions with concrete and drip liners.

4.2.1 Summary of Findings

The objective of this Sodium Technology Gap Analysis was to:

- Identify safety-relevant phenomena in the area of sodium technology,
- Establish criteria and evaluate importance of the phenomena to safety,
- Assess the status of knowledge pertaining to the phenomena, and
- Identify knowledge or capability gaps as well as suggest a path to bridge these gaps.

The panel evaluation involved:

- Defining the relevant accident scenarios and the safety-relevant features and components relevant to sodium technology phenomena,
- Identifying the key phenomena active in the scenarios, and
- Assessing the importance of those phenomena to the SFR safety case, assessing the knowledge level currently available to address these issues for establishing the safety case of a SFR.

The technology areas with inadequate understanding (i.e., gaps) are then identified, allowing one to define safety-related R&D needs.

Sodium coolants add the dimension of chemical compatibility and reactivity phenomena when a sodium leak occurs that must be considered in the evaluation of SFR reactor safety. This work focuses on the phenomena that would exist after a sodium leak occurs and does not focus on SFR inspection and leak detection technologies. The panel considered that sodium leaks and interactions can be classified into three general broad accident areas:

- Sodium leakage from the primary or intermediate cooling system at high-pressure in a compartment;
- Sodium leakage from the primary or intermediate cooling system at low-pressure into a compartment;
- Coolant leakage (water or supercritical carbon dioxide [CO_2]) into sodium within the power-cycle heat exchanger.

The distinction between high and low pressure was qualitative, based on the concept that leaks at higher pressures (~1 MPa) cause a dispersed sodium spray in a containment compartment, whereas leaks at lower pressures (0.1 MPa) could be characterized more as a jet of sodium.

Given these accident scenarios, the panel identified a group of seven general phenomena, which were then subdivided into specific phenomena for ranking of their importance and their knowledge base. Most of these phenomena are important in determining the containment response during sodium leakage events and severe core damage accidents.

- Sodium spray dynamics

- Sodium jet dynamics
- Sodium-fluid interactions
- Sodium-pool fire on an inert substrate
- Aerosol dynamics
- Sodium-cavity-liner interactions
- Sodium-concrete-melt interactions

The key evaluation criteria or figure of merit used for ranking these phenomena is radioactive material release to the public from fission products and other sources in the plant. This is the common criteria for all SFR gap analyses. Two refined evaluation criteria were identified by this expert panel:

- Radiological consequence criteria: dose at the site boundary, worker dose, radioactive inventory; and
- Functional criteria: potential impact of leak on system or component operability or functionality.

The panel identified the following sodium technology gaps in each of the seven phenomena areas:

Sodium Spray and Jet Dynamics: Given a sodium leak as a spray, a substantial sodium surface area is produced that is subject to evaporation and/or oxidation. The size of droplets that form is difficult to predict, particularly the range of droplet sizes or the full distribution of droplet sizes. Since this range of droplet sizes has a strong influence on the degree of evaporation/oxidation before impact on a surface, an experimental program to understand relevant droplet size distributions is recommended. A related gap that can be addressed, in concert with this phenomenon, is in the prediction of liquid breakup when very large droplets impact surfaces and splash. Associated aspects (oxidation, ignition, optical properties) can be investigated simultaneously.

Sodium-Fluid Interactions: CO_2 is being considered for the power conversion fluid in advanced supercritical cycles for future sodium fast reactors. The intermediate loop for the SFR uses non-radioactive sodium coolant as the heat transfer medium between the sodium-cooled reactor and the CO_2 power cycle. Thus, the intermediate heat exchanger is where sodium - CO_2 interactions may occur given a leak of the high-pressure gas into the low-pressure sodium flow channels. Available research and understanding of the fluid interaction between sodium-CO_2 is limited for operational as well as safety issues. While the importance to safety would be greater for designs which remove the intermediate loop, these designs were considered extremely long term options and were not considered in this report. Experiments and supporting analysis for sodium-CO_2 interactions is needed to determine their safety significance given such advanced power conversion systems.

Sodium Surface Pool-Fire on Inert Substrate: Substantial research has already been carried out to quantify the gross behavior of sodium pool fires at a variety of scales ranging up to fires involving cubic meters of sodium. This collection of information (i.e., test data and codes developed on the basis of that data) may be sufficient to support licensing activities for currently conceived fast reactor designs. To support development of advanced computational models that are increasingly being utilized to support design and licensing issues, additional data are needed, such as

- Radiation heat flux from a burning pool,
- Overall pool mass burning rate with oxide crust present,
- Oxide crust behavior, and
- Source term for sodium aerosols.

Sodium-Cavity Interactions

Sodium-Liner Interactions: Experiments focused upon steel liner corrosion with various ratios of sodium metal, oxide, hydroxide, and peroxide with steam present should be performed to provide data for model development and to understand the complex chemistry. In addition, failure of flawed liners can occur when sodium metal leaks behind the liner and reacts with underlying concrete. Large-scale experiments with sodium metal, sodium fire, and purposely-flawed liners with reactive aggregates need to be performed to evaluate the potential and to aid in model development of liner failure for this scenario.

Sodium-Concrete Interactions: Given liner failure, sodium concrete reactions have been observed both experimentally and operationally, and they can pose a serious threat to reactor operations and can even challenge containment integrity. A new series of experiments need to be conducted at large scale for both siliceous and carbonate concretes in order to better understand why experiments have not been reproducible and modeled appropriately. These experiments need to be conducted at large scale because vigorous reactions were not always observed at small scale. The experiments need to be conducted with and without sodium fire present and include aerosol production measurements.

Sodium-Concrete Interactions with Core Melt: Sodium concrete reactions with core melt are expected to enhance the rate of concrete ablation. Experiments should also be performed that will provide data for model development of fission product migration and partitioning between core melt, sodium metal, sodium concrete reaction products, and aerosols.

Aerosol Dynamics: The panel concluded that no major gaps in our knowledge on aerosol dynamics exist, although two areas were identified as uncertain. This may be especially true when considering the effect of sodium aerosols on a mechanistic radioactive source term released during a core damage accident. The uncertainty in the agglomeration process of sodium aerosols and other aerosols coming from fuel and cladding can result in uncertainties in mechanistic source terms. The degrees of importance of these uncertainties were better defined by the source term gap analysis team (see Section 4.2).

While historical codes exist that can model sodium fires, these codes need to be updated to incorporate the existing database from sodium fires test and maintained in a regulatory code such as MELCOR.

In 2005, funding was briefly provided to incorporate the existing state-of-the-art code which models sodium fires and calculates containment behavior, Contain-LMR, into MELCOR, but funding was cut before the project was complete. Contain-LMR originated as a Sandia National Laboratories (SNL) code that was given to Japan and then transferred back to SNL. It is unknown if the Japanese have improved Contain-LMR since the transfer back to SNL.

Additionally, a series of experiments were recommended to close outstanding gaps in the understanding of the details of large pool fire and sodium spray dynamics. The large pool fire gaps can be closed using existing facilities at SNL. ANL also has operational sodium fire facilities which may be utilized to close gaps associated with smaller scale testing. PNNL may have facilities that can be revived, but this has not been confirmed yet.

4.2.2 Identified Gaps

The Sodium Technology gap report identified 26 gaps of varying degrees of importance relative to establishing a safety case. To facilitate the prioritization process with gaps from the other four reports, these gaps were consolidated into the following six topical areas. The gap report also included brief comments outlining the details of the gap and how the gap can be filled. These comments are also included in Table 5.

Table 5. List of Research Gaps Associated with High Level Gap Topical Areas for Sodium Technology.

Gap	Experimental Database	Ability to Model	Details
		Sodium Spray Dynamics (ST01)	
Single drop particle average size	**M**	**M**	The Weber number gives average size, given the droplet velocity. Weber number correlations are well known for simple geometries (circular orifices) for both models and data, but correlations are not well known for irregular cracked geometries.
Single drop particle size distribution	**M**	**L**	Considerably more difficult to obtain size distribution data than average diameter data. Empirical correlation models exist for some cases. Only recent data are considered “good” data. Models will require data.
Pre-ignition phase dynamics.	**M**	**M**	While the Makino model is considered acceptable and the Morewitz et al. data are good (Makino, 2003; Morewitz et al., 1977), improvements would be desirable.
Basic evaporation and combustion.	**M**	**H**	There are good data and models for a single droplet but data for interacting sodium droplets (spray) is sparse.
Crust formation on droplets	**L**	**L**	Wick boiling is documented in the literature for sodium pools but data and models are lacking.
Source of Sodium Aerosol	**L**	**L**	There is no model known by panel and the data is sparse.
Model radiation transfer with/from aerosols	**L**	**L**	Missing absorption/emissivity data.
Inertial impact of molten sodium.	**L**	**M**	Data exist for water primarily but there is some metal spray data. Low-pressure spray droplets will have non-spherical shapes; there is a gap involving non-spherical droplets. The high-pressure spray model and data are adequate.
Burning of droplets on surface of a sodium pool	**L**	**L**	Models are not accurate in their present form and the data is almost non-existent.

Table 5. List of Research Gaps Associated with High Level Gap Topical Areas for Sodium Technology (cont.).

Gap	Experimental Database	Ability to Model	Details
Sodium-Fluid Interactions (ST02)			
High pressure fluid jet leak into sodium in heat exchanger	**L**	**L**	This general phenomenon is considered important but knowledge is good for sodium-water interactions and is lacking for sodium-CO_2 interactions.
Sodium Surface Pool Fire on an Inert Substrate (ST03)			
Radiation net heat flux	**L**	**H**	Models are good but parameters are poor with low accuracy (surface and aerosol optical properties; optical properties are linked with sprays).
Mass Burning Rate	**L**	**H**	When at high temperature burning, the models are good. For smoldering fires (burning through the crust) the models are poor. Most experiments were conducted using non-representative insulated surfaces.
Oxide crust behavior on pool substrate	**L**	**L**	Difficult to measure experimentally because of the low residence time of oxide to hydroxides.
Near-surface aerosol size/distribution	**L**	**L**	No good model is available.
Surface aerosol production	**L**	**L**	Interfacial effect at the crust is not well known.
Aerosol Dynamics (ST04)			
Sodium aerosol source term	**L**	**L**	The gap panel did not have the expertise to list specific research areas, but sodium aerosol source term was agreed to be an area where significant R&D is needed.
Hydrolysis of peroxides	**M**	**L**	Hydrolysis may not be lacking data, but aerosol behavior is the key concern.
Cavity Liner (ST05)			
Liner failure pressure or thermal response	**M**	**M**	Likely no composite model exists for liner failure (because of the complexity involved in the modeling process and the necessary constraints). Also, there are little data for combined effects.
Reaction Product Swelling Behavior	**L**	**L**	There is very little modeling known between steel and sodium mixed with sodium oxides, peroxides, and hydroxides at elevated temperatures
Corrosion of Liner	**M**	**M**	The Japanese conducted some steel immersed into sodium tests (Aoto et al., 1998; 2001).

Table 5. List of Research Gaps Associated with High Level Gap Topical Areas for Sodium Technology (cont.).

Gap	Experimental Database	Ability to Model	Details
Sodium-Concrete-Melt Interactions (ST06)			
Aerosol source term without melt	**L**	**L**	There are no known models.
Inert concrete-sodium interactions without melt	**M**	**L**	Some testing has been done at Cadarache, but this testing is insufficient to close the gap.
Basaltic concrete-sodium interactions without melt	**M**	**L**	Low ability to model because swelling is excluded in current models. Experiments have been performed but little confidence in whether the data are understood.
Limestone concrete-sodium interactions without melt	**M**	**M**	While the data are not understood, basic models are accurate for small-scale tests. Large scale testing and theoretical model development is needed.
Sodium-concrete reaction with sodium fire	**M**	**L**	Open chamber test and covered tests exist but additional testing may be needed to understand the underling phenomena.
Fission product dissolution and partitioning in melt and gases	**M**	**M**	Partitioning may not be accurately known.

Table 6 is a consolidated list of the highest-priority gap topical areas in the area of sodium technology. While detailed gaps were identified in the Sodium Technology gap report, Table 6 is intended to provide a high-level overview of the programs that need to be funded to close the remaining gaps. It should be noted that the H/M/L categorization was based on general observation of all relevant phenomena within each category, not solely the isolated gaps highlighted above. The colors in the table are provided to highlight differences in rankings.

Importance to safety H/M/L designations reflects the expert panel's belief of the relative importance of each gap in defending a SFR safety case. The gap report assigns values for importance to safety to each of the gaps listed in Table 5. The importance to safety within category column in Table 6 is an aggregated representation of the rankings associated with the individual gaps. The overall lowest knowledge state is the lowest value assigned to either ability to model gap or experimental database. Because not all gap reports listed both modeling and experimental states of knowledge, the overall lowest knowledge state is needed for comparison across gap reports.

Table 6. Sodium Technology Gap Topical Areas.

Gap ID	Name of Gap Topical Areas	Importance to Safety Within Category	Ability to Model Gap	Experimental Database	Overall Lowest Knowledge State
ST01	Sodium spray dynamics	H	L	L	L
ST02	Sodium-fluid interactions (S-CO_2)	H	H	L	L
ST03	Sodium-pool fire on an inert substrate	H	H	L	L
ST04	Aerosol dynamics	M	M	H	M
ST05	Sodium-cavity-liner interactions	H	M	M	M
ST06	Sodium-concrete-melt interactions	H	M	M	M
ST07	Sodium tech. knowledge management	H	N/A	N/A	M

4.2.3 Gap Closure Criteria

Table 7 summarizes expert opinions on topics that may be used to help prioritize funding of the gaps identified in Table 6. The colors in the table are provided to highlight differences in parameter ranges. The following acronyms in Table 7 have not been previously defined: Small Modular Reactor (SMR), Japan Atomic Energy Agency (JAEA), Commissariat à l'énergie atomique (CEA), Rosatom Russian Nuclear Energy State Corporation (ROSATOM). A brief description of facilities can be found in Table 19.

Table 7. Sodium Technology Gap Closure Estimates.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
ST01	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB/ Surtsey	-	SA	H/M	No
ST02	1M-10M	SMR / Solar and Fossil Energy***	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed / Scaled up version of SNAKE	DISCO2 (CEA)	DBE	H/M	Yes
ST03	1M-10M	Solar and Fossil Energy***	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB / Surtsey	SAPFIRE	SA	H/M	No
ST04	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10*	ST7	B308 AMPB / Surtsey	SAPFIRE (JAEA)	AOO	H/M	No
ST05	1M-10M	Solar and Fossil Energy***	JAEA / CEA / ROSATOM	5 – 10	5 – 10*	ST7	SNL**, ANL**	SAPFIRE (JAEA)	DBE	H/L	No
ST06	10M-100M	SMR	JAEA / CEA / ROSATOM	10 – 15*	5 – 10	ST7, Core-Concrete-Sodium Test Facility	MCCI	PLIINIUS (CEA) (SAPFIRE?)	SA	M/L	No
ST07	100K-1M	-	-	5 – 10*	< 5	0	ANL**, SNL**, PNNL**,	Japan**	O, AOO, DBE, BDBE, SA	H/M	No

* The experts did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** When the exact facility name is unknown, the laboratory designation is used.

*** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

4.3 Fuels and Materials

The Fuels and Materials gap analysis focused on the current licensability of in-core SFR fuels and materials. The status of ex-core materials was derived from a previously conducted gap report (Natesan, 2008).

4.3.1 Summary of Findings

The current state of knowledge of SFR fuel and structural material performance is sufficient for designing and licensing a SFR today within the envelope of the existing database. The boundaries of the existing database would be a fuel burnup of 10 at% or less, either metallic or oxide fuel, a peak cladding temperature of 600 ^{o}C or less, a peak dpa of 100 or less, and with fuel that has not been reprocessed. Both the steady-state and off-normal irradiation database would be sufficient to support such a design. The only qualifications to the above statement are the following: The existing data must be retrievable and in a form, from a quality assurance standpoint, that is acceptable to the licensing body. Fabrication experience for fuel, cladding, and ducts must also be retrieved to provide assurance that the core materials could be replicated such that the existing database is applicable. It must be appreciated that few, if any, vendors of these materials exist. Thus for fuel from zero to moderate burnup (10 at% or less), two gaps exist:

- An effort should be made to inventory the existing fuel performance database, collect the hard copy information and store it in approved storage locations, and transfer this information to an electronic database that can be readily queried.
- Exactly the same effort should be carried out for the fuel fabrication processes.

Should a reactor be designed for fuel burnup up to 20 at% the database weakens substantially for both metal and oxide fuel. The number of fuel pins taken to 20 at% is limited and these pins were not taken to high burnup without reconstitution of the irradiation subassembly in which they were located. Thus, there is no whole assembly experience or whole core experience at high burnup. Without the availability of a test reactor, a high burnup design could not be licensed. Thus, the major gaps for fuel irradiated beyond 10 at% are the following:

- A need for irradiation of a significant number of prototypic assemblies to high burnup in the steady-state conditions.
- Subject a number of high burn pins to off-normal tests in a fast-pulse reactor.

SFRs have been viewed as means to fission the minor actinides, americium, neptunium, and curium that arise from the reprocessing of LWR fuel in order to reduce the heat load and radio-toxicity of a spent fuel repository. The technological database is weak for either oxide or metal fuel that contains substantial quantities of minor actinides. Experiments are under way in the Advanced Test Reactor (ATR) to study the performance of metal and oxides fuel that contain additions of minor actinides. However, the fuel capsules are small and the neutron energy spectrum does not duplicate that of a fast reactor. The following gap exists for fuel with additions of minor actinides:

- Irradiation data gained from the ATR must eventually be augmented with the irradiation of full-size capsules in a SFR test reactor or modeled to the extent that the results from the small ATR capsules can be convincingly extrapolated to full size fuel pins.

It is unlikely that SFR fuel would be reprocessed using the PUREX process, which has minimal carryover of fission products to the reprocessed fuel. Pyro-processing or UREX have the potential for substantial carryover of fission products. The database is weak for the performance of either metal or oxide fuel that contains a substantial quantity of carried-over fission products. For oxide fuel, fabrication may be problematic; for metal fuel, the migration of lanthanide fission products to the fuel cladding interface may result in low melting compounds. Experiments are under way in the ATR to aid in the resolution of these issues. The following gap exists for fuel with fission product carryover:

- Irradiation data gained from the ATR must eventually be augmented with the irradiation of full-size capsules in a SFR test reactor or modeled to the extent that the results from the small ATR capsules can be convincingly extrapolated to full size fuel pins.

The last U.S. variation of 316 stainless steel, that being cold-worked with titanium and other alloy additions, designated as D9, is suitable for both oxide and metallic fuel cladding and ducts up to modest burnup levels and dpa less than 100. Vendors for this steel are readily available. The only identified gap was that more information is needed relative to fuel-cladding chemical interaction for reprocessed fuel with fission product carryover, particularly the issue of lanthanide migration to the fuel-cladding interface in metallic fuel (Mariani et. al., in press).

The ferritic/martensitic alloys have the potential to solve the irradiation enhanced swelling issue for both cladding and ducts up to at least 150 dpa and perhaps 200 dpa (Bridges et. al., 1993; Garner, 1994). However, the majority of the high dose data originates from a duct that operated at a relatively low temperature compared to fuel cladding temperatures. Thus the following gaps exist for both HT-9 and for the advanced cladding T91 (9Cr1Mo):

- High-dose/high-temperature swelling data do not exist for HT-9 or T91. Any data that exist or will be generated will originate from foreign SFRs.
- Recent attempts to obtain a small test sample of HT-9 revealed that there are no vendors readily available to produce reactor grade material. T91 is easier to acquire than HT9, but has a smaller metallic fuel experimental database.

Fuel performance information is used to validate fuel performance codes which are then used to support a license application. LIFE-METAL is the current metallic fuel performance code [Billone et al., 1986; ANL-IFR-169, 1992] which would support a SFR license application. It has been developed to predict the behavior of metallic fuel pins in fast reactors environment as a function of reactor operating history. Several gaps were identified in the discussion of fuel performance codes.

- Virtually all the gaps were related to the fact that there has been little attention given to fuel performance code development for the last two decades. Most of the code routines are empirically based as opposed to mechanistically based and thus are useful primarily for interpolation within the existing database.
- In addition, few people are adept in exercising the codes with documentation less than adequate for the training of new users.

In the area of structural materials it was noted that the panel borrowed from the results of previous gap analysis (Natesan, 2008). It was generally concluded that should a SFR be designed in the near future, using a Rankine cycle, that the technology base was likely adequate to license the reactor, provided that the burnup was limited to 10 at%. The only exceptions were associated with the incorporation to the SFR design of a rotating plug which is used to move fuel assemblies (this plug has yet to be designed) and large primary system electro-mechanical (EM) pumps (there is not fabrication capability to build a large EM pump, nor is there performance data to support licensing). The overall materials technological base for the Brayton cycle would require a significant research effort due to the combination of high temperatures, pressures, and small component sizes; though, this cycle offers many advantages to more traditional power cycles.

Two overarching gaps were apparent throughout the gap analysis discussions. These gaps were:

- Uncertainty in the preservation state of the existing knowledge base. Operating information, fuel performance data, and fabrication experience exist in a number of locations. Some exist on electronic media, which may or may not be queried easily, some on hard copy reports that are stored in substandard locations, and some may be lost.
- The need for an irradiation SFR facility such as EBR-II or FFTF and a transient behavior facility such as TREAT or Annular Core Research Reactor (ACRR) to enhance the existing knowledge base.

It is extremely important to preserve the existing database because without EBR-II, FFTF, and TREAT information cannot be duplicated. Even in the event that such facilities become available in the future, duplication of these irradiations would be expensive and time-consuming.

4.3.2 Identified Gaps

Three short-term and high-priority issues were identified as potentially needing immediate action:

- The new test segment of HT9 needs to be demonstrated to be comparable with the 1980s heat of HT9. This was identified as a challenge to the primary barrier to fission product release.
- Fuel performance codes such as the LIFE codes need to be maintained in terms of documentation, personnel, and funding. If this gap is not closed soon, no person in the DOE complex will have experience with these codes.
- A comprehensive knowledge management program is needed to not only record but analyze the FFTF, TREAT, and EBR-II data. Much of these data are not easily decipherable and will need experts from the corresponding facility to properly understand.

One high-priority gap that was not deemed as time-sensitive as the three listed above is a TREAT-type test to examine post-metal-fuel failure behavior under low-flow conditions. While TREAT restart may be the most obvious option to fill this gap, it is possible that these tests could be conducted in the ACRR at SNL. If post-failure fuel testing is to be performed at the ACRR, novel techniques may need to be developed to track fuel motion. It is estimated that full length fuel pins in multi-pin geometries can be analyzed in the ACRR. More information on both TREAT and ACRR can be found in Section 5.1.2.

The Fuels and Materials gap report identified 20 gaps of varying degrees of importance. In order to facilitate the prioritization process with gaps from the other four reports, these gaps were consolidated into the 10 topical areas shown in Table 8.

Table 8. List of Research Gaps Associated with High Level Gap Topical Areas for Fuels and Materials.

Gap	State of Knowledge	Details
	High-Burnup Fuel Characterization (FM01)	
Fuel swelling and Fuel Cladding Mechanical Interactions (FCMI) above 10at% burnup.	**M**	The experimental database between 10 at% and 20 at% is limited and almost non-existent above 20 at%
Gas release above 20 at% burnup.	**L**	The experimental database above 20 at% is almost non-existent. This phenomenon is understood below 20 at%.
Fuel Cladding Chemical Interactions (FCCI) above 10 at% burnup.	**M**	The experimental database between 10 at% and 20 at% is limited and almost non-existent above 20 at%. This gap is also highly dependent on the choice of cladding.
Fuel swelling and Fuel Cladding Mechanical Interactions (FCMI) above 10at% burnup.	**M**	The experimental database between 10 at% and 20 at% is limited and almost non-existent above 20 at%.
	Fission Product Carryover Fuel Characterization (FM02)	
Fuel swelling and FCMI above 10 at% burnup.	**M**	Fission product carryover fuel is not expected to behave differently to normal driver fuel for this gap.
Gas release above 20 at% burnup.	**L**	Fission product carryover fuel is not expected to behave differently to normal driver fuel for this gap.
FCCI at all burnups.	**L**	Very limited FCCI data exists for fuel with fission product carryover.
	MA Carryover Fuel Characterization (FM03)	
Fuel swelling and FCMI at all burnups.	**L**	Very limited data exists for fuel with MA.
Gas release at all burnups.	**L**	Very limited data exists for fuel with MA.
FCCI at all burnups.	**L**	Very limited data exists for fuel with MA.
	Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS (FM04)	
Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS	**M**	No vendors readily available to produce reactor grade HT-9. T91 is easier to acquire than HT9, but has a smaller metallic fuel experimental database.
	Advanced Cladding and Duct Material Properties (FM05)	
Creep rate at high temperature and dpa levels.	**M**	Limited data exists for the creep rates of advanced claddings in reactor environments

Table 8. List of Research Gaps Associated with High Level Gap Topical Areas for Fuels and Materials (cont.).

Gap	State of Knowledge	Details
	Duct/Bundle Performance Experience (FM06)	
Potential loss of historical database.	**L**	Validation database for bundle performance codes such as NUBOW are currently missing.
Bundle-bundle interactions at all temperatures and dpa levels.	**M**	If validation database can be found, knowledge of Stainless Steel 316 bundle performance is adequate. No experimental data exists for advanced cladding full core bundle performance.
Bundle-duct interactions at all temperatures and dpa levels.	**M**	If validation database can be found, knowledge of Stainless Steel 316 bundle performance is adequate. No experimental data exists for advanced cladding full core bundle performance.
	Structural Material Issues (FM07)	
Rotating Plug general knowledge.	**L**	Materials and surrounding environment has not been selected, thus large uncertainties exist regarding expected performance
Intermediate Heat eXchanger (IHX) degradation mechanisms.	**M**	Knowledge of degradation mechanisms should be improved to inform maintenance and service life decisions.
Electro Magnetic (EM) Pump fabrication capability and operational experience for large pumps.	**L**	Small EM pumps are available but large EM pumps, such as those needed for the primary system, are not currently manufactured nor does the capability currently exist.
	Brayton (S/CO_2) Materials Issues (FM08)	
Brayton (S/CO_2) Materials Issues	**L**	Most components, excluding potentially the Recuperators, need to select potential materials, develop fabrication capacity and improve the experimental database.
	SFR Fuels and Materials Knowledge Base Preservation (FM09)	
SFR Fuels and Materials Knowledge Base Preservation	**L**	The identification, consolidation, and interpretation of information needed for licensing is an ongoing but underfunded process.
	Fuel Performance Code Documentation and Training Issues (FM10)	
Fuel Performance Code Documentation and Training Issues	**L**	LIFE-Metal currently has one steward who can locate the supporting database for LIFE-Metal. Updates to the code have been performed but not documented.

Table 9 is a consolidated list of the highest-priority gap topical areas for fuels and materials. The colors in the table are provided to highlight differences in importance.

Importance to safety H/M/L designations reflects the expert panel's belief of the relative importance of each gap in defending a SFR safety case. The gap report assigns values for importance to safety to each of the gaps listed in Table 8. The importance to safety within category column in Table 9 is an aggregated representation of the rankings associated with the individual gaps. The Fuels and Materials gap reports did not list both modeling and experimental states of knowledge for each gap, thus the reported knowledge state represents the overall lowest knowledge state for the gap or gap topical area.

Table 9. Fuels and Materials Gap Topical Areas.

Gap ID	Name of Gap Topical Areas	Importance to Safety Within Category	State of Knowledge
FM01	High Burnup Fuel Characterization	H	M
FM02	Fission Product Carryover Fuel Characterization	H	L
FM03	MA Carryover Fuel Characterization	H	L
FM04	Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS	H	M
FM05	Advanced Cladding and Duct Material Properties	H	M
FM06	Duct/Bundle Performance Experience	H	L
FM07	Structural Material Issues, Rotating Plug, IHX, EM Pump	M	L
FM08	Brayton (S/CO_2) Materials Issues	H	L
FM09	SFR Fuels and Materials Knowledge Base Preservation	H	L
FM10	Fuel Performance Code Documentation and Training Issues	H	L

4.3.3 Gap Closure Criteria

Table 10 summarizes expert opinions on topics that may be used to help prioritize funding of the gaps identified in Table 9. Most of these gaps fall under the purview of the Advanced Fuels Campaign (AFC) and are currently receiving active funding, although many of these gaps cannot be completely filled without access to a fast neutron irradiation facility. A more detailed overview of the current work being conducted by the AFC can be found in Section 5.1.3.1. Indeed the only fuels and materials gaps that are not currently receiving sufficient funding are

- FM1, which cannot be filled without a robust irradiation program, and
- FM9 and FM10, which concern preservation of historical resources for future use.

The colors in the table are provided to highlight differences in the parameter ranges assigned to each subject.

Table 10. Fuels and Materials Gap Closure Estimates.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
FM01	+100M*	AFC	Not for Metal Fuel	15 +	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	M/M	Yes
FM02	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated FP fuel	-	CEFR / BN60	O	H/M	Yes
FM03	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated MA fuel	-	CEFR / BN60	O	H/M	Yes
FM04	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	-	-	-	O	H/L	Yes
FM05	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	H/L	Yes
FM06	1M-10M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM07	10M-100M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10	Large sodium test loop	ANL**	-	O , AOO , DBE , BDBE , SA	H/M	No
FM08	1M-10M	Solar & Fossil Energy***	Not for Metal Fuel	5 – 10*	5 – 10	Coupled Na/S-CO_2 test loop	-	-	DBE	H/M	Yes
FM09	100K-1M	AFC	Not for Metal Fuel	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/M	No
FM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No

* The experts did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** When the exact facility name is unknown, the laboratory designation is used.

*** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

4.4 Source Term Gap Analysis

The Source Term Characterization gap analysis focused on the current ability to accurately track radionuclides through the fuel, coolant, primary system, containment, and into the environment.

4.4.1 Summary of Findings

The technical areas of most immediate interest are those that are thought to have a high importance to the mechanistic modeling of the source term for a sodium-cooled fast reactor and a high need for research. Only seven issues were identified by the experts. Without further experimental information in these areas, the mechanistic modeling of the source term would be judged by the experts as seriously deficient and potentially unreliable.

The next tier of interest is the class of phenomena that have a high importance for modeling the accident source term but only a medium need for additional research. This classification implies that there is some understanding of the phenomena and even some data, but this understanding could be substantially enhanced by further research. This improved understanding could be expected to improve substantially the accuracy and the reliability of the model predictions.

The only issue identified by the experts as having a medium importance but a high need for additional research was the issue of bubble swarm rise velocities in sodium pools.

All other relevant phenomena were thought to have a medium importance and no more than a medium need for additional research. The experts felt that these issues might better be addressed once first steps had been taken to develop a mechanistic model. Phenomena of medium importance should be included in a “first cut” model. Sensitivity and uncertainty analysis of the model could then provide a more quantitative indication of the need for more experimental investigation of topics of medium importance.

4.4.2 Identified Gaps

The Source Term Characterization gap report identified 20 gaps of varying degrees of importance. To facilitate the prioritization process with gaps from the other four reports, these gaps were consolidated into the following five topical areas listed in Table 11.

Table 11. List of Research Gaps Associated with High Level Gap Topical Areas for Source Term Characterization.

Gaps	State of Knowledge
Radionuclide Release From Fuel Debris Into a Quiescent Sodium Pool (STC01)	
High-temperature release of radionuclides from fuel during a temperature excursion event.	**H**
Fuel morphology and the rates of radionuclide leaching by liquid sodium.	**H**
Rates of fuel dissolution or ablation in a liquid sodium pool.	**M**

Table 11. List of Research Gaps Associated with High Level Gap Topical Areas for Source Term Characterization (cont.).

Gaps	State of Knowledge
Radionuclide Behavior in Containment (STC02)	
Thermal decomposition of sodium iodide in the containment to form molecular iodine.	**H**
Reaction of iodine species in the containment to form volatile organic iodides.	**H**
Revaporization of radionuclide deposits in the reactor coolant system.	**M**
Radionuclide Transport Within a Sodium Pool (STC03)	
Enrichment of free surfaces of sodium by dissolved or suspended radionuclides.	**H**
Sodium vapor bubble growth and scrubbing of radionuclides from the bubble during a thermal excursion and fuel failure.	**M**
Mass transport within a rising sodium vapor and noble gas bubble that results in the deposition of radionuclide particles and vapor into liquid sodium.	**M**
Fission bubble transport in the sodium pool.	**M**
Solubility of radionuclides in sodium containing various amounts of dissolved oxygen.	**M**
Nucleation and growth of radionuclide particles in liquid sodium.	**M**
Bubble swarm rise velocities in sodium pools.	**M**
Plateout of dissolved radionuclides on structural surfaces within the bulk sodium pool or at its perimeter	**M**
Plateout of dissolved radionuclides on solids suspended in the sodium pool, or	**M**
Radionuclide Chemistry in Sodium Bond Between Fuel and Cladding (STC04)	
Entrainment during fuel rod depressurization of radionuclide-contaminated, liquid sodium, making up the “sodium bond” between metal fuel and the cladding.	**H**
Accumulation during normal operations of radionuclides in the sodium bond in metal fuel.	**M**
Chemical form of radionuclides in the fuel and the fuel-cladding gap.	**M**
Chemical activities of radionuclides in the fuel.	**M**
Mechanical Release of Radionuclides From the Surface of a Sodium Pool (STC05)	
Gas phase velocity over the sodium pool (thermal hydraulics issue).	**M**
Multicomponent gas phase diffusion of radionuclides across the boundary layer at the gas-liquid sodium interface.	**M**
Entrainment of liquid sodium into the gas phase by the bursting of bubbles at the sodium surface.	**M**

Because of the lack of recent U.S. research in this area, the chair of the Source Term gap analysis included a brief description of the type of experiment that would be required to fill the identified licensing gaps.

Radionuclide Release From Fuel Debris Into a Quiescent Sodium Pool (STC01)

Energetic fuel-coolant interactions that could produce substantial radionuclide release can only take place when the fuel is substantially molten. Experimental investigations can be undertaken with simulant fuel rather than with irradiated fuel. Some planning of such experiments is being considered at the Cadarache Research Centre in France.

Experimental studies of radionuclide leaching by sodium from reactor fuel will require the use of irradiated fuel. Interfacial actions of a solvent like sodium and radionuclides are too complicated to accurately predict. The experiments need not be done in-pile, but they need to be done in realistic temperature and pressure regimes with sodium appropriately contaminated with dissolved oxygen. Because the experiments will use irradiated fuel, they will have to be done in hot cells. Suitable locations exist in the United States such as facilities at Idaho National Laboratory (INL) and ORNL. Experience with similar investigations and suitable facilities are also to be found at the TransUranium Institute in Germany with modifications to their existing facilities. Several years of experimental investigations will be needed to compile a comprehensive database with respect to the radionuclides of interest and the temperature regimes of interest. It is unlikely that a meaningful experimental research program can be undertaken for less than $1 million per year. The program should not be initiated without at least the promise of realistically irradiated fuel being available once experimental capabilities have been developed and tested with non-radioactive materials.

Release of radionuclides from fuel into flowing sodium during a temperature excursion event will probably need to be experimentally investigated in-reactor in sodium loop. Release conditions will depend upon the particular accident sequence that causes cladding breach. In general, breaching is anticipated to create coolant channel pressures on the order of the gas-plenum pressure of the pin prior to breach.

Radionuclide Behavior in Containment (STC02)

Radionuclides released from the sodium pool will be in the form of vapors or aerosol particles. As the radionuclides are convected away from sodium surface, they will encounter cooler conditions. Flow pathways will be disrupted by structures. As a result, both vapors and aerosols will deposit on surfaces. Aerosol particles may deposit by inertial mechanisms, turbulent mechanisms, or simple gravitational settling. Vapors may condense or chemically react with structural surfaces. Radionuclides accumulated on a surface will be heated by the decay heat of the radionuclides as well as by any convective heating that can occur. Temperatures in the accumulation may be sufficient that radionuclides can revaporize from the deposits. Highly volatile radionuclides such as isotopes of iodine, cesium, and antimony are particularly susceptible to revaporization. Notably, metal iodides can decompose to produce iodine as molecular iodine or other gaseous form. The counter ion reacts with the surface. Aerosol deposits can be resuspended in the gas flow through the reactor system by either changes in the flow velocity of the ambient gas phase or as a result of shock and vibration of the substrate for the deposited particles.

Traditionally, testing of deposition, revaporization, and resuspension have been done in out-of-pile experimental programs with unirradiated material. For aerosol studies, it is critical to have proper geometries and flow velocities. For vapor interactions with surfaces, it is critical to have good representations of actual surface materials and surface conditions. It has been found that vapors often react with low concentration impurities in surface materials rather than with the bulk materials. It has also been found that surface stresses affect reactivity. Meaningful work can be done with modest projects funded at levels of about $0.5 million per year. Work can really only be undertaken when there are specifications of designs and materials as well as some understanding of how surfaces be changed by events of an accident. A useful experimental and modeling database could be established in less than five years. Many of the national laboratories and universities have capabilities suitable for the required investigations. There is similar work with respect to aerosol under way at the Paul Scherrer Institut in Switzerland though it is not now directed toward sodium reactors. Pertinent theoretical work and experimental work on aerosol is being undertaken in the United Kingdom under the direction of M. Reeks. Again, the work is not now directed toward analysis of sodium-cooled reactors. Modeling work appropriate for analysis of revaporization is being planned at the Cadarache Nuclear Centre in France. SNL's VICTORIA model could be modified to address the revaporization issue and to identify critical issues needing experimental investigation.

Radionuclide Transport Within a Sodium Pool (STC03)

Sodium will "leach" radionuclides from exposed fuel. At the sodium-fuel interface, the concentrations of radionuclides can reach saturation. Convective forces will pull the sodium concentrated in radionuclides away from the exposed fuel surfaces and mix this concentrated fluid with the bulk sodium. Bulk sodium will necessarily be cooler than sodium at the interface with the quenched fuel. Consequently, bulk sodium may become supersaturated in dissolved radionuclides. Supersaturation may be relieved by:

- Plateout of dissolved radionuclides on structural surfaces within the bulk sodium pool or at its perimeter, or
- Plateout of dissolved radionuclides on solids suspended in the sodium pool, or
- Nucleation of particles within the bulk sodium pool.

Research needed to understand regimes where the various possible pathways for relieving radionuclide supersaturation will dominate can be done in small-scale, but well-scaled, tests. Because of the low levels of solubility for many of the radionuclides of interest, experimental investigations may be facilitated by the use of radioactive tracers, which will necessitate the use of specialized hot cells or hot boxes. Similar research facilities can be used to investigate the entrainment of sodium by evolved gases. A meaningful research program can be undertaken at funding levels of about $0.5 million per year. An adequate database should be in hand within five years at this level of funding. A key point of investigation in the study will be confirmation of German findings that entrainment of uranium particles can take place without either high gas flows or gases bubbling through the liquid sodium.

Radionuclide Chemistry in Sodium Bond Between Fuel and Cladding (STC04)

Studies of radionuclide release associated with the rupture of cladding and the expulsion of bond material are needed only if metallic fuel is selected for the sodium-cooled reactor. Such "gap release" experiments will have to be done with irradiated fuel. The water-cooled pool type reactor CABRI previously had a sodium loop suitable venue for in-pile testing, although this capability has been lost. In the past, such tests have been done out of pile with single rods.

Because of the use of irradiated rods, specialized facilities are needed. Such facilities exist at INL and ORNL. They would require substantial modifications for the needed testing. It is likely that only a few tests would be needed.

Mechanical Release of Radionuclides From the Surface of a Sodium Pool (STC05)

Regardless of the pathway adopted for relief of any supersaturation produced by leaching of radionuclides from fuel, the bulk sodium will contain dissolved radionuclides and suspended radionuclide particles. Gas bubbles rising through the sodium pool can entrain radionuclide contaminated sodium at the surface. The rising gases may be either sodium vapor if the bulk sodium is boiling or fission gases such as xenon and krypton being released from the fuel. Gases may also be produced by sodium reactions with materials to form hydrogen and the like. At low gas flows, entrainment of contaminated liquids will occur when bubbles burst at the surface. Bubble bursting produces very high accelerations that fragment liquid sodium into droplets of aerosol dimensions. At high gas flows either through or over the sodium, entrainment of bulk sodium can occur. For surface flows, the entrainment is thought to occur by shearing of capillary waves on the surface.

Table 12 is a consolidated list of the highest-priority gap topical areas for the source term. While detailed gaps were identified in the Source Term, Table 12 is intended to provide an overview of the types of programs that need to be funded to close the remaining gaps. The colors in the table are provided to highlight the differences in the ranking of the identified gaps.

Importance to safety H/M/L designations reflects the expert panel's belief of the relative importance of each gap in defending a SFR safety case. The gap report assigns values for importance to safety to each of the gaps listed in Table 11. The importance to safety within category column in Table 12 is an aggregated representation of the rankings associated with the individual gaps. The Source Term gap reports did not list both modeling and experimental states of knowledge for each gap, thus the reported knowledge state represents the overall lowest knowledge state for the gap or gap topical area.

Table 12. Source Term Gap Topical Areas.

ID	Name of Gap Topical Areas	Importance to Safety Within Category	State of Knowledge
STC01	Radionuclide release from fuel debris into a quiescent sodium pool	H	L
STC02	Radionuclide behavior in containment	H	L
STC03	Radionuclide transport within a sodium pool	H	M
STC04	Radionuclide chemistry in sodium bond between fuel and cladding	H	M
STC05	Mechanical release of radionuclides from the surface of a sodium pool	H	M
STC06	MELCOR/Contain-LMR Integration	H	L

4.4.3 Gap Closure Criteria

Table 13 summarizes expert opinions on topics that may be used to help prioritize funding of the gaps identified in Table 12. The colors in the table are provided to highlight the different ranges for the evaluation parameters.

Table 13. Source Term Gap Closure Estimates.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
STC1	1M-10M	AFC	-	10 – 15*	< 5	-	-	Cadarache Research Center**	BDBE, SA	H/L	No
STC2	1M-10M	-	-	10 – 15*	< 5	-	SNL (VICTORIA code)	JAEA (Contain-LMR)	SA	H/L	No
STC3	10M-100M	AFC	-	10 – 15*	< 5	Irradiated Fuel	Hot Cell Facilities at INL and ORNL	TransUranium Institute**	BDBE, SA	H/M	No
STC4	10M-100M	AFC	-	10 – 15*	< 5	Irradiated Fuel, Upgraded Domestic Facilities	Upgraded Out-of-Pile Test Facilities at INL and ORNL	CABRI Reactor	BDBE, SA	H/L	No
STC5	1M-10M	AFC	-	10 – 15*	< 5	Irradiated Fuel	SNL (VICTORIA code)	Paul Scherrer Institut**, Cadarache** (Modeling)	SA	H/L	No
STC6	1M-10M	-	-	10 – 15*	< 5	-	SNL (Contain-LMR/MELCOR)	JAEA (Contain-LMR)	DBE, BDBA, SA	H/M	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** When the exact facility name is unknown, the laboratory designation is used.

4.5 Codes and Models Gap Analysis

The Codes and Models gap analysis focused on the current state of SFR computer analysis tools and their readiness to support a license application.

4.5.1 Summary of Findings

This report qualitatively assessed currently available computer codes and models for accident analysis and reactor safety calculations of advanced sodium fast reactors.

Three assessment categories were defined for use during the review. These are titled "Code Maturity Level," "Fidelity Adequacy," and "Code Support Status." The maturity level assessment was further subdivided into the issues of code and solution verification, software quality engineering, and code validation. The geometric representation and the physics modeling were also considered separately for the Fidelity Adequacy assessment.

The assessment results were presented in the form of nine tables, organized into groups of three for each risk category. For each risk category the first table summarizes the assessment ratings and scores from the panel members. The second table in each set provides a compilation of short notes that panel members added for context or clarification. The third table in each set is a compilation of reviewer responses to the question posed about the most significant gap or weakness (limited to U.S. computer codes) in each risk category.

Only a limited and partial assessment of codes for sodium leakage scenarios is provided because only one expert panel member felt qualified to provide input. Additional efforts may need to be pursued in another setting to obtain a more satisfactory assessment of codes available for these scenarios.

The following is a bulleted list of notable conclusions that can be drawn from the assessment:

- Although current U.S. codes are primarily legacy tools that do not leverage advanced computational technologies, they are adequate for licensing as long as the required safety margins are significant. However, in general the panel did not rate available U.S. codes adequate if the required safety margins are small.
- Support of available SFR U.S. safety codes is considered weak, and concerns were expressed about the loss of knowledgeable and experienced users for these codes. Reactor safety codes model many interacting and complex phenomena and must be applied by knowledgeable users who understand both the computer code (e.g., the numerics, models, limitations, etc.) and the underlying physics being simulated.
- When assessing code maturity, panel members generally gave lower scores to the "Validation with Uncertainty Quantification and Sensitivity Analysis" subcategory than to the other subcategories. This subcategory relates to the quality of the quantification, not the accuracy of the model itself. Based on panel discussions, an important reason for this is the lack of high-quality data, such as validation and verification data for complex reactor geometries.
- In general, seismic event driven scenarios and severe accident scenarios have the lowest assessment scores. This reflects a view that the most significant gaps are in settings where the geometry is uncertain or changes with time because of fuel rod failure, blockage or voiding with large reactivity insertions, and are directly affected by the

accident initiation. These types of scenarios can result in large changes in reactivity feedback in the reactor core.

- From a code modeling perspective, panel members identified the following weaknesses or gaps.
    - Models for transient natural convection processes in the reactor system.
    - The need for improved subchannel and multi-pin analysis capabilities.
    - The modeling of gas bubble entrainment and the effects of sodium–water interaction.
    - Lack of advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content.
    - Models to predict source term releases from fuel in LMR accidents.

It was clear from this activity that in the United States the SAS4A/SASSYS-1 code system would be a central tool used in the analysis of a large majority of the scenarios considered here, and that it was generally assessed as adequate to support these activities for licensing. However, several panel members highly recommended that work was needed to support modernization of the code architecture and to establish a more vigorous code verification and quality assurance plan for code maintenance, configuration management/control, and testing of software through improved Software Quality Engineering (SQE) practices. In their view modernization of the code system was needed to:

- Support updating the memory management scheme to remove various nodalization limits,
- Support parallel applications, and
- Create an input processor and user interface to improve user friendliness and reduce potential input errors.

Such an activity would improve the performance of the code system by taking advantage of standard parallel computing platforms and making codes suitable for applications beyond the standard use. Such applications could include running SAS4A/SASSYS-1 calculations as the simulation engine for automated design optimization, uncertainty quantification, and sensitivity analysis schemes. It was suggested that if a SFR design is to withstand the regulatory scrutiny, the software system that supports the license application will likely be required to have these capabilities in place.

While the codes and methods report focused on the licensing acceptability of SAS4A, two additional codes were discussed that need increased funding:

- MELCOR (LMR) –The U.S. NRC helped developed MELCOR for LWR severe accident analysis. Integration of Contain-LMR, which is not currently supported in the U.S., with MELCOR would create a well-maintained radionuclide tracking, structure performance and containment response code. Additionally, by integrating Contain-LMR with MELCOR, the U.S. NRC will be more familiar with the licensing toolset.
- This code needs to be written and, eventually, a user base will need to be established.
- LIFE-METAL – SFR fuel performance code. This code's underlying validation database documentation needs to be updated and a user-base needs to be established. Currently, there is one code user.

4.5.2 Identified Gaps

The Codes and Models gap analysis report identified 13 gaps of varying degrees of importance. To facilitate the prioritization process with gaps from the other four reports, these gaps were consolidated into the following five topical areas listed in Table 14. The codes and models report evaluated the degree of code maturity, fidelity, and Verification and validation for a variety of accident scenarios. This evaluation process required a "post-processing" step to determine the importance to safety and state of knowledge from the report.

Table 14. List of Research Gaps Associated with High Level Gap Topical Areas for Codes and Methods.

Gap	State of Knowledge	Details
Modeling of Seismic Events (CM01)		
Experimental SFR seismic data	**L**	Lack of seismic data associated with LMR performance (e.g., coolant movement into or out of an assembly, assembly distortions)
Common cause effects of a seismic event on the reactor systems	**M**	Specifically oscillatory motion of the structure of the core and reactivity feedback given physics uncertainties.
Models for Transient Natural Convection Processes in the Reactor System (CM02)		
Modeling of transient natural convection	**M**	Validation data for complex reactor geometry and model development
LIFE-Metal/Life 4 Update (CM03)		
Re-calibration and validation of LIFE-METAL	**L**	Calibration and validation effort of the code should be conducted once further data from EBR-II experiments are collected through current knowledge management programs
Sub-channel and Multi-pin Analysis Capabilities (CM04)		
Sub-channel + multi-pin analysis capability	**L**	An improved sub-channel + multi-pin analysis capability (to simulate entire sub assembly and flow blockages) would be beneficial as an additional modeling option under SASSYS-1.
Modeling of Gas Bubble Entrainment and the Effects of Sodium–Water Interaction (CM05)		
Modeling of Gas Bubble Entrainment and the Effects of Sodium–Water Interaction	**L**	Physics models sodium–steam/water interaction are complex and need to be improved.
Advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content (CM06)		
High-Actinide Fuel Performance Models	**L**	Lack of advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content.
Models to predict source term releases from fuel in LMR accidents (CM07)		
Predict source term releases from fuel in LMR accidents	**L**	See Section 4.4 on Source Term Characterization

Table 14. List of Research Gaps Associated with High Level Gap Topical Areas for Codes and Methods (cont.).

Gap	State of Knowledge	Details
	SAS4A Code Modernization, Support and Knowledgeable User-base (CM08)	
Support updating the memory management scheme to remove various nodalization limits	**L**	The internal structure of SAS4A has not been updated since the early 1990s. New computational techniques can be employed which will take advantage of current generation hardware.
Support parallel applications	**L**	New computational techniques can be employed which will take advantage of current generation hardware.
Create an input processor and user interface to improve user friendliness and reduce potential input errors	**L**	Most current state-of-the-art codes have graphical user interfaces which can ease new users into the code and can help debug input files.
	MELCOR/Contain LMR Update (CM09)	
MELCOR/Contain LMR Update	**L**	Integration of Contain-LMR, which is not currently supported in the U.S., with MELCOR would create a well-maintained radionuclide tracking, structure performance and containment response code.
	Fuel Performance Code Documentation and Training Issues (CM010)	
Documentation of LIFE-METAL	**L**	A detailed documentation and a revalidation effort is needed to release LIFE-METAL to the national code center. This will also help increase the user base of the code

Table 15 is a consolidated list of the highest-priority gap topical areas in the area of computer codes and models. The colors in the table are provided to highlight differences in the ranking of the identified gaps.

Table 15. Codes and Models Gaps.

ID	Name of Gap Topical Areas	Importance to Safety Within Category*	State of Knowledge*
CM01	Modeling of Seismic Events	H	M
CM02	Models for transient natural convection processes in the reactor system	M	M
CM03	LIFE-Metal/Life 4 Update	H	L
CM04	Subchannel and multi-pin analysis capabilities	M	M
CM05	Modeling of gas bubble entrainment and the effects of sodium–water interaction	M	L
CM06	Advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content	H	L
CM07	Models to predict source term releases from fuel in LMR accidents	H	L
CM08	SAS4A Code Modernization, Support and Knowledgeable User-base	H	L
CM09	MELCOR/Contain LMR Update	H	L
CM10	Fuel Performance Code Documentation and Training	H	L

* Due to the ranking criteria used in the Codes and Models Report, values for Importance to Safety and State of Knowledge were impossible to directly extract from the report. Instead, the authors of this report consulted with the Codes and Models Chair to assign H/M/L values to the identified gaps.

4.5.3 Gap Closure Criteria

Table 16 summarizes expert opinions on topics that may be used to help prioritize funding of the gaps identified in Table 15. It can be noted that because of the nature of this report, focusing on computer codes, no facilities were directly identified in the gap closure table. The colors in the table are provided to highlight differences in the ranges of the evaluation parameters.

Table 16. Codes and Models Gap Closure Estimates.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
CM01	1M-10M	-	JAEA	5 – 10	< 5	-	-	-	DBE, BDBA, SA	H/M	No
CM02	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	DBE, BDBA, SA	H/L	No
CM03	100K-1M	AFC	-	5 – 10	5 – 10	Senior personnel, FM09	-	-	O , AOO , DBE , BDBE , SA	M/L	No
CM04	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	O , AOO , DBE , BDBE , SA	H/L	No
CM05	100K-1M	-	-	5 – 10	< 5	-	-	-	DBE	H/L	No
CM06	1M-10M	-	-	15 +	< 5	TREAT / ACRR	-	-	BDBE, SA	H/L	No
CM07	1M-10M*	-	-	5 – 10	< 5	-	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM08	100K-1M	-	-	< 5	< 5	-	-	-	DBA BDBA SA	H/L	No
CM09	1M-10M*	-	-	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

4.6 Summary of All Gap Reports

Table 17 summarizes the expert opinions from all five gap analysis reports that may be used to help prioritize funding of the identified gaps. Table 18 summarizes the gap closure estimates covering the gap topical areas listed in Table 17.

The following guidelines were used in the formation of these tables:

- Values were taken directly from the gap analysis reports, or extrapolated from the reports if no direct value was given in the report.
- If the gap is associated with a design option that is not typical of a generic SFR design, the value indicated in the Importance to Safety Within Category column evaluates the safety significance of that gap if that design option is to be used. Judgments concerning the relative importance of the design option are left to the decision maker.
- Because of the ranking criteria used in the Codes and Models Report, values for Importance to Safety and State of Knowledge were impossible to directly extract from the report. Instead, the authors of this report consulted with the Codes and Models Chair to assign H/M/L values to the identified gaps.
- The colors are only intended to highlight differences in cell values; no judgment is intended by the choice of any color.
- In Table 18, the authors did not reach a consensus regarding some ranges regarding the timing or cost associated with closing a gap. These estimates are marked with the following symbol: *. In each case, the highest range was placed in the table.

Table 17. All Gap Topical Areas.

ID	Name of Gap	Importance to Safety Within Category	State of Knowledge
AIS01	Steady State Intact Fuel and Fuel Changes	H	M
AIS02	Transition to Natural Convective Cooling, Sodium Stratification	H	M
AIS03	Thermal Response of Structures, Thermal Striping	H	M
AIS04	Decay Heat Rejection, Radiation Heat Transfer from Vessels	H	M
AIS05	Power Conversion Cycle, S-CO_2 Accident Analysis	H	L
AIS06	Fuel Transient Behavior – Length Effects	M	L
AIS07	Severe Core Damage, Metal Fuel Motion, Dispersal and Morphology	H	M
AIS08	Seismic Isolator	H	M
CM01	Modeling of Seismic Events	H	M
CM02	Models for transient natural convection processes in the reactor system	M	M
CM03	LIFE-Metal/Life 4 Update	H	L
CM04	Subchannel and multi-pin analysis capabilities	M	M
CM05	Modeling of gas bubble entrainment and the effects of sodium–water interaction	M	L
CM06	Advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content	H	L
CM07	Models to predict source term releases from fuel in LMR accidents	H	L
CM08	SAS4A Code Modernization, Support and Knowledgeable User-base	H	L
CM09	MELCOR/Contain-LMR Update	H	L
CM10	Fuel Performance Code Documentation and Training Issues	H	L

Table 17. All Gap Topical Areas (cont.).

ID	Name of Gap	Importance to Safety Within Category	State of Knowledge
FM01	High Burnup Fuel Characterization	H	M
FM02	Fission Product Carryover Fuel Characterization	H	L
FM03	MA Carryover Fuel Characterization	H	L
FM04	Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS	H	M
FM05	Advanced Cladding and Duct Material Properties	H	M
FM06	Duct/Bundle Performance Experience	H	L
FM07	Structural Material Issues, Rotating Plug, IHX, EM Pump	M	L
FM08	Brayton (S/CO_2) Materials Issues	H	L
FM09	SFR Fuels and Materials Knowledge Base Preservation	H	L
FM10	Fuel Performance Code Documentation and Training Issues	H	L
ST01	Sodium spray dynamics	H	L
ST02	Sodium-fluid interactions (S-CO_2)	H	L
ST03	Sodium-pool fire on an inert substrate	H	L
ST04	Aerosol dynamics	M	M
ST05	Sodium-cavity-liner interactions	H	M
ST06	Sodium-concrete-melt interactions	H	M
ST07	Sodium Tech. Knowledge Management	H	M
STC01	Radionuclide release from fuel debris into a quiescent sodium pool	H	L
STC02	Radionuclide behavior in containment	H	L
STC03	Radionuclide transport within a sodium pool	H	M
STC04	Radionuclide Chemistry in Sodium Bond between Fuel and Cladding	H	M
STC05	Mechanical release of radionuclides from the surface of a sodium pool	H	M
STC06	MELCOR/Contain-LMR Integration	H	L

Table 18. All Gap Closure Estimates.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
AIS01	10M-100M*	AFC	JAEA / CEA / ROSATOM	10 – 15*	5 – 10	Fuel Testing Facility	ATR and/or HFIR	JOYO	DBA, BDBA	H/M	No
AIS02	1M-10M*	-	Most GIF Members	10 – 15*	< 5	Sodium Component Test Facility	-	PLANDTL, CCTL	DBA, BDBA	H/M	No
AIS03	1M-10M*	-	-	10 – 15*	< 5	Sodium Component Test Facility	MELT Facility once Built	PLANDTL, CCTL	DBA, BDBA	M/M	No
AIS04	1M-10M	-	JAEA / CEA / ROSATOM	10 – 15*	< 5	-	NSTF	ATHENA	DBA, BDBA	H/L	Yes
AIS05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	10 – 15*	< 5	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	-	DBA, BDBA	M/M	Yes
AIS06	10M-100M*	AFC	-	10 – 15	< 5	Irradiated Fuel and (TREAT or ACRR)	TREAT or ACRR	-	DBA, BDBA	M/L	No
AIS07	10M-100M*	AFC	-	10 – 15	< 5	Irradiated Fuel and (TREAT or ACRR)	TREAT or ACRR, Possibly CAMEL	MELT-II	SA	H/L	No
AIS08	1M-10M	-	JAEA	10 – 15	5 – 10	-	-	-	DBA BDBA	H/L	Yes
CM01	1M-10M	-	JAEA	5 – 10	< 5	-	-	-	DBE, BDBA, SA	H/M	No
CM02	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	DBE, BDBA, SA	H/L	No
CM03	100K-1M	AFC	-	5 – 10	5 – 10	Senior personnel, FM09	-	-	O , AOO , DBE , BDBE , SA	M/L	No
CM04	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	O , AOO , DBE , BDBE , SA	H/L	No
CM05	100K-1M	-	-	5 – 10	< 5	-	-	-	DBE	H/L	No
CM06	1M-10M	-	-	15 +	< 5	TREAT / ACRR	-	-	BDBE, SA	H/L	No
CM07	1M-10M*	-	-	5 – 10	< 5	-	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM08	100K-1M	-	-	< 5	< 5	-	-	-	DBA BDBA SA	H/L	No
CM09	1M-10M*	-	-	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No

Table 18. All Gap Closure Estimates (cont.).

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
FM01	+100M*	AFC	Not for Metal Fuel	15 +	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	M/M	Yes
FM02	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated FP fuel	-	CEFR / BN60	O	H/M	Yes
FM03	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated MA fuel	-	CEFR / BN60	O	H/M	Yes
FM04	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	-	-	-	O	H/L	Yes
FM05	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	H/L	Yes
FM06	1M-10M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM07	10M-100M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10	Large sodium test loop	ANL	-	O , AOO , DBE , BDBE , SA	H/M	No
FM08	1M-10M	Solar & Fossil Energy**	Not for Metal Fuel	5 – 10*	5 – 10	Coupled Na/S-CO_2 test loop	-	-	DBE	H/M	Yes
FM09	100K-1M	AFC	Not for Metal Fuel	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/M	No
FM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
ST01	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB/Surtsey	-	SA	H/M	No
ST02	1M-10M	SMR / Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	DISCO2	DBE	H/M	Yes
ST03	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB/Surtsey	SAPFIRE	SA	H/M	No
ST04	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10*	ST7	B308 AMPB/Surtsey	SAPFIRE	AOO	H/M	No
ST05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10	5 – 10*	ST7	SNL, ANL	SAPFIRE	DBE	H/L	No
ST06	10M-100M	SMR	JAEA / CEA / ROSATOM	10 – 15*	5 – 10	ST7, Core-Concrete-Sodium Test Facility	MCCI	PLIINIUS SAPFIRE?	SA	M/L	No
ST07	100K-1M	-	-	5 – 10*	< 5	Senior Personnel	ANL, SNL, INL, PNNL	Japan	O, AOO, DBE, BDBE, SA	H/M	No

Table 18. All Gap Closure Estimates (cont.).

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
STC1	1M-10M	AFC	-	10 – 15*	< 5	-	-	Cadarache Research Center	BDBE, SA	H/L	No
STC2	1M-10M	-	-	10 – 15*	< 5	-	SNL (VICTORIA code)	JAEA (Contain-LMR)	SA	H/L	No
STC3	10M-100M	AFC	-	10 – 15*	< 5	Irradiated Fuel	Hot Cell Facilities at INL and ORNL	TransUranium Institute	BDBE, SA	H/M	No
STC4	10M-100M	AFC	-	10 – 15*	< 5	Irradiated Fuel, Upgraded Domestic Facilities	Upgraded Out-of-Pile Test Facilities at INL and ORNL	CABRI Reactor	BDBE, SA	H/L	No
STC5	1M-10M	AFC	-	10 – 15*	< 5	Irradiated Fuel	SNL (VICTORIA code)	Paul Scerrer Institut, Cadarache (Modeling)	SA	H/L	No
STC6	1M-10M	-	-	10 – 15*	< 5	-	SNL (Contain-LMR/MELCOR)	JAEA (Contain-LMR)	DBE, BDBA, SA	H/M	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

5 POTENTIAL RESOLUTION OF GAPS

5.1 Overview of Programs, Equipment, Facilities

The DOE lab complex has many facilities that can be utilized to close identified SFR licensing gaps. The following sections summarize facilities and programs that are either currently closing gaps or could be re-tasked to close gaps.

5.1.1 Facilities

Table 19 lists the DOE facilities that could be adapted for SFR research. The facilities were divided into four categories: currently available, available within 5 years, within 10 years, or would take more than 10 years. This list was compiled through a combination of previous DOE surveys and though expert elicitation (DOE/NE, 2006; INL, 2009, 2010; NEA, 2011).

Table 19. List of Current and Potential SFR Related Testing Facilities.

Location	Currently Available	< 5 years	< 10 years	10+ years
Idaho National Laboratory	Advanced Test Reactor (ATR) • Target irradiation Electron Microscopy Laboratory (EML) • Equipment used in sample preparation and evaluation Fuels and Applied Science Building (FASB) • Equipment used in fuel development and fabrication Fuel Manufacturing Facility (FMF) • Equipment used in fuel development/manufacturing Hot Fuel Examination Facility (HFEF) • Equipment used in fuel examination and testing	Transient REActor Test (TREAT) • Restart of thermal-spectrum transient fuel testing reactor which was last operated in 1994	N/A	Fast Irradiation Facility • New EBR-II/FFTF like fast reactor fuel irradiation facility.
Argonne National Laboratory	Chemical Engineering Building • materials processing and testing equipment NTSF • Decay Heat Removal Test Loop SNAKE • Sodium/CO_2 reaction experiment B308 AMPB • Sodium reaction experiments	N/A	N/A	N/A

Table 19. List of Current and Potential SFR Related Testing Facilities (cont.).

Location	Currently Available	< 5 years	< 10 years	10+ years
Oak Ridge National Laboratory	Advanced I&C Lab • Development and testing of reactor instrumentation and control systems High Flux Isotope Reactor (HFIR) • Target/material irradiation IFEL • Equipment for the examination of irradiated fuels Irradiated Materials Examination and Testing (IMET) • Equipment for the examination and testing of irradiated materials Sol-Gel Laboratory • Uranium particle fuel fabrication Bldg. 4515 • High temperature materials testing equipment Bldg. 3525 • Equipment used in the examination and testing of irradiated fuels	N/A	N/A	N/A
Pacific Northwest National Laboratory	Radiochemical Processing Laboratory (RPL/PSF) • Radiochemical processing and analytical equipment	105-DR Large Sodium Fire Facility • Closed/Possible to revive Sodium Storage Facility and Sodium Reaction Facility • Closed/Possible to revive	N/A	N/A
Sandia National Laboratories	Sandia Brayton Loop • Material testing for use in S-CO_2 power conversion cycles Surtsey • Sodium spray fire vessel and outdoor pool fire facility	Annular Core Research Reactor (ACRR) • Integral sodium test loop for both pre- and post-failure fuel transient testing Sandia Brayton/Sodium Component Test Loop • Material testing for use in S-CO_2/Sodium coupled power conversion cycles	N/A	N/A
NASA	Marshall Space Flight Center • 3 NaK test loops which use EM pumps, heat exchangers, and flow-meters, etc.			

Table 19. List of Current and Potential SFR Related Testing Facilities (cont.).

Location	Currently Available	< 5 years	< 10 years	10+ years
Los Alamos National Laboratory	LANSCE • Materials Test Station	N/A	N/A	N/A
Massachusetts Institute of Technology	N/A	N/A	MITR • Irradiations in a sealed sodium loop	N/A
International Facilities	**JAPAN** Joyo • Fast irradiation facility, offline Monju • Fast irradiation facility, offline • Sodium Training Facility PLANt Dynamics Test Loop (PLANDT) • Sodium reactor core thermal hydraulics test • transient thermal hydraulics in primary heat removal system • thermal striping MELT • Fuel-Coolant Interactions Core Component Thermal / Hydraulics Test Loop (CCTL) Safety Phenomenology Tests on Sodium Leak Fires and Aerosol (SAPHIRE) **FRANCE** Phenix • Fast irradiation facility, currently offline DISCO2 • Sodium-CO_2 tests Cadarache • Inert concrete-sodium testing and in-sodium materials tests **RUSSIA** BOR60 • Oxide-fueled power and test reactor, operational **KAZAKHSTAN** Impulse Graphite Reactor (IGR) • Operational transient facility similar to TREAT	**JAPAN** ATENA • JAEA Severe Accident Facility MELT-II • Out of core simulant testing for post-core melt relocation **FRANCE** CABRI • Source term studies Plinus/Krotos • Corium-coolant reaction • Currently used for LWRs, needs conversion for SFR studies **CHINA** CEFR • Oxide-fueled EBR-II type facility which is currently undergoing startup. CEFR reached 40% power but was shutdown to conserve fuel. **RUSSIA** BN800 • MOX fueled SFR currently under construction. **INDIA** Prototype Fast Breeder Reactor (PFBR) **BELGIAN** Multi-purpose hYbrid Research Reactor for High-tech Application (MYRRHA) • accelerator driven system / critical facility	**FRANCE** Cadarache • Sodium test loops **CHINA** CDFRs • Two SFRs on order from Russia (BN-800s) in 2009 with construction projected to start in 2013.	**FRANCE** ASTRID • SFR concept **KOREA** KALIMER • IFR type reactor

5.1.2 Performing Transient Fuel Testing

To accommodate any future testing of SFR fuel, two domestic facilities could be made available within 5 to 10 years funding allotment, TREAT at INL and ACRR at SNL. Establishing a cost effective means to conclude post-irradiation transient testing will require addressing any remaining needs against the strengths of the available facilities. The decision to utilize one or both facilities will be dictated by the specific fuel testing demands and particular performance capabilities of each facility.

As presented in the gap section, delineations can be readily established for burnup levels of <10 at%, <20 at%, and >20 at%. Presently, characterization of low burnup (<10 at%) fuel may be sufficient for regulatory purposes given the pre-existing SFR fuels databases. Addressing any later identified gaps for this regime could have a limited scope given the extensive database, limiting the testing requirements. The ACRR facility at SNL can accommodate a range of testing sizes including: small-sized samples, single full-length fuel pins, and/or potentially bundles containing up to 7 FFTF-sized fuel pins. These fuel tests might allow for a lower-cost option (relative to TREAT) to address fuel testing demands where limited or no fuel motion monitoring is needed. To address the needs of higher burnup fuels or new fuels, a significantly larger and robust test program could make restarting TREAT more attractive and/or necessary. Additionally, unlike TREAT or any other transient facility, ACRR's neutron spectrum can be tailored similar to that of a SFR. A highly moderated spectrum will result in an incorrect spatial shielding in the fuel, which would cause an incorrect heating profile in the pin. An ongoing LDRD program at Sandia is focused on developing techniques for instrumentation including optical monitoring.

Currently, the ACRR is resistant to any significant modifications to either the reactor or the test cavity. Thus, an integrated test loop which can fit inside the ACRR cavity (9 inch diameter and greater than 20ft in height) is the only viable testing configuration currently being considered. In order to be brought online, TREAT would need to undergo upgrades, which is an uncertain process regarding cost, to return to operating status. An estimated expense for restarting and upgrading TREAT would be on the order of $100 million (DOE/NE, 2008). Another large differentiator is the overhead for each facility. ACRR would be a shared program with weapons, so overhead would be a shared cost. Because TREAT would be strictly used for transient testing, it would likely have approximately twice the overhead costs to the fuels program.

In addition to existing DOE facilities, if a private initiative were undertaken to design and build a SFR facility at a DOE site, the potential for a government/private partnership exists to make the facility available for irradiation testing of advanced fuels. Such a partnership could take the form of the government taking the lead in fuel design and fabrication while the private party would take the lead for plant design and licensing. In parallel with a TREAT restart program would be an activity to re-establish the software system for reduction, analysis, and display of data from the existing test-fuel-motion monitoring system at TREAT (the fast neutron hodoscope).

5.1.3 Programs Already Closing Gaps

This section summarizes DOE programs currently working to close gaps identified by the gap reports.

5.1.3.1 Advanced Fuel Campaign (AFC)

Fission Product Carryover Fuel Characterization (FM02)

With reprocessed fuel, particularly from the pyro-processing of metal fuel, a significant concentration of the fission product lanthanide elements are likely to remain in the reprocessed fuel. Should these lanthanide elements diffuse to the fuel-cladding interface during irradiation, then the possibility exists for a low-melting alloy to form with consequent premature cladding breach.

Metal fuel that had not been reprocessed reached burnups of 20 at% without failure even though a significant concentration of lanthanide fission products formed during irradiation. However, recent ATR experiments on metal fuel that was pre-doped with lanthanide elements experienced cladding breach at relatively low burnup. Although all the examinations have not been completed, it appears that clumps of the lanthanide elements may have been present from the beginning of the irradiations. Thus, the lanthanide elements may not have been uniformly distributed upon fabrication of the fuel slugs. The question then arises as to whether the reprocessed fuel feed material, which comes from the electro refiner, has the carryover lanthanide uniformly distributed or if the lanthanides exist in clumps and will cause early cladding breach.

This issue is being actively studied in the AFC led by the INL and includes all national laboratories and several universities. This gap is being adequately addressed by the AFC and should be resolved within two years provided the AFC continues to receive adequate funding. Should this gap be resolved such that lanthanide carryover does not affect performance, then metal fuel with fission product carryover can be licensed to 10 at% burnup.

Minor Actinide Carryover Fuel Characterization (FM03)

The minor actinides americium, neptunium, and curium are responsible for the majority of the heat load and toxicity of a spent fuel repository after about 200 years. Thus, it would be desirable to remove these elements from spent LWR fuel and fission them as fuel in a fast reactor.

Studies are under way within the AFC to address the possibility of incorporating substantial quantities of minor actinides in either oxide or metal fuels. Fabrication studies are under way at INL and Los Alamos National Laboratory (LANL) and irradiations of fuel capsules are under way in the ATR. Metal fuels appear to be unaffected with inclusion of minor actinides from a fabrication standpoint. However, open questions remain on how inclusions of minor actinides will affect the pellet sintering process for oxide fuel. Americium is troublesome for both oxide and metal fuel fabrication because of its high vapor pressure and subsequent volatility losses during fabrication.

The ongoing program within the AFC appears to be adequately addressing this issue and with continued funding should solve this issue within four years. Should this gap be resolved such that MA additions do not affect performance, then metal and oxide fuels can be licensed to 10 at% burnup as discussed in FM1.

Advanced Cladding and Duct Fabrication (FM04)

Before the shutdown of EBR-II and FFTF in the 1990s, a vigorous program was in place to fabricate and utilize the low swelling ferritic/martensitic (FM) alloys for cladding and hexagonal ducts. The Car-Tech company had developed reactor grade processes for the large-scale manufacture of HT-9. Other FM alloys were under development on a laboratory scale. The search was ongoing to find FM alloys with better high-temperature strength than HT-9.

With the shutdown of the nation's fast reactor test facilities and the diminished interest on the part of DOE to pursue SFR technology, industry lost interest in the large-scale capability to fabricate FM components. Because two decades have passed, many of the personnel who developed the specifications and processes to fabricate FM components have left the workforce and in parallel the documentation has fallen into a state of uncertain retrieval.

Within the AFC, directed by the INL, an attempt to obtain a small sample of HT-9 is ongoing. However, this only amounts to the initial steps that were taken in the 1980s and 1990s to organize a viable industry to fabricate cladding and duct components.

There will be little industry interest in the renewed development of this fabrication capability until DOE becomes committed to the licensing and construction of a prototype SFR. In the meantime every effort should be made to preserve the fabrication knowledge of HT-9. As well, the effort ongoing within the AFC program to develop advanced FM alloys should continue to receive adequate funding.

Advanced Cladding and Duct Material Properties (FM05)

The limiting factor for the achievement of high burnup of either oxide or metal fuel in a SFR is the irradiation-enhanced swelling and creep of the hexagonal ducts that contain the fuel pins. Excessive duct distortion will result in high pull forces during fuel handling and the possibility of unintended reactivity effects from duct bow during reactor operation.

Austenitic steels such as 316 or a titanium-modified version of 316 were used as cladding and duct material but these alloys exhibit high swelling and creep rates above 100 dpa or 2×10^{23} n/cm^2. About 20 years ago it was discovered that FM steels swell much less. Thus an FM steel, HT-9, was chosen as the likely successor to austenitic steels for cladding and duct applications. However, HT-9 had two possible shortcomings. The first is that the high-temperature strength of HT-9 is inferior to austenitic steels, where 600 °C is the limit for cladding applications. At low temperatures, below 400 °C, HT-9 exhibits an increase in the ductile to brittle transition (DBT) upon irradiation. Should the DBT increase to fuel handling temperatures an issue may exist in the fracture of ducts during fuel handling.

Work is ongoing within the AFC to study the properties of HT-9 and in parallel develop higher-strength FM alloys. However, there is a limitation of what can be accomplished. HT-9 cladding and duct material achieved about 150 dpa before FFTF was shut down. Duct material and tubes from the MOTA test in FFTF are being subjected to a variety of property tests within the AFC campaign. Plans have been developed to achieve additional exposure of this material in foreign reactors. The advanced FM alloys are being subjected to ion bombardment as an unproven means to simulate reactor neutron irradiation.

As the situation now stands, the AFC program is accomplishing the best possible without the availability of a SFR test reactor. Funding of the AFC program for property studies of advanced cladding and duct materials should be continued.

5.1.3.2 Advanced Reactor Concepts (ARC)

Structural Material Issues, Rotating Plug, IHX, EM Pump (FM07)

The selection of structural materials for the initial commercial reactors will be largely based on the past performance, reliability, availability, and cost of the materials. The design of these reactors will be based on the broad technology base that has been developed in SFR programs in the United States and worldwide and the operating experience of several fast reactors around the world, in particular the EBR-II and FFTF in the United States. The large database and lessons learned from operating SFRs can be applied in the design, materials selection, and construction of components for the initial reactors. Furthermore, information relative to the thermal-hydraulic performance of sodium-heated steam generators, the reliability and performance of large sodium pumps, piping, flow meters, and valves, and requirements for sodium-purity control and monitoring equipment is readily available and can be directly applied in the design of these reactors.

The major systems in such a reactor include the reactor vessel containing the reactor core and the primary heat transport system; the intermediate heat transport system; and a power conversion unit that includes sodium-to-steam heat exchanger with a Rankine cycle steam turbine or, alternately, a sodium-to-CO_2 heat exchanger with a Brayton cycle gas turbine. The materials selected for the design of commercial reactors need to comply with the elevated temperature structural integrity criteria (ASME Code Section III Subsection NH). It should be noted that, currently, the U.S. NRC has not approved the Subsection NH and a version of Subsection NH will need to be adopted by the U.S. NRC before a SFR can be commercially licensed in the United States. At present, only five alloys are approved for nuclear application in Subsection NH, i.e., 304SS, 316SS, 2.25Cr-1Mo, Alloy800H, and Mod. 9Cr-1Mo (grade 91), and most probably these alloys will be selected for application in the initial sodium-cooled reactors. The U.S. NRC pre-licensing reviews of CRBR and the PRISM identified several regulatory questions concerning compliance with the elevated temperature structural integrity criteria (ASME Code Section III Subsection NH).

The major gaps identified for the current materials are as follows:

- Need for materials property allowable data/curves for 60-year design life. Additional modeling and experimental effort needed.
- A validated weldment design methodology.
- Reliable creep-fatigue design rules, including hold time creep-fatigue data.
- Improved mechanistically based creep-fatigue life predictive tools for reliable extrapolation of short-term data to 60-year life.
- Methodology for analyzing Type IV cracking in 9-12Cr FM steel welds.
- A validated thermal striping materials and design methodology.
- Materials degradation under irradiation for reactor internal components.
- Materials degradation under thermal aging.
- Materials degradation in sodium environment.
- Degradation under sodium-water reaction.

Development of advanced materials is needed with emphasis on higher temperature capability, component reliability, and flexibility, and to lower the capital and operational cost of the reactor's materials/components. Such a development will be long-term and will be needed for future cost-effective reactors.

SFR EBR-II Test Assembly Knowledge Management and Preservation (FM09)

As part of DOE-NE's ongoing ARC SFR Safety and Licensing R&D program, a modest effort has been initiated at ANL for development of online databases to cover the landmark EBR-II passive safety demonstration tests, TREAT tests, and SFR metal-fuel irradiation experiments. The implementation of all three databases is based on all open-source software, using mySQL relational database management system, Apache web-server, and Perl-based cgi-interface to facilitate controlled access.

The EBR-II test database covers all of the experiments conducted between 1984 and 1987 during a comprehensive testing program including the data collected from the Shutdown Heat Removal Tests (SHRT), Balance of Plant (BOP) Tests, and Plant Inherent Control Tests (PICT). The tests are organized based on the five testing windows during which they were performed (each with unique core, plant, and data acquisition system configuration). During the shutdown heat removal and BOP testing program, 106 files of measured data were collected in four testing windows from June 1984 through April 1986. This collection of data files also included the Integral Fast Reactor (IFR) concept demonstration tests intended to study the inherent safety features of EBR-II as a liquid-sodium cooled fast reactor design with metallic fuel. In November 1987, a final sequence of PICT was also performed in a fifth testing window to demonstrate "load-following" features of the reactor, resulting in eight additional files of measured data. To indicate their nature, the tests are also arranged in several groups/categories, such as the protected (with scram) or unprotected (without scram) loss of flow, loss of heat sink, or various reactivity perturbation tests. Up to 900 instruments recorded during each test are grouped into 60 broad categories. The database provides a test-specific list of recorded data acquisition system instruments and allows the users to select a subset of instruments to plot (and, if desired, to tabulate) the corresponding data. The database also includes a document archive related to the conducted tests under the DOE's IFR Program. The plans are under way to extend this database to include the passive safety demonstration tests conducted at the FFTF reactor through collaborations with PNNL.

A parallel database development effort under the ARC program focuses on information regarding the numerous experiments that were conducted in the TREAT Facility during its operation from 1960 to 1994. Basic experiment-information structure of the TREAT test database is also relational, providing cross-coupling among information sets consisting of

- Metadata for documents regarding individual experiments,
- Metadata for documents regarding groups of experiments,
- PDF files of those documents,
- Categorized parametric information regarding the type and content of those documents,
- Categorized parametric information describing individual experiments, and
- Numerical data from many experiments.

The web-interface specific to the TREAT test database provides layers of search/query forms to the user, retrieves ASCII data or PDF documents, and displays the search results on dynamically written web pages that include links to relevant documents as well as available numerical data (listings and/or plots). Users are presented with general categories of experiment-related information along with options within those categories to help them develop queries that will quickly narrow the searches to the relevant experiments, the types of information being sought, the types of documents that contain the information, and so on. Four main experiment categories are presented:

- Experiment principal objectives and/or outcomes (with twelve sub-categories),
- Test sample characteristics (with six subcategories),
- Test conditions (with four subcategories), and
- Diagnostics and analyses (with three subcategories).

Many of the subcategories are further subdivided, and two auxiliary categories (experimenter organizations and post-test examination locations) are also included. The experiment documents are also characterized in terms of the types of information they contain to help users locate information according to information categories that span multiple tests.

Finally, the SFR Fuels Database is intended to cover the irradiation experiments at EBR-II and possibly at FFTF reactors. The irradiation experiments conducted at EBR-II provided a wealth of information in relation to the development of both metallic and oxide fuels and their performance in fast reactors. The associated archive of information is available as a set of separate documents and computer data files. However, it remains incomplete and does not stand at a development stage that allows for its systematic use by both reactor and fuel designers in support of ARC activities and other DOE programs such as Fuel Cycle Technology (FCT) Fuels Campaign and Nuclear Energy Advanced Modeling and Simulation (NEAMS). The current database development effort under ARC provides a framework for an advanced metallic-alloy fuels database that includes the available information related to the fuels irradiation experiments performed at EBR-II. This database also has a web-based interface that communicates with an open source relational database structure that links (1) available documentation generated through the EBR-II experimental program and documentation used in the calibration and validation of physics and fuel performance codes (e.g., REBUS, LIFE series of codes, etc.), and (2) detailed core loading data, fuels fabrication information, Post Irradiation Experiments (PIE) results, and operating parameters. It is intended to cover detailed pin-by-pin fuel irradiation history information generated using an ANL suite of codes and provide improvement of the quality of information provided to end users (e.g., generation of improved digital images of original PIE results of fuel samples and fuel pins, which provide essential fuel behavior information).

Despite the ambitious scope of these three databases, the amount of funding these efforts receive from DOE-NE under that ARC program is not sufficient to complete the work in a timely manner. Considering the time-sensitive nature of all knowledge preservation and maintenance efforts with critical staff with knowledge of these systems and tests fast approaching retirement age, at a minimum a continued support for these activities, and ideally increased funding, will be important.

5.2 Emphasis on Knowledge Management

A common gap identified from all panels was the need to preserve the sodium-cooled fast reactor database that took over five decades to accumulate at great expense. It has been over 17 years since the last SFR was operated in the United States. If the government or a private institution desired to design and construct a SFR at this point, it would be difficult to ascertain what information is required and could be retrieved from the existing database to aid in the design and licensing. This issue has been of concern for at least the past ten years. There has been general realization that the reduction of SFR research was so precipitous in the mid-1990s that little effort was taken to preserve what had been accomplished. Now many of the facilities where the work was accomplished have been shut down (EBR-II, FFTF, TREAT) and research groups have been dispersed. Most of the researchers who participated in this work are aging and have retired.

There have been islands of effort directed at the problem for the last ten years. Examples of these are:

- At the INL a metal fuels database has been developed and another effort has resulted in much of the EBR-II data being digitized and entered into a computer database. At PNNL the information generated from FFTF is being retrieved.
- At ANL the transient testing database is being retrieved. Also, an effort has been initiated to save irradiation experiment data and documents as well as establishing a detailed metallic fuel information database.
- At the INL, documents related to EBR-II design, operation and testing were collected, stored and those considered most important were digitized.
- The national repository for SFR documentation was located at the DOE site in Santa Susana. Documents remaining there have been surveyed and the most important copied and transferred to ANL for storage.

Conversely, in the area of SFR fuel and materials fabrication the situation likely has had little attention, and as well the documentation for codes and the status of data for code validation are uncertain.

The current situation is one with a variety of attempts at database consolidation, likely leading to a variety of formats. Additionally, there are probably some important areas that have received no attention. The location and storage condition of original data in the form of reports is uncertain and in some cases parlous, with some information already destroyed. Furthermore, the availability of individuals who remember what was done and the value of it is diminishing. A problem of this magnitude must be approached on many fronts.

5.2.1 Knowledge Preservation Gap Teams

It is proposed that a dedicated team of senior researchers undertake a “gap analysis” of the state of knowledge preservation for SFRs. It is recommended that they accomplish the following tasks.

1. Identify the locations and individuals where knowledge preservation activities have been completed or are in progress.
2. Visit those locations and interview the individuals involved.

3. Determine the quality of the effort and learn what further needs to be accomplished from these individuals. Also, learn the storage condition of original data.
4. Produce a report that
   a. Reviews the work that has been accomplished,
   b. Recommends how the existing work can be improved and integrated,
   c. Recommends what additional areas need to be covered, and
   d. Recommends the level of effort and urgency for the work to be accomplished.

5.2.2 Document Preservation Through OSTI

Many organizations, such as the National Aeronautics and Space Administration, have used OSTI to scan, store, and manage access for important documentation that may be needed in the future. While some individual SFR-affiliated DOE programs may decide to procure OSTI's services for document preservation and management, a lightly funded piecemeal approach may cause OSTI to de-prioritize SFR-related work in favor of larger customers. A larger DOE-NE directed effort to utilize OSTI for document preservation and management may streamline the process and ensure that the important documents can be cataloged before key personnel are no longer available to the DOE.

In addition, the INL has developed a data-base for collection of digitized documents related to reactor development programs conducted in Idaho, including the SFR. These documents, those residing at OSTI and other international databases, may be accessed through use of the search engine needle which is available from the DOE-NE website. Access to this information is significantly constrained, however, since much of it is designated as Applied Technology.

5.2.3 Process for Appropriately Applying Applied Technology Stamps on Old Documents

During the reviews associated with the safety and licensing gap analysis, several discussions revolved around the fact that most of detailed SFR technical information retains an "Applied Technology" marking.

A very large volume of information exists within the United States on SFR technology. Much of that information resides in the bibliographic databases maintained by OSTI of the DOE. A quick search by OSTI indicates their database contains more than 5,000 conference reports and more than 15,000 technical reports on SFRs. In general, the conference reports are open-literature publications having no access restrictions except copyright-related information. However, most conference reports are overview or summary type documents and do not provide detailed information needed to support technical knowledge management activities. Most of the technical reports as well as those reports dealing with safety or design of CRBR, PRISM, SAFR, or ALMRs and supporting technical information fall into a category of DOE unclassified sensitive information known as Applied Technology (AT). AT is a category of information established by the DOE-NE and its predecessors to preserve the foreign trade value of certain NE-funded reports containing engineering, development, design, construction, or operation information pertaining to particular programs, including SFR technology. AT has access restrictions and must be protected according to requirements defined by DOE-NE and implemented by OSTI, as the distributor of AT for DOE-NE.

Guidance for handling of AT information was issued by DOE-NE on April 13, 2006. It contains the following points:

- Access to AT documents and software is controlled and is granted by approval of DOE-NE on a need-to-know basis to keep the information contained there in domestic hands.
- AT information cannot be referenced in journals or presentations or referenced in non-AT documents without prior approval of the DOE-NE office. This restriction includes not referencing information in the document and not referencing the title or report number.
- OSTI is the sole distributor of AT documents except when the originating organization is instructed by DOE-NE to make outside distribution directly to a DOE-NE–approved list. Requests for AT-marked information are to be sent to OSTI, which follows an approved process to respond to such requests.
- If DOE-NE directs OSTI in writing to remove the AT designation from individual or classes of documents, or if the activity generating the AT information has been closed for more than 10 years and no action has been taken to extend the AT protection in writing, or if an AT document is more than 25 years old (i.e., published in 1986 or earlier), the AT designation is removed when the document is requested. The document must then be reviewed for Export Controlled Information (ECI) before release. If the document is determined to contain ECI, the access restrictions for this type of unclassified sensitive information must be followed.

The above guidance is based on the 2005 memo on Applied Technology sent by S. Johnson (DOE-NE).

Based on the experience of some of members of gap teams, the process of obtaining AT marked information is tedious and time-consuming, particularly for universities and consulting firms and even within the DOE laboratory system. Each individual report request is sent from OSTI to DOE-NE for approval, which can take weeks to months. OSTI does not remove the markings for reports greater than 25 years old unless a report is requested. In order to address the Export Control Review the report is then sent to the originating organization for ECI review. In the case of SFR technical information, many of these originating organizations no longer exist and those that are in existence may not have the qualified staff available to make the ECI decisions. This then results in further delay.

Although it is important to preserve technical information for possible trade value outside the United States, the immediate need for knowledge management and preservation of the information within the United States suggests a review of the AT guidance process may be warranted to determine if streaming of the current process might be possible. This would allow the U.S. SFR researchers more rapid and timely access to information than is possible under the current guidelines. Additionally, the AT markings are likely to impede the licensing process since the U.S. NRC currently has no process for obtaining, storing, or sharing AT marked information among the staff. It is the authors' recommendation that storage and protection of AT information by individuals within their offices and on their computers should be revisited to determine if it can be made similar to proprietary information handling.

Finally, several years ago the information pertaining to Gas Cooled Reactor Technology had the AT marking removed and can be obtained without restrictions from OSTI other than those associated with ECI. This process may serve as a guideline for the removal of sodium-related AT markings.

6 POTENTIAL ROADMAPS

This section outlines six budget priority scenarios that can be used to determine which gaps get funded and which gaps remain unfilled in the near term. In an attempt to ensure that the results were design neutral, any optional design feature considered was evaluated on the basis that that design feature would be included in the licensing case. It was left to the writers of this report to flag these design optional gaps and for the eventual decision maker to determine the relative importance of funding these research areas. It is the authors' opinion that these gaps, while they ultimately may be required to ensure economic feasibility of the design, should be a lower priority than gaps that will affect any SFR design. Design optional gaps that improve the economic efficiency of a design should be completely ignored, but instead should only be procured through cost-sharing partnerships with interested commercial partners. The six priorities examined are:

1. Less than 1 Million Dollar Gaps
2. Prioritize Time Sensitive Gaps
3. Potential for International Cooperation
4. Gaps Related by Precursors
5. Gaps Significant to Normal Operation of a SFR
6. New Fully Funded SFR Program

6.1 Less than 1 Million Dollar Gaps

Table 20 lists the gaps categories that were estimated at under a million dollars a gap to fill, that is, the smallest price window examined by this report. Of the seven gaps identified in this price range, four gaps related to knowledge preservation and management, two gaps related to code modernization, and one gap related to model development.

Table 20. Gap Topical Areas Under 1 Million Dollars.

Gap ID	Name of Gap Topical Areas	Importance to Safety Within Category	State of Knowledge
CM03	LIFE-Metal/Life 4 Update	H	L
CM05	Modeling of gas bubble entrainment and the effects of sodium–water interaction	M	L
CM08	SAS4A Code Modernization, Support and Knowledgeable User-base	H	L
CM10	Fuel Performance Code Documentation and Training	H	L
FM09	SFR Fuels and Materials Knowledge Base Preservation	H	L
FM10	Fuel Performance Code Documentation and Training Issues	H	L
ST07	Sodium Tech. Knowledge Preservation and Management	H	L

The four knowledge preservation and management gaps were highlighted by the two gap report topics with a high degree of previous experimental effort: Sodium Technology and Fuels and Materials. Domestic experiments related to sodium fires have been conducted at PNNL, INL, ANL, and SNL and internationally in Japan, France, and Russia. These experiments have been used to support the development of various fire progression and containment response codes necessary for licensing, but no money has been available to catalog the current state of sodium technology knowledge. These same statements are true for the two gaps related to fuels and materials, with the exception that no other country has an experimental database for metallic fuel performance. If any information is lost or rendered unrecoverable, the DOE cannot request access to international experimental databases. FM10, relating to LIFE-Metal's need for improved documentation and training, is extremely sensitive because detailed knowledge of this licensing code is one employee deep. There is an extreme need to transfer ownership of this code and the underlying supporting documentation to a mixture of early- and mid-career staff to ensure that the only metallic fuel performance licensing code is not lost. It is vital to fill knowledge preservation and management gaps early for two reasons:

- Some of the historical database may have been lost or destroyed since it was created. If lost or destroyed data are used to support historical correlations and models, these correlations and model codes would no longer be acceptable to support a licensing case.
- If additional sodium technology testing will be conducted in the future, a completed knowledge management and preservation program will be needed to determine if a subset of the historical experiments needs to be reproduced.

In addition to improving LIFE-Metal's documentation and expanding its user base, LIFE-Metal needs support money to keep the code up to date and incorporate the results of recent irradiations conducted at international facilities. The latest calibration of the LIFE-Metal code was performed just before the termination of the IFR project in 1994 (Billone, 1994). Sets of verification test problems that correspond to data from different EBR-II experiments are available and have been used systematically to verify the code calculations.

Minor changes have been made to the LIFE-METAL since its calibration. Those changes did not affect the code's calibration and were mainly aimed at correcting a code error associated with FCCI for fuels with long irradiation periods. Since its last validation activity, the code has been used in a few occasions to support the evaluation of metallic fuel designs associated with advanced fast reactors designs such as the 4S and ARC reactors (Yacout, Tsuboi and Ueda, 2009). Currently, the code has limited number of users and is not released to the national code center as it needs detailed documentation and a revalidation effort to release it. Further, calibration and validation effort of the code can be done once further data from other EBR-II experiments are collected.

SAS4A, the state-of-the-art sodium reactor transient code supported by ANL, also needs additional code support funds. This code will be relied upon to build and validate a licensing accident code, but funding has not been available to keep the code up to date since the IFR program closed. As mentioned previously, this funding would also be used to

- Support updating the memory management scheme to remove various nodalization limits,
- Support parallel applications, and
- Create an input processor and user interface to improve user friendliness and reduce potential input errors.

In order to ensure a vibrant code user-base both internal and external to ANL, funding is needed to support user workshops. These workshops are conducted by other state-of-the-art codes such as MCNP, SCALE, and MELCOR and will help alleviate the learning curve external organizations face when attempting to use SAS4A.

The final gap that was estimated to be resolved for fewer than 1 million dollars was CM05, modeling of gas bubble entrainment and the effects of sodium–water interactions. This gap was the only low-cost gap to rank a medium for importance to safety and thus would likely be a lower priority than the previously mentioned gaps.

The gap closure evaluation for the gaps estimated at less than 1 million dollars can be found in Table 21. Of these gaps, the most time sensitive would be improving the user base for LIFE-Metal and Life 4. In concurrence with resolving this gap, it is recommended that organized knowledge management and preservation effort begin for the areas of Fuels and Materials and Sodium Technology. SAS4A modernization can begin immediately, but updating the LIFE-Metal and Life 4 codes starts only after the fuels and materials knowledge preservation and management program is near completion.

Table 21. Gaps Closure Evaluations Under 1 Million Dollars.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
CM03	100K-1M	AFC	-	5 – 10	5 – 10	Senior personnel, FM09	-	-	O , AOO , DBE , BDBE , SA	M/L	No
CM05	100K-1M	-	-	5 – 10	< 5	-	-	-	DBE	H/L	No
CM08	100K-1M	-	-	< 5	< 5	-	-	-	DBA BDBA SA	H/L	No
CM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM09	100K-1M	AFC	Not for Metal Fuel	< 5	< 5	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	H/M	No
FM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	H/L	No
ST07	100K-1M	-	-	5 – 10*	< 5	Senior personnel	ANL, INL, SNL, PNNL	Japan	O , AOO , DBE , BDBE , SA	H/M	No

6.2 Prioritize Time Sensitive Gaps

Both Table 22 and Table 23 list gap categories for which inaction may result in a loss of capability for the DOE. Table 22 summarizes the five gaps for which inaction over the next 5 years will result in complete loss of capability that will eventually be expensive to replicate.

Table 22. Short Time Sensitivity Gap Topical Areas, Complete Loss of Capability.

Gap ID	Name of Gap Topical Areas	Importance to Safety Within Category	State of Knowledge
CM08	SAS4A Code Modernization, Support and Knowledgeable User-base	H	L
CM09	MELCOR/Contain-LMR Update	H	L
CM10	Fuel Performance Code Documentation and Training	H	L
FM09	SFR Fuels and Materials Knowledge Base Preservation	H	L
FM10	Fuel Performance Code Documentation and Training	H	L

Four of the five gaps identified in Table 22 were described in detail in Section 6.1 listing the gaps that cost less than 1 million dollars to fill. The lone exception is the MELCOR/Contain-LMR Update.

The four overlapping gaps relate to knowledge management and preservation efforts. Because an extremely important component of knowledge management and preservation is interpretation of the historical data, these programs must be started while experienced senior personnel are still available. The most time-sensitive of all these gaps is FM10 and CM10, Fuel Performance Code Documentation and Training. Addressing this gap would effectively require expansion of the LIFE-Metal user base beyond its one current steward.

An organized, constant, and robustly funded preservation and management of these capabilities are vital if the DOE is intent on supporting future fast-reactor development and should be one of the highest priorities. Three areas were identified to be of particular concern:

- Supporting documentation for NUBOW, which predicts core radial expansion;
- Improved documentation and succession training for Life and other fuel performance codes which have only one active user; and
- Categorization and preservation of raw numeric and other data from EBR-II, FFTF, TREAT, and a host of other sodium facilities.

The creation of a SFR version of MELCOR was determined to be a gap of high time sensitivity by the authors. Currently, the United States does not have a supported code for sodium fire and containment response. Not only would such as code be required by the regulator during a licensing process, it would be extremely helpful in guiding research in the areas of Sodium Technology and Source Term Characterization. Alternatively, these gaps could be closed and their results could be implemented into the MELCOR update. The absence of current capacity in SFR containment performance and source term modeling dictates the need for near-term action to expand the capabilities of MELCOR into SFR response.

Table 23 summarizes the gaps for which either work is ongoing (Fuels and Materials Related Gaps) or for which domestic capability may be lost (Sodium Technology). Further work is required in these areas to prevent a partial loss of capability.

Table 23. Short Time Sensitivity Gap Topical Areas, Partial Loss of Capability.

Gap ID	Name of Gap	Importance to Safety Within Category	State of Knowledge
FM02	Fission Product Carryover Fuel Characterization	H	L
FM03	MA Carryover Fuel Characterization	H	L
FM04	Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS	H	M
FM05	Advanced Cladding and Duct Material Properties	H	M
FM06	Duct/Bundle Performance Experience	H	L
FM07	Structural Material Issues, Rotating Plug, IHX, EM Pump	M	L
FM08	Brayton (S/CO_2) Materials Issues	H	L
ST01	Sodium spray dynamics	H	L
ST02	Sodium-fluid interactions (S-CO_2)	H	L
ST03	Sodium-pool fire on an inert substrate	H	L
ST04	Aerosol dynamics	M	M
ST07	Sodium Tech. Knowledge Management	H	L

Information concerning the current efforts to resolve the Fuels and Materials gaps can be found in Section 5.2.2. If funding is cut to these programs before a natural stopping point is reached, additional money would be needed to re-conduct a subset of this work in the future. Of these gaps, Duct/Bundle Performance Experience and Structural Material Issues, Rotating Plug, IHX, and EM Pump should be the highest priority because they are common to almost all pool-type SFR designs.

The short time sensitivities concerning the sodium technology gaps are because of a fear that the sodium fire testing infrastructure will be lost if not used soon. For example, the Surtsey Vessel was recently built using LDRD funds to examine sodium spray fire dynamics in a controlled environment. If SNL cannot determine potential future funding sources for this vessel, it may be demolished in an effort to save maintenance and safety costs associated with the facility. Thus, experiments requiring a Surtsey type facility have a short time sensitivity to ensure that money is not wasted recreating previously available infrastructure. The loss of this vessel will limit the DOE's domestic testing capacity, but international facilities exist that may be able to fill the identified sodium technology gaps. To leverage this information, international agreements either need to be maintained or created. It is the authors' opinion that it would be more cost-effective to maintain current domestic testing capabilities and leverage international agreement to fill gaps for which there is no domestic testing capability, that is, gaps that require fast flux irradiation facilities.

Table 24 lists short time sensitivity gap closure estimates that may result in complete loss of capability; Table 25 lists such estimates that may result in partial loss of capability.

Thus, to avoid a loss of domestic sodium fire testing capability it is recommended that:

- The first sodium fire testing should be started in the next few years to avoid losing infrastructure, and
- The subsequent sodium technology gaps can be organized into successive initiatives, thus ensuring capabilities will not be lost and stretching the limited DOE budget.

It should be noted that ST02, sodium-fluid interactions (S-CO_2), is the only design option in the sodium technology and source term gap analyses. Thus, as stated previously, it is recommended that resolving this gap should be considered a lesser regulatory priority unless an industrial partnership is established to share the cost of resolving the identified safety-related gaps.

Table 24. Short Time Sensitivity Gap Closure Estimates, Complete Loss of Capability.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
CM08	100K-1M	-	-	< 5	< 5	-	-	-	DBA BDBA SA	H/L	No
CM09	1M-10M*	-	-	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM09	100K-1M	AFC	Not for Metal Fuel	< 5	< 5	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	H/M	No
FM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	H/L	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

Table 25. Short Time Sensitivity Gaps, Partial Loss of Capability.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
FM02	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated FP fuel	-	CEFR / BN60	O	H/M	Yes
FM03	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated MA fuel	-	CEFR / BN60	O	H/M	Yes
FM04	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	-	-	-	O	H/L	Yes
FM05	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	H/L	Yes
FM06	1M-10M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Senior personnel	-	-	DBE	H/L	No
FM07	10M-100M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10	Large sodium test loop	ANL	-	DBE	H/M	No
FM08	1M-10M	Solar & Fossil Energy**	Not for Metal Fuel	5 – 10*	5 – 10	Coupled Na/S-CO_2 test loop	-	-	DBE	H/M	Yes
ST01	1M-10M	SMR	JAEA/CEA/ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB / Surtsey	-	SA	H/M	No
ST02	1M-10M	SMR / Solar and Fossil Energy**	JAEA/CEA/ROSATOM	5 – 10*	5 – 10	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	DISCO2 (CEA)	DBE	H/M	Yes
ST03	1M-10M	Possible Solar and Fossil Energy**	JAEA/CEA/ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB/Surtsey	SAPHIRE	SA	H/M	No
ST04	1M-10M	SMR	JAEA/CEA/ROSATOM	5 – 10*	5 – 10*	ST7	B308 AMPB/Surtsey	SAPHIRE (JAEA)	AOO	H/M	No
ST07	100K-1M	-	-	5 – 10*	< 5	Senior personnel	ANL SNL PNNL	Japan	0	H/M	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

6.3 Potential for International Cooperation

Table 26 summarizes the gap topical areas that have the potential for international cooperation. Potential international cooperation is important because it can spread the cost of closing the gap over multiple interested parties, thus reducing the financial burden to the DOE. While the desired use of metallic fuel reduces the potential pathways for international cooperation, there still exists a wide range of SFR related safety gaps that have the potential for collaboration.

Removing the design optional gaps from consideration, that is, AIS04, AIS05, AIS08 and ST02, leaves gaps related to:

- Sodium pool behavior experiments (transition to natural convection and thermal response of structures),
- Modeling capabilities (seismic response, transition to natural convection and subchannel analysis), and
- Sodium technology (sodium sprays dynamics, sodium-pool fire on an inert substrate, aerosol dynamics, and sodium-cavity-liner interactions).

Japan and France both have particularly strong sodium-related experimental capabilities. Most of the international experimental facilities listed in Table 19 are CEA or JAEA facilities. Because the DOE currently lacks the experimental capability to resolve the sodium pool behavior gaps, pursuing international cooperation in these areas is a potential path forward. International cooperation to resolve sodium technology gaps should only be made a high priority if the United States loses its current experimental capability.

International cooperation to close the modeling gaps should be pursued in the medium term but should not prevent more time-sensitive gaps from being funded.

Table 26. Gap Topical Areas with Potential for International Cooperation.

Gap ID	Name of Gap	Importance to Safety Within Category	State of Knowledge
AIS02	Transition to Natural Convective Cooling, Sodium Stratification	H	M
AIS03	Thermal Response of Structures, Thermal Striping	H	M
AIS04	Decay Heat Rejection, Radiation Heat Transfer from Vessels	H	M
AIS05	Power Conversion Cycle, S-CO_2 Accident Analysis	H	L
AIS08	Seismic Isolator	H	M
CM01	Modeling of Seismic Events	H	M
CM02	Models for transient natural convection processes in the reactor system	M	M
CM04	Subchannel and multi-pin analysis capabilities	M	M
ST01	Sodium spray dynamics	H	L
ST02	Sodium-fluid interactions (S-CO_2)	H	L
ST03	Sodium-pool fire on an inert substrate	H	L
ST04	Aerosol dynamics	M	M
ST05	Sodium-cavity-liner interactions	H	M

Table 27 provides closure estimates associated with gaps which may leverage international cooperation.

Table 27. Gap Closure Estimates with Potential for International Cooperation.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
AIS02	1M-10M*	-	Most GIF Members	10 – 15*	< 5	Sodium Component Test Facility	-	PLANDTL, CCTL	DBA, BDBA	H/M	No
AIS03	1M-10M*	-	JAEA	10 – 15*	< 5	Sodium Component Test Facility	MELT Facility once Built	PLANDTL, CCTL	DBA, BDBA	M/M	No
AIS04	1M-10M	-	JAEA / CEA / ROSATOM	10 – 15*	< 5	-	NSTF	ATHENA	DBA, BDBA	H/L	Yes
AIS05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	10 – 15*	< 5	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	DISCO2	DBA, BDBA	M/M	Yes
AIS08	1M-10M	-	JAEA	10 – 15	5 – 10	-	-	-	DBA BDBA	H/L	Yes
CM01	1M-10M	-	JAEA	5 – 10	< 5	-	-	-	DBE, BDBA, SA	H/M	No
CM02	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	DBE, BDBA, SA	H/L	No
CM04	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	O , AOO , DBE , BDBE , SA	H/L	No
ST01	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB / Surtsey	SAPHIRE	SA	H/M	No
ST02	1M-10M	SMR / Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	DISCO2	DBE	H/M	Yes
ST03	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB-/- Surtsey	SAPHIRE	SA	H/M	No
ST04	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10*	ST7	B308 AMPB-/Surtsey	SAPHIRE	AOO	H/M	No
ST05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10	5 – 10*	ST7	SNL, ANL	SAPHIRE	DBE	H/L	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

6.4 Gaps Related by Precursors

Table 28 lists the 31 gap topical areas which had identified precursors that are needed before the gap can be filled. A precursor can either increase or decrease the urgency associated with resolving a gap; that is, needing senior personnel would increase the priority of a gap while needing the construction of an expensive experimental facility may decrease the priority of a gap. Of 31 identified gap topical areas, 9 were considered to be design-specific.

Table 28. Gap Topical Areas With Precursors.

Gap ID	Name of Gap	Importance to Safety Within Category	State of Knowledge
AIS01	Steady State Intact Fuel and Fuel Changes	H	M
AIS02	Transition to Natural Convective Cooling, Sodium Stratification	H	M
AIS03	Thermal Response of Structures, Thermal Striping	H	M
AIS05	Power Conversion Cycle, S-CO_2 Accident Analysis	H	L
AIS06	Fuel Transient Behavior – Length Effects	M	L
AIS07	Severe Core Damage, Metal Fuel Motion, Dispersal and Morphology	H	M
CM03	LIFE-Metal/Life 4 Update	H	L
CM06	Advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content	H	L
CM09	MELCOR/Contain-LMR Update	H	L
CM10	Fuel Performance Code Documentation and Training	H	L
FM01	High Burnup Fuel Characterization	H	M
FM02	Fission Product Carryover Fuel Characterization	H	L
FM03	MA Carryover Fuel Characterization	H	L
FM04	Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS	H	M
FM05	Advanced Cladding and Duct Material Properties	H	M
FM06	Duct/Bundle Performance Experience	H	L
FM07	Structural Material Issues, Rotating Plug, IHX, EM Pump	M	L
FM08	Brayton (S/CO_2) Materials Issues	H	L
FM09	SFR Fuels and Materials Knowledge Base Preservation	H	L
FM10	Fuel Performance Code Documentation and Training Issues	H	L
ST01	Sodium spray dynamics	H	L
ST02	Sodium-fluid interactions (S-CO_2)	H	L
ST03	Sodium-pool fire on an inert substrate	H	L
ST04	Aerosol dynamics	M	M
ST05	Sodium-cavity-liner interactions	H	M
ST06	Sodium-concrete-melt interactions	H	M
ST07	Sodium Tech. Knowledge Management	H	L
STC03	Radionuclide transport within a sodium pool	H	M
STC04	Radionuclide Chemistry in Sodium Bond between Fuel and Cladding	H	M
STC05	Mechanical release of radionuclides from the surface of a sodium pool	H	M

Six of the remaining 21 gap topical areas require senior personnel. These six categories can loosely be organized into four groups:

- LIFE-Metal – Two of these gaps are related to the previously discussed need to improve the current situation surrounding LIFE-Metal. Code documentation and training of new stewards need to be completed before the current user retires. Improvements and updates to LIFE-Metal should at least be started when the current steward is on staff and should be completed before the current steward is unavailable for consulting.
- MELCOR – Improvements to MELCOR to incorporate sodium performance subroutines from Contain-LMR should be completed while members of the original MELCOR development team are still available to help update the code. While new staff can potentially expand the MELCOR code after the original developers stop contracting with SNL, the cost of the upgrade will most likely increase.
- Duct/Bundle Interactions – The achievable burnup of an individual fuel pin may not be realized if the performance of the fuel assembly is limiting. Bundle interactions between fuel pins may restrict coolant flow, bow and swelling of the hexagonal duct may result in excessive fuel handling forces, and the motion of the assemblies from bow may result in unintended reactivity effects. The performance characteristics of the assembly directly depend on the material properties of the cladding and ducts, the temperature and neutron flux gradients that are a result of the reactor design, and the core restraint system. The effort should continue to accumulate all the available observations, measurements, and analyses while personnel are still available who know what was done and where and how record of it is stored.
- General Knowledge Management and Preservation – This group applies to the Fuels and Materials and Sodium Technology gap topical areas. Senior personnel will be needed to direct the knowledge management and preservation effort and to interpret any data that may not be properly annotated.

The remaining gaps identified in the Accident Initiators and Sequences, Codes and Methods, and Source Term Characterization reports either require new domestic facilities or the use of international facilities. For generic sodium reactor related gaps, joint international initiatives may allow for the closure of these gaps without additional U.S. facilities. The remaining gaps require new irradiated metallic fuel. While ATR and HFIR may be able to partially fill in these gaps, fast neutron irradiated fuel at power reactor, temperatures and neutron fluxes will be needed to satisfy the regulator. Short of building a new U.S. irradiation facility, agreements will be need to be made with Chinese or Russian reactors to irradiate U.S. metallic fuel for testing. It is likely that by the time CM06 (for example, advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content) is maturing, the United States will be at a point where a domestic fast neutron irradiation facility is available.

The structural materials issues regarding the Rotating Plug, IHX, and EM Pumps will need a large sodium test facility to determine their long-term reliability in a high-temperature sodium environment. This facility is currently being constructed at ANL and a similar facility is being proposed by SNL that will couple to their S-CO_2 loop. It is likely that both of these facilities will be needed if the open structural materials issues are to be closed in a reasonable time frame. Finally, most of the sodium technology related gaps require the knowledge preservation and management program to be concluded before these gaps can be put to rest. This is caused by the high degree of uncertainty regarding the availability of the underlying data supporting the Contain-LMR code.

Table 29 lists the gap closure estimates with precursors.

Table 29. Gap Closure Estimates with Precursors.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
AIS01	10M-100M*	AFC	-	10 – 15*	5 – 10	Fuel Testing Facility	ATR and/or HFIR	JOYO	DBA, BDBA	H/M	No
AIS02	1M-10M*	-	Most GIF Members	10 – 15*	< 5	Sodium Component Test Facility	-	PLANDTL, CCTL	DBA, BDBA	H/M	No
AIS03	1M-10M*	-	JAEA	10 – 15*	< 5	Sodium Component Test Facility	MELT Facility once Built	PLANDTL, CCTL	DBA, BDBA	M/M	No
AIS05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	10 – 15*	< 5	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	-	DBA, BDBA	M/M	Yes
AIS06	10M-100M*	AFC	-	10 – 15	< 5	Irradiated Fuel and TREAT or ACRR	TREAT or ACRR	-	DBA, BDBA	M/L	No
AIS07	10M-100M*	AFC	-	10 – 15	< 5	Irradiated Fuel and TREAT or ACRR	TREAT or ACRR, Possibly CAMEL	MELT-II	SA	H/L	No
CM03	100K-1M	AFC	-	5 – 10*	5 – 10	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	M/L	No
CM06	1M-10M	-	-	15 +	< 5	Irradiated Fuel and TREAT or ACRR	-	-	BDBE, SA	H/L	No
CM09	1M-10M*	-	-	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM01	+100M*	AFC	Not for Metal Fuel	15 +	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	M/M	Yes
FM02	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated FP fuel	-	CEFR / BN60	O	H/M	Yes
FM03	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated MA fuel	-	CEFR / BN60	O	H/M	Yes
FM05	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	H/L	Yes
FM06	1M-10M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM07	10M-100M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10	Large sodium test loop	ANL	-	O, AOO, DBE, BDBE, SA	H/M	No
FM08	1M-10M	Solar & Fossil Energy**	Not for Metal Fuel	5 – 10*	5 – 10	Coupled Na/S-CO_2 test loop	-	-	DBE	H/M	Yes
FM09	100K-1M	AFC	Not for Metal Fuel	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/M	No
FM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No

Table 29. Gap Closure Estimates with Precursors (cont.).

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
ST01	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB/Surtsey	SAPHIRE	SA	H/M	No
ST02	1M-10M	SMR / Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	Coupled Na/S-CO_2 loop	S-CO_2 loop at SNL but Na test loop is needed/Scaled-up version of SNAKE	DISCO2 (CEA)	DBE	H/M	Yes
ST03	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10*	5 – 10	ST7	B308 AMPB/Surtsey	SAPHIRE	SA	H/M	No
ST04	1M-10M	SMR	JAEA / CEA / ROSATOM	5 – 10*	5 – 10*	ST7	B308 AMPB/Surtsey	SAPHIRE (JAEA)	AOO	H/M	No
ST05	1M-10M	Solar and Fossil Energy**	JAEA / CEA / ROSATOM	5 – 10	5 – 10*	ST7	SNL, ANL	SAPHIRE (JAEA)	DBE	H/L	No
ST06	10M-100M	SMR	JAEA / CEA / ROSATOM	10 – 15*	5 – 10	ST7, Core-Concrete-Sodium Test Facility	MCCI	PLIINIUS (CEA) (SAPHIRE?)	SA	M/L	No
ST07	100K-1M	-	-	5 – 10*	< 5	Senior personnel	ANL SNL PNNL	Japan	O, AOO, DBE, BDBE, SA	H/M	No
STC3	10M-100M	AFC	-	10 – 15*	< 5	Irradiated Fuel	Hot Cell Facilities at INL and ORNL	TransUranium Institute	BDBE, SA	H/M	No
STC4	10M-100M	AFC	-	10 – 15*	< 5	Irradiated Fuel, Upgraded Domestic Facilities	Upgraded Out-of-Pile Test Facilities at INL and ORNL	Cabris Reactor	BDBE, SA	H/L	No
STC5	1M-10M	AFC	-	10 – 15*	< 5	Irradiated Fuel	SNL (VICTORIA code)	Paul Scerrer Institut (Switzerland), Cadarache (Modeling)	SA	H/L	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

** If Solar and Fossil Energy can be convinced to uses sodium as their thermal storage medium for solar concentration and then decided to couple as S-CO_2 power conversion loop

6.5 Gaps Significant to Normal Operation of a SFR

Table 30 lists the 16 gap topical areas that had a direct influence over normal operation of a SFR. It has been suggested that normal operation is the most important mode during which to ensure a large degree of safety, because otherwise the reactor would not be built regardless of the regulator. Of 15 identified gap topical areas, six were considered design optional. DOE focus on resolving regulatory concerns associated with these gaps is only recommended if an industrial partner is identified.

Table 30. Gap Topical Areas Related to Normal Operation.

Gap ID	Name of Gap	Importance to Safety Within Category	State of Knowledge
CM03	LIFE-Metal/Life 4 Update	H	L
CM04	Subchannel and multi-pin analysis capabilities	M	M
CM07	Models to predict source term releases from fuel in SFR accidents	H	L
CM09	MELCOR/Contain-LMR Update	H	L
CM10	Fuel Performance Code Documentation and Training	H	L
FM01	High Burnup Fuel Characterization	H	M
FM02	Fission Product Carryover Fuel Characterization	H	L
FM03	MA Carryover Fuel Characterization	H	L
FM04	Advanced Cladding and Duct Fabrication, HT-9, 9Cr-1Mo, ODS	H	M
FM05	Advanced Cladding and Duct Material Properties	H	M
FM06	Duct/Bundle Performance Experience	H	L
FM07	Structural Material Issues, Rotating Plug, IHX, EM Pump	M	L
FM08	Brayton (S/CO_2) Materials Issues	H	L
FM09	SFR Fuels and Materials Knowledge Base Preservation	H	L
FM10	Fuel Performance Code Documentation and Training Issues	H	L
ST07	Sodium Tech. Knowledge Management	H	L

When the design optional gap topical areas are removed, the remaining gaps coincidentally affect all potential reactor states from normal operation to severe accidents. Many of these gaps have been highlighted in other prioritization schemes and many revolve around knowledge management. This conclusion is logical because of the long history associated with sodium reactors. SFRs have been operated for a sufficient period of time that day-to-day operational issues have been addressed unless

- A new technology or material is employed or
- The old records documenting historical success have been lost.

Table 31 lists the gap closure estimates related to normal operation.

Table 31. Gap Closure Estimates Related to Normal Operation.

Gap ID	Estimated Cost Range	DOE Funding Programs Other than ARC?	International Funding Programs?	Time Sensitivity (years)	Time Required to Fill Gap (years)	Precursors	US Facilities	International Facilities	Event Category (O, AOO, DBE, BDBE, SA)	Importance	Optional Design Feature
CM03	100K-1M	AFC	-	5 – 10*	5 – 10	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	M/L	No
CM04	1M-10M*	-	JAEA / CEA / ROSATOM	5 – 10	< 5	-	-	-	O , AOO , DBE , BDBE , SA	H/L	No
CM07	1M-10M*	-	-	5 – 10	< 5	-	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM09	1M-10M*	-	-	< 5	< 5	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
CM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O, AOO, DBE, BDBE, SA	H/L	No
FM01	+100M*	AFC	Not for Metal Fuel	15 +	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	M/M	Yes
FM02	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated FP fuel	-	CEFR / BN60	O	H/M	Yes
FM03	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility, Irradiated MA fuel	-	CEFR / BN60	O	H/M	Yes
FM04	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	-	-	-	O	H/L	Yes
FM05	1M-10M	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Access to a fast flux irradiation facility	-	CEFR / BN60	O	H/L	Yes
FM06	1M-10M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10*	Senior personnel	-	-	O , AOO , BDE , BDBE , SA	H/L	No
FM07	10M-100M*	AFC	Not for Metal Fuel	5 – 10*	5 – 10	Large sodium test loop	ANL	-	O , AOO , DBE , BDBE , SA	H/M	No
FM09	100K-1M	AFC	Not for Metal Fuel	< 5	< 5	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	H/M	No
FM10	100K-1M	AFC	Not for Metal Fuel	< 5	5 – 10*	Senior personnel	-	-	O , AOO , DBE , BDBE , SA	H/L	No
ST07	100K-1M	-	-	5 – 10*	< 5	Senior personnel	ANL SNL PNNL	Japan	O, AOO, DBE, BDBE, SA	H/M	No

* The authors did not reach a consensus regarding these ranges. In each case, the highest range was placed in the table. In no case did the range vary more than one classification (i.e., Author A chose 5-10 years and Author B chose < 5 years. At no point did Author A chose 10-15 years and Author B chose < 5 years.)

6.6 New Fully Funded SFR Program

As an alternate scenario, it was proposed that the authors consider a scenario where DOE SFR program funding increased dramatically. In this scenario, the new program would be provided with an approximately $400M/year budget and tasked with constructing a fast neutron irradiation facility and closing the remaining design independent safety-related gaps listed in Table 17.

First, expansive knowledge preservation and management programs would need to be funded and implemented in all identified areas to ensure that no safety-related gaps were overlooked and no additional data were lost or destroyed. LIFE-Metal staffing would be increased and recalibration and validation of LIFE-Metal would most likely be a critical-path initiative in order to validate the new irradiation facilities' first core loading before construction would begin. Additionally, the update of MELCOR would need to be finished in a fairly short period of time to ensure that a current containment performance code was available for licensing.

Experimentally, the decay heat rejection tests scheduled for the NSTF facility would be given high priority because these tests are needed to ensure adequate defense-in-depth of the decay heat removal system in the reactor design. International sodium technology and source term testing facilities (SAPHIRE, PLINIUS and MELT-II) would need to be leveraged along with domestic sodium technology testing facilities (Surtsey and B308 AMPB) to ensure that the necessary licensing data are available to support the MELCOR-LMR update.

Even with a large budget, it may take at least 5 to 10 years to increase DOE SFR related staffing to the required levels to design and build a new irradiation facility. During this time, the relative advantages of TREAT and ACRR should be compared to determine which one, or both, facilities would provide the most cost-beneficial capability to satisfy SFR transient fuel testing requirements. That comparison is underway as part of the DOE-led Analysis of Alternatives for Resumption of Transient Testing of Nuclear Fuels and includes HFIR and ATR in addition to ACRR and TREAT. All relevant aspects of the four reactors, including facility modification and modernization, are being evaluated in terms of the capability of the reactor itself as well as the experiment-support facilities that would be available at each site, considering the entire range of physical capabilities needed to conduct the in-reactor transient testing required in the future. This work, which is being performed as a prerequisite to approval to initiate restarting TREAT, may result in the conclusion that one, or a combination of two or more, of the candidate reactors will provide the best means by which to develop the needed expansion of the transient fuels database.

6.7 Observations

The following observations can be drawn from the six funding priorities analyzed in this report:

- Knowledge Preservation and Management is a cross-cutting short-term need in the SFR community. The consequences of delays in filling this gap will be costly to fix and these gaps are currently some of the cheapest to fill.
- Special attention is needed for LIFE-Metal. The status quo of relying on one steward, who is nearing the end of his career, to keep the code available is extremely untenable and needs to be corrected quickly.

- There is currently no maintained SFR containment response modeling ability. The development of an LMR-MELCOR will be important to support any licensing effort. The passive safety of current SFR designs will not prevent the need for a containment code.

Any gap requiring future irradiation of metallic fuel to high burnup is not likely to be filled within the next 10 to 15 years.

7 OBSERVATIONS AND RECOMMENDATIONS

This report was intended to both make research recommendations based on five previously conducted safety-related gap analysis reports. While the eventual adoption path for funding recommendations will be subject to the changing needs and budget, the identification of cross-cutting gaps, that is, a coordinated Knowledge Management and Preservation effort, may be the most important information highlighted by this report.

7.1 Review of the Project

This report began with a history of SFR involvement by the DOE and around the world. Historical licensing efforts with CRBR, SAFR, and PRISM were summarized. While these regulatory efforts were guided by the deterministic licensing mindset of the day, the increasing adoption of probabilistic elements to the licensing framework may change the level of regulatory scrutiny applied to a future licensing effort. This shifting emphasis is reflected in the proposed SFR regulatory standard, ANSI/ANS 54.1. These regulatory shifts must be understood when starting new research and development initiatives to improve the licensability of a SFR.

Four of the five underlying topic areas considered in this gap analysis are the same as those identified to improve the licensability of the NGNP. Because of the specific safety concern related to handling sodium, a fifth topic area relates to sodium phenomenology. These five gap analyses were led by recognized experts in their respective fields and participated by panels of additional experts to identify regulatory hurdles facing the topical area. These panels also attempted to rank the current state of knowledge and regulatory significance of each gap. In an attempt to ensure that the results were design-neutral, any optional design feature considered was evaluated on the basis that that design feature would be included in the licensing case. It was left to the writers of this report to flag these design optional gaps and for the eventual decision maker to determine the relative importance of funding safety and licensing related issues in these research areas. It is the authors' opinion that these gaps, while they ultimately may be required to ensure economic feasibility of the design, should be a lower priority than gaps that will affect all SFR designs. Design optional gaps that improve the economic efficiency of a design should not be completely ignored, but instead should only be procured through cost-sharing partnerships with interested commercial partners.

7.2 Observations from Prioritization Scenarios

As a result of the gap evaluation process discussed above, six budget priority scenarios were developed to help decision-makers determine which gaps get funded in the near term. The following observations were seen to be cross-cutting or high priority issues for each of the five gap reports.

Accident Sequences and Initiators

- Knowledge Preservation and Management – While independent efforts are currently underway throughout the SFR community to secure and analyze historical databases, these efforts are only sporadically coordinated and often underfunded. A DOE-NE led effort to ensure that the historical database is not lost or destroyed will be instrumental in supporting any future licensing effort.

- Due to economic considerations, any gap requiring high-burnup metallic fuel is not likely to be filled in the foreseeable future. Without domestic fast neutron irradiation facilities, any new irradiations would need to be conducted in one of the few remaining international fast reactors. Due to host restrictions, these experiments would most likely be limited to pin scale tests. In order to remove regulatory hurdles associated with the identified high-burnup gaps, full assembly and/or full core experiments will be required.

Sodium Technology

- Some U.S. facilities, such as SNL's Surtsey, are currently under-utilized and are capable of addressing many high-priority gaps. It may be cost effective for the DOE to fund domestic or to host internationally funded experiments in these under-utilized facilities before they are closed due to lack of funding.
- There is currently no maintained capability within the DOE for modeling of sodium fire phenomenology. This gap is related to the containment response gap discussed in Source Term Characterization below.

Fuels and Materials

- Consistent with the comment on Knowledge Preservation and Management in Accident Sequence and Initiators, the user-base and expertise for the fuel performance and licensing code LIFE-Metal computer code has almost disappeared. Currently, one steward is relied upon to keep the code accessible to future users. This situation is untenable; thus, sufficient funding should be provide in the near term to train the next generation of users not only how to operate the code, but how to explain and defend the code to a licensee or regulator.
- Due to economic considerations, any gap requiring future irradiation of metallic fuel to high burnup is not likely to be filled within the next 10 to 15 years.

Source Term Characterization

- There is currently no maintained capability for modeling of the containment response and estimation of the corresponding source term. The extension of capabilities under a severe accident code such as MELCOR for SFRs will be important to support any licensing effort. The passive safety of current SFR designs may not avoid the need for a containment code.
- While source term related research requiring irradiated fuel will likely need to be postponed until SFR funding levels increase, research requiring only radionuclide tracers, e.g., radionuclide release from fuel debris into a quiescent sodium pool and radionuclide behavior in containment, can be conducted using existing facilities at INL and ORNL. Because the accidents in Fukushima will likely require that the SFR source term will be part of an advanced reactor license application, it is logical to begin research in this area in the near to mid-term in order to avoid severe delays if a SFR license is submitted.

Codes and Methods

- See MELCOR comment in Source Term Characterization.

- Due to Fukushima, experimental data concerning the response of SFR core materials and structures, systems, and components to earthquakes and other external events need to be collected and used to improve current computational models.
- The state-of-the-art concerning transitions from full power to natural circulation will need to be improved to defend passive safety as a layer within defense-in-depth.
- SAS4A needs to be modernized and improved (in terms of modeling accuracy, functionality, and usability), and adapted to modern software engineering practices.

Common Issues

- The current process for handling AT documentation was determined to be at best complicated and at times counterproductive. The current process makes removing AT designations on documents which no longer need to be protected extremely difficult. Additionally, the U.S. NRC is not set up to handle AT documents within their current knowledge management system. This makes any AT document unusable to a licensee. DOE-NE needs to develop a new process for streamlining the AT process if a SFR license is submitted.

7.3 External Feedback

Volume I of this report was distributed to a selected group in April 2012 for feedback. Some comments were accepted and incorporated directly into the report. Other comments were not applicable to the scope of this report; therefore they are acknowledged in this section, grouped by sources. Names of the commenters were replaced by generic identifiers to ensure anonymity.

It should be recognized that while the authors may not have adopted all of the commenters' suggestions, understanding of the diverse views across all SFR stakeholders is considered extremely important to both current and future decision-makers who may produce different conclusions than the authors.

7.3.1 Laboratory Feedback

Three comments were received from DOE laboratory staff members. Their comments are paraphrased below.

Feedback # 1

Laboratory commenter # 1 stated that designing and building a test and/or demonstration facility was the most cost effective way to close the gaps identified in this report. It was suggested that such a facility could be built by reaching out to an industrial partner to build their reactor concept on a DOE facility with the DOE providing the fuel for the design. Because fuel costs are relatively high for a SFR, such an agreement may provide enough incentive for commercial investment.

Feedback # 2

Laboratory commenter # 2 disagreed with the level of confidence reflected in the report concerning ACRR's ability to perform the necessary post failure fuel testing, especially in regards to multi pin and full-length pin tests.

Feedback # 3

Laboratory commenter # 3 questioned the assertion that the U.S. NRC is not able to handle Applied Technology documents since they deal with proprietary information on a daily basis. Further, AT was handled the course of the CRBRP, SAFR and PRISM reviews, thus there must have been a mechanism to remove the AT restriction for licensing purposes.

Authors' Comments

Recently, the U.S. NRC and RES attempted to create an U.S. NRC knowledge management program to familiarize U.S. NRC staff with SFR design and technical support information. During this process, the U.S NRC stressed that the knowledge management program could not handle AT documents. This rule limited the information content of the U.S. NRC's library to older documents, summary level information, or technical journal publications in which AT information and references were stripped. AT information can only be referenced or distributed to others after undergoing a lengthy and difficult DOE approval process for each document of interest. The NRC-RES has created a SFR training program but cannot reference AT information in either the course content or handouts. It is unclear how the AT issue was overcome internally within NRC during the licensing of CRBR and PRISM/SAFR.

Proprietary information is handled by the U.S. NRC through the use of a non-disclosure document to ensure that distribution of the information is limited. This document allows signatories the ability to distribute or discuss the relevant proprietary information amongst other signatories within the agency. With appropriate transmittal protection, such documents also allow the U.S. NRC to distribute to third parties, such as their contractors outside NRC, who also sign a non-disclosure agreement. Documents containing proprietary information in announcements or presentations can be referenced, since there is both a public and a proprietary version. In the public version, the proprietary information has been redacted but the rest of the document is available. Thus, referencing of documents containing proprietary information in publically available documents is allowed, unlike the restrictions placed upon AT documents.

In addition to distribution restriction being limited to DOE-NE approval, the other restrictive issue concerning the AT designation is that, if a document references an AT document, the referencing document becomes AT based on the 2005 Johnson letter.

It is proposed that DOE reshape AT into something similar to proprietary information. The designation was created for the same business sensitivity reason.

7.3.2 Industry Feedback

Three comments were received from industrial stakeholders interested in SFRs. Due to the extensive nature of some of these comments, the comments were separated into views that aligned and diverged with the authors' recommendations. Divergent views were then responded to by the authors.

Feedback # 1

Industry commenter # 1 appreciated the opportunity to review the research plan before final publication. No specific comments were provided but the report was deemed to be very thorough.

Feedback # 2

Alignment

Industry commenter # 2 stated that the plan outlined in this report covers the correct topics and was well conceived.

Divergent

Industry commenter # 2 viewed the R&D plan presented in this report as being too optimistic in both timescale and budget. Additionally, the commenter estimated that the impact of Fukushima greatly increase the role of severe accident in the safety case for the SFR.

Authors' Comments

It should be noted that project costs are notoriously uncertain and funding for this report did not allow for detailed gap closure cost estimates. Even with these caveats, the estimates in this report should help gauge relative costs for the identified gaps.

Feedback # 3

Alignment

Industry commenter # 3 agreed with the report in the following areas.

- The need to improve seismic hazard analysis techniques for SFRs. Computational models were stated to be weak, especially as they relate to reactivity and thermal-hydraulic effects of core geometry deformations; both during the initial transient and due to plastic deformation of structures.
- Seismic isolation is a relatively common design tool to reduce seismic risk, but has little regulatory guidance for long term maintenance requirements. They have been used in many non-nuclear applications, and in nuclear power plants in South Africa.
- There is a need for improved spatial kinetics techniques for large cores.
- Understanding material behavior above 150 dpa.

Divergent

Industry commenter # 3 expressed concerns that the report did not mention NaK behavior and did not significantly prioritize high-burnup fuel research. The commenter urged for an improved understanding of NaK (sodium-potassium eutectic) behavior; especially for use in supporting systems such as those used for decay heat removal. NaK behavior is a phenomenon not covered in any of the gap analysis reports. Regarding advanced fuels research, the commenter suggested a greater focus more on metallic fuels with low zirconium (~60 wt% Zr) fuels.

Authors' Comments

The authors currently project that supporting system would utilize sodium, not NaK, to reduce potential contamination issues in the event of a leak. Thus, limited DOE safety related funds should not be directed at NaK behavior unless a cooperative agreement could be reached with an interested industrial partner.

Regarding high burnup fuels research, the authors agreed that advanced fuels research is important but the prioritization outlined in this research plan inherently considered the limited funding available for safety related research. The authors hope that new research facilities can be constructed in the near future to facilitate this research, but the current safety related research budget cannot be expected to finance such facilities without industrial support.

Feedback # 4

Industrial commenter # 4 provided extensive feedback. Some comments were directly addressed in the text of the report, while others are listed below. For clarity, each divergent opinion will be addressed by a corresponding numbered response in the authors comment section.

Alignment

Industry commenter # 4 agreed with a number of the proposed recommendations and insights including the following areas.

- The need for improved modeling of containment performance and seismic isolation.
- The need for robust in-service inspection.
- The limited high-burnup fuel database, especially for accident conditions.
- The need for high-burnup fuel performance models.
- The need for V&V for computer simulations of the transition to natural convection.
- The use of Level 1 and Level 2 PRA in assessing prevention and mitigation measures (not explicitly stated in the report but the author completely agree with this comment).
- The need for models to support Level 2 PRA development (i.e., MELCOR and/or Contain-LMR).
- Improvement and modernization of Contain-LMR (the authors believe that these improvements would be more maintainable within MELCOR code structure, but the authors agree that least one of these codes should be maintained).
- The need for sodium aerosol behavior modes for both in containment and in the environment.

Divergent

Industry commenter # 4 also disagreed with some of our conclusions. They are numbered below.

1. SFR behavior dominating residual risk (e.g., large reactivity excursions, complete failure of decay heat removal, severe accident phenomenology) should be highlighted more in the analysis and recommendations.
2. Severe accident consequences should have been assigned its own gap analysis (e.g., energetic accident consequences, molten-fuel-coolant-interaction, and core catcher performance).
3. Sodium toxicity and chemical risk-PRA development should also have been assigned its own gap analysis.
4. The need for improved neutron and robust post-accident sensors, fuel sub-assembly identification, spent fuel heat loading measurement, assembly level plutonium loading estimation, and remote temperature estimation were neglected by the I&C experts.

5. There is a need for rapid core-unloading in case of fault detection in the core support structures.
6. Knowledge of sodium/water (and other industrial liquids such as those potentially used in fire-fighting) interactions need to be better characterized.
7. Radiological and chemical consequences of contaminated sodium aerosol fires are important areas for future R&D not characterized in this report.
8. Fuel knowledge extends beyond the reactor conditions listed and may include differences between past and present manufacturing techniques.
9. No knowledge gaps existing for manufacturing a rotatable plug, as listed in this report.
10. Simulants may not be adequate for fuel-coolant chemical interactions, thus irradiated fuel should be used. Additionally, the facility at Cadarache will be focused on molten fuel to coolant energy transfer and would likely not be available to research radionuclide leaching.

Authors' Comments

The authors' response corresponding to the enumerated comments listed above can be found below. It should be noted that the authors deeply valued the industrial commenter's insights and even though perfect agreement may not have been achieved, the exchange is valuable for both sides and for future decision-makers. Additionally, some differences of opinion may result from differing international approaches to regulation.

1. The authors believe that while residual-risk is important, many of the accidents listed do not apply to current metallic-fueled SFR designs. Where this is not the case (e.g., the reliability of decay heat removal) existing gap panels evaluated the appropriate modeling and experimental needs.
2. While the comment is understandable, it is believed that most of the relevant severe-accident phenomena relevant to licensing a metallic fueled SFR were captured in the existing gap reports.
3. The authors agree that these issues are important but are unsure if the current state of knowledge is sufficient for licensing. This area did not appear in any of the five expert panels. A study may be warranted once a robust sodium test is established.
4. These topical areas may be important and will likely need to be examined before a safety case can be finalized. This report summarized first-order listing safety related I&C research areas but did not have the funding or scope to conduct a robust expert elicitation in areas outside the five analyses summarized in Section 4 and provided in Volume II of this report.
5. The authors note that this is not a requirement for LWRs. Even though SFR fuel is not loaded in an optimal neutronic configuration, it is not clear that a regulator would require such a capability or what time frame would be considered "rapid."
6. The panel did not address chemical interaction of Na with firefighting materials other than water. Usually Na fires are extinguished using materials designed for metal fires or by reducing the supply of oxygen by injecting gases such as nitrogen, there are no compatibility issues with these fire-fighting processes.

7. The authors believe that if the current state of knowledge for radiological sodium hazard is insufficient, resolving the gaps associated with sodium technology and source term characterization would allow for adequate understanding of this hazard. A discussion of sodium aerosols is found in the sodium technology gap technology report.
8. It is believed that the first series of fuel loadings would be manufactured consistent with historical practices until methods are proven reliable.
9. Current metallic fueled SFR designs have not finalized the design of the rotatable plug. It may turn out that no technology gaps exist for this component, but this will not be determined until the design is finalized. This is a design issue not a safety issue except for leakage of cover gas.
10. It is the authors' belief that simulants may provide needed information for regulatory-significant radionuclides. Some discussion of the use of simulants appears in the source term report.

7.4 Final Remarks

Most gaps associated with SFRs are related to either the loss of historical data and capabilities or to new technologies designed to make SFR economically competitive. Assuming that much of the historical experimental database from the IFR program can be recovered and the licensing codes can be revalidated to current standards, a SFR can most likely be licensed with metallic fuel clad with stainless steel, in either a pool or a loop configuration, with a Rankine power conversion cycle, with a burnup within the range demonstrated by metallic fuel. Variations of this theme have been constructed multiple times in both the U.S. and internationally, if both oxide metallic fuel forms are considered. A more aggressive design, with different cladding options, higher burnups, possible use of TRU fuel elements or targets and advanced power conversion cycles, will likely require a new irradiation testing facility be built.

8 REFERENCES

8.1 Previous Gap Reports

1. M. Corradini, A. Suo-Anttila, J. Hewson, T. J. Olivier, J. Philips, M. Farmer, G. Flanagan, and S. Miyhara. *Advanced Burner Reactor Sodium Technology Gap Analysis*. **FCR&D-REAC-2010-000034**. February 2010.

2. D.A. Powers, B. Clément, R. Denning, S. Ohno, and R. Zeyen, *Advanced Sodium Fast Reactor Accident Source Terms: Research Needs*. **SAND2010-5506**, Albuquerque, NM, September 2010.

3. J. Sackett, R. Wigeland, R. Bari, R. Budnitz, J. Cahalan, C. Grandy, D. Wade, M. Corradini, R. Denning, G. Flanagan, and S. Wright. *Advanced Sodium Fast Reactor Accident Sequences Technology Gap Analysis*. **FCR&D-REAC-2010-000126**. March 2010.

4. R. Schmidt, T. Sofu, T. Wei, J. Thomas, R. Wigeland, J. Carbajo, H. Ludewig, M. Corradini, H. Jeong, F. Serre, H. Ohshima, and Y. Tobita. *Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety*. **SAND2011-4145**. Albuquerque, NM. June 2010.

5. L. Walters, J. Lambert, K. Natesan, A. Wright, A. Yacout, S. Hayes, D. Porter, F. Garner, L. Ott, and M. Denman, *Sodium Fast Reactor Fuels and Materials: Research Needs*. **SAND2011-6546**, Albuquerque, NM, September 2011.

8.2 General References

1. 10 CFR 50 Appendix A. http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appa.html. Last Reviewed/Updated January 31, 2012.

2. ANS/ANSI 54.1, 1989.

3. K. Aoto, Y. Hirakawa, and T. Kuroda. “Corrosion Test of Mild Steel in High-Temperature Sodium Compounds.” *Proceedings of the Symposium on High Temperature Corrosion and Materials Chemistry*, **98-9**, The Electrochemical Society, Inc., Pennington, NJ, pp. 275-323, 1998.

4. K. Aoto and E. Yoshida. “Corrosion Behavior of Carbon Steel in Molten Sodium Compounds at High Temperatures and Effect of Oxygen Potential in Atmosphere.” *Materials at High Temperatures*. **18.** 2001 Science Reviews. pp. 187-191. 2001.

5. M.C. Billone. unpublished work. Argonne National Laboratory. 1994.

6. A. A. Bridges, A. E. Waltar, R.D. Leggett, R. B. Baker and J. L. Ethridge. “A Liquid-Metal Core Demonstration Experiment Using HT-9.” *Nuc. Tech.*. **102**. pp. 353. 1993.

7. DOE/NE. *DOE Complex-Wide Capability Report*. Unpublished. March 2006.

8. F. A. Garner. Chapter 6: "Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors." Vol. **10A.** *Materials Science and Technology: A Comprehensive Treatment*. VCH Publishers. pp. 419-543, 1994.

9. INL. *Next Generation Nuclear Plant Methods Technical Program Plan*. **INL/EXT-06-11804.** Idaho Falls, ID. September 2010.

10. INL. *Post Irradiation Examination Capabilities Guide.* **INL/MIS-09-16342.** Idaho Falls, ID. 2009.

11. W. Ivans. *The Current Nuclear Regulatory Environment: A PRISM Perspective.* M.S. Thesis. Department of Nuclear and Radiological Engineering. Gainesville, FL: University of Florida, 2006.

12. A. Makino. “Ignition delay and limit of ignitability of a single sodium droplet: theory and experimental comparisons.” *Combust. Flame*. **134(1-2)**. pp. 149-152. 2003.

13. R. D. Mariani, D. L. Porter, T. P. O'Holleran, S. L. Hayes and J. R. Kennedy. “Lanthanides in Metallic Nuclear Fuels: Their Behavior and Methods for Their Control.” *Journal of Nuclear Materials*, in press.

14. H.A. Morewitz, R.P. Johnson, and C.T. Nelson. "Experiments on Sodium Fires and Their Aerosols." *Nucl. Eng. Des.*, **42(1)**. pp. 123-135. 1977.

15. NEA. *Task Force on Advanced Reactor Experimental Facilities (TAREF)*. **NEA/CSNI/R(2010)12**, 2011.

16. E. J. Parma, M. E. Vernon, S. A. Wright, C. D. Peters, V. Tikare, P. S. Pickard, J. E. Kelly, and H. J. MacLean. *Global Nuclear Energy Partnership Fuels Transient Testing at the Sandia National Laboratories Nuclear Facilities: Planning and Facility Infrastructure Options*. **SAND2007-7308**. Albuquerque, NM. December 2007.

17. U.S. NRC. *Policy Statement on the Regulation of Advanced Reactors*. **73 FR 60612**. October 2008.

18. U.S. NRC. *Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor*. **NUREG-1369**. December 1991.

19. U.S. NRC. *Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Reactor.* **NUREG-1368**. February 1994.

20. U.S. NRC, http://www.nrc.gov/reading-rm/basic-ref/glossary/defense-in-depth.html, 2011.

21. A.M. Yacout, Y. Tsuboi, and N. Ueda. "Development of 4s and Related Technologies (2): Long Life Metallic Fuel." *Proceedings of ICAPP '09.* # 9195. Tokyo, Japan. May 10-14. 2009.

22. DOE/NE. *Nuclear Energy: Policies and Technology for the 21st Century, Nuclear Energy Advisory Committee.* http://www.ne.doe.gov/neac/neacPDFs/NEAC_Final_Report_Web%20Version.pdf. accessed 3/13/2012. November 2008.

9 BIBLIOGRAPHY

9.1 Sodium Technology References

1. Nuclear Regulatory Commission, Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), NUREG/CR-6944, Volumes 1-6, March 2008, Oak Ridge National Laboratory, Oak Ridge TN.
2. Scholtyssek, W. and K.K. Murata, *Sodium spray and jet fire model development within the contain-LMR code*. 1994, SAND93-2200C, Sandia National Laboratories.
3. Makino, A. and H. Fukada. *Ignition and combustion of a falling, single sodium droplet.* in *30th International Symposium on Combustion*. 2004. Chicago, IL: Combustion Inst.
4. Villermaux, E., *Fragmentation*. Annual Review of Fluid Mechanics, 2007. **39**: p. 419-446.
5. Eggers, J. and E. Villermaux, *Physics of liquid jets*. Reports on Progress in Physics, 2008. **71**(3).
6. Villermaux, E., P. Marmottant, and J. Duplat, *Ligament-mediated spray formation.* Physical Review Letters, 2004. **92**(7).
7. Marmottant, P.H. and E. Villermaux, *On spray formation*. Journal of Fluid Mechanics, 2004. **498**: p. 73-111.
8. Newman, R.N., *The Ignition and Burning Behavior of Sodium Metal in Air*. Prog. Nucl. Energy, 1983. **12**(2): p. 119-147.
9. Yuasa, S., *Spontaneous Ignition of Sodium in Dry and Moist Air Streams*. Proc. Combust. Instit., 1984. **20**: p. 1869-1876.
10. Makino, A., *Ignition delay and limit of ignitability of a single sodium droplet: theory and experimental comparisons*. Combust. Flame, 2003. **134**(1-2): p. 149-152.
11. Richard, J.B., R. Delbourgo, and P. Laffitte, *Spontaneous Ignition and Combustion of Sodium Droplets in Various Oxidizing Atmospheres at Atmospheric Pressure.* Proc. Combust. Instit., 1969. **12**: p. 39-48.
12. Morewitz, H.A., R.P. Johnson, and C.T. Nelson, *Experiments on Sodium Fires and Their Aerosols*. Nucl. Eng. Des., 1977. **42**(1): p. 123-135.
13. Jordan, S., et al., *Sodium Aerosol Behavior in Liquid-Metal Fast Breeder Reactor Containments*. Nuclear Technology, 1988. **81**(2): p. 183-192.
14. Modest, M.F., *Radiative Heat Transfer*. Second ed. 2003: Academic Press.
15. Clough, W.S. and J.A. Garland, *Behavior in the Atmosphere of the Aerosol from a Sodium Fire*. J. Nucl. Energy, 1971. **25**(9): p. 425-435.
16. Yoon, S.S., et al., *Experimental investigation on splashing and nonlinear fingerlike instability of large water drops*. Journal of Fluids and Structures, 2007. **23**(1): p. 101-115.
17. Newman, R.N. and J.F.B. Payne, *Burning Rates of Sodium Pool Fires.* Combust. Flame, 1978. **33**(3): p. 291-297.
18. C. Latge, G. Rodriguez and N. Simon, *Supercritical CO_2 Brayton Cycle for SFR: Na- CO_2 interaction and consequences on design and operation*, GLOBAL 2005, Oct. 9-13, 2005.
19. H. Ishikawa, S. Miyahara and Y. Yoshizawa, *Experimental Study of Sodium-Carbon dioxide Reaction*, Proceedings of ICAPP 2005, Paper 5688, 2005.
20. J.H. Choi, S.D. Suk, D. Cho, J.M. Kim, D. Hahn, J.H. Cahalan, *Capsule Test for Investigating the Sodium/Carbon-Dioxide Interaction*, Proceedings of ICAPP 2006, Paper 6369, Reno, NV, 2006.
21. H. Mikami, A. Shono, and H. Hiroi, “Sodium Leak at Monju (I) – Cause and Consequences,” *Tech. Committee Meeting on Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors*, O-aria, Ibaraki, Japan, November 11-14, 1996, International Atomic Energy Agency, Vienna, Austria..

22. M. T. Farmer, J. P. Schneider, B. Bonomo, G. Theofanous and B. W. Spencer, "Modeling and Database for Melt-Water Interfacial Heat Transfer," 2nd CSNI Specialist Meeting on Core Debris-Concrete Interactions, Karlsruhe, Germany, April 1-3, 1992.

23. M. T. Farmer, J. J. Sienicki and B. W. Spencer, "The MELTSPREAD-1 Computer Code for the Analysis of Transient Spreading in Containments," ANS Winter Meeting Session on Thermal Hydraulics of Severe Accidents, Washington, DC, November 11-15, 1990.

24. M. T. Farmer and S. Basu, "Melt Spreading Code Assessment, Modifications, and Initial Applications to the EPR Core Catcher Design," Proceedings ICAPP '09, Tokyo, Japan, May 10-14, 2009.

25. Aerosol Behavior Code Validation and Evaluation (ABOVE), Hilliard, R.K and Muhlstein, L.D., HEDL-SA-3113-FP, CONF-8409112-2, 1984.

26. Jordon and Cherdron, et.al. "Sodium Aerosol Behavior in Liquid Metal Fast Breeder Reactor Containments," Nuclear Technology Vol. 81, May 1988.

Additional reference material provided to expert panel members.

1. Anderson, M. H., Corradini. M.L., Bonazza, R., *Measurements of vapor flow regimes in liquid NaK pools*, Fusion Engineering and Design, V63-64, pp383-389, 2002.

2. Barthoux and J. Saroul, "Study of the Behavior of Fission Products in Liquid Sodium," Proc. Symp. Alkali Metal Coolants, International Atomic Energy Agency, 1967

3. Chang, Y.I., P.J. Finck, and C. Grandy, 2006, Advanced Burner Test Reactor Preconceptual Design Report, ANL-ABR-1 (ANL-AFCI-173). Argonne National Laboratory, Argonne, IL.

4. Buksah, Y.K., et al., 1997, Operation Experience of the BN-600 Fast Reactor, in *Nuclear Engineering and Design*, Vol. 173, p. 67-79.

5. Cherdron, W. & J. Charpenel, 1985, *Thermodynamic Consequences of Sodium Spray Fires in Closed Containments, Part 1-Experiments.* Laboratorium für Aerosolphysik und Filtertechnik, Kernforschungszentrum, Karlsruhe.

6. Cherdron, W. & S. Jordan, 1988, *Aerosol Release from Sodium Fires and their Consequences for Reactor Components.* International Atomic Energy Agency International Working Group Fast Reactors, Specialists' Meeting, Sodium Fires. Obninsk, USSR, June 6-9, pp. 44-56. Report Numbers XA0201146-1171, IWGFR/67.

7. Department of Energy Report to Congress, *Spent Nuclear Fuel Recycling Program Plan*, 2006.

8. Dostal V., Hejzlar P., Driscoll M.J., and N.E. Todreas, "A Supercritical CO_2 Brayton Cycle for Advanced Reactor Applications," Transactions of the American Nuclear Society, Vol. 85, Reno, Nevada, November 11-15, 2001, 2001.

9. Dostal V., Hejzlar P., Driscoll M.J., and N.E. Todreas, "A Supercritical CO_2 Gas Turbine Power Cycle for Next Generation Nuclear Reactors," ICONE10-22192, 10th International Conference on Nuclear Engineering, Arlington, Virginia, April 14-18, 2002.

10. Dostal V., Hejzlar P., Driscoll M.J. and Wang Y., "Supercritical CO_2 Cycle for Fast Gas-Cooled Reactor," Paper GT2004-54242, Proceedings of ASME TurboExpo: Power for Land, Sea and Air, Vienna, Austria, June 14-17, 2004.

11. P. Dufour, P. Anzieu, D. Lecarpentier, J.P. Serpantie, *Research and Development Plan for the Future Sodium Fast Reactors in France,* Proc. of ICAPP 2006, Paper 6285, Reno, NV, 2006.

12. S. Eckert, G. Gerbeth, *Velocity Measurements in Liquid Sodium by means of Ultrasonic Doppler Velocimetry,* Experiments in Fluids, V32, p542-550, 2002.

13. S. Eckert, G. Gerbeth, *Velocity measurements at high temperature by Ultrasonic Doppler Velocimetry using an acoustic wave guide,* V35, p381-388, 2003.

14. King, T.L., R.R. Landry, E.D. Throm, and J.N. Wilson, 1991, Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor, NUREG-1369. Nuclear Regulatory Commission, Washington, D.C.

15. Krolikowski, Theresa S., 1968, *Violently Sprayed Sodium-Air Reaction in an Enclosed Volume*, ANL-7472. Argonne National Laboratory, Argonne, Illinois.

16. Luster, V.P. and K.F. Freudenstein, 1996, Feedback from Practical Experience with Large Sodium Fire Accidents, in *Technical Committee Meeting on Evaluation of Radioactive Materials Release and Sodium Fires in Fast Reactors*, O-aria, Ibaraki, Japan, November 11-14, 1996. International Atomic Energy Agency, Vienna, Austria.

17. Makino, Atsushi, *2005 Ignition and Combustion of a Falling, Single Sodium Droplet*. Proc. Combust. Instut., Vol. 31, pp 20747—2054.

18. Makino, Atsushi, *2006 Ignition Delay and Limit of Ignitability for Sodium Pool (Theory and Experimental Comparisons)*. JSME International Journal, Series B, Vol. 49, No. 1.

19. Malet, J.C. et al., 1981, *Potential Results of Spray and Pool Fires. Nuclear Engineering and Design*, North-Holland Publishing Company (68:195-206).

20. Miyake, Osamu *et al*., 1991, *Sodium Pool Combustion Codes for Evaluation of Fast Breeder Reactor Safety*. Journal of Nuclear Science and Technology, 28[2], pp 107-121.

21. Ohno, S. *et al*., 1990, *Test and Code Development for Evaluation of Sodium Fire Accidents in the FBRs*. International Fast Reactor Safety Meeting, Power Reactor and Nuclear Fuel Development Corporation, Japan, pp 241-250.

22. Olivier, T.J., R.F. Radel, S.P. Nowlen, T.K. Blanchat, and J.C. Hewson, Metal Fire Implications for Advanced Reactors, Part 1: Literature Review, SAND2007-6332. Sandia National Laboratories, Albuquerque, NM, October 2007.

23. Poplavsky, V.M. *et al*., 2004, Review of Fast Reactor Operational Experience Gained in the Russian Federation; Approaches to the Co-Ordinated Research Project, in *Operational and Decommissioning Experience with Fast Reactors meeting*, Cadarache, France, March 11-15, 2002. International Atomic Energy Agency, Vienna, Austria.

24. Randich, E., Smaardyk, J.E. and A.U. Russell, 1983, *Large-Scale Exploratory Tests of Sodium/Limestone Concrete Interactions*, SAND82-2315. Sandia National Laboratories, Albuquerque, NM.

25. Randich, E. & A.U. Russell, 1983, *Large Scale Exploratory Test of Sodium/Magnetite Concrete Interactions*, SAND83-0356. Sandia National Laboratories, Albuquerque, NM.

26. Scholtyssek, W. and K.K. Murata, 1993, *Sodium Spray and Jet Fire Model Development within the CONTAIN-LMR.* Sandia National Laboratories: Albuquerque, NM.

27. Suo-Anttila, A.J., 1983, *SLAM - A Sodium-Limestone Concrete Ablation Model*. Sandia National Laboratories: Albuquerque.

28. Suresh Kumar, K.V. et al., 2004, Fast Breeder Test Reactor; 15 Years of Operating Experience, in *Operational and Decommissioning Experience with Fast Reactors meeting*, Cadarache, France, March 11-15, 2002. International Atomic Energy Agency, Vienna, Austria.

29. Tsai, S.S., 1980, *The NACOM Code for Analysis of Postulated Sodium Spray Fires in LMFBRs*. Brookhaven National Laboratory: Upton, NY.

30. N.S. Bharasi, et al., "Effect of flowing sodium on corrosion and tensile properties of AISI type 316N stainless steel at 823 K," J. Nuclear Materials, 377 (2008) 378-384.

31. A.W. Castleman, Jr. and I.N. Tang, "Fission Product Vaporization from Sodium Systems," BNL 13099, Brookhaven National Laboratory, November 1968.

32. C.F. Clement and M. J. Dolias, "Sodium Aerosol Formation in an Argon Flow Over Hot Sodium," AERE R 12489, Harwell Laboratory, UK, January 1987.

33. H.J. Ettinger, W.D. Moss, and H. Busey, “Characteristics of the Aerosol Produced from Burning Sodium and Plutonium,” Nuclear Science and Engineering, 30 (1967) 1-13.

34. S. Jordan, “Release of Fission Products from Contaminated Sodium Fires,” KfK, Kernforschungszentrum Karlsruhe, Germany.

35. R.N. Morris, “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT) Volume 3: Fission-product Transport and Dose PIRTs,” NUREG/CR-6944, Vol. 3, Oak Ridge National Laboratory, Oak Ridge, TN, March 2008.

36. S.R. Pillai, N.S. Barasi, H.S. Khatak, A.L.E. Terrance, R.D. Kale, M. Rajan, and K.K. Rajan, “Corrosion Behavior and Tensile Properties of AISI 316LN Stainless Steel Exposed to Flowing Sodium at 823 K,” Journal of Materials Engineering and Performance, 98 (2000) 98-102.

37. M. Farmer, D. Cho, Private Communication, Argonne National Lab, June 2006.

38. C. Forest, et al, *The Dynamo in the Corn-Field*, http://www.physorg.com/news2575.html

39. J.P. Herzog, M.L. Corradini, *Modelling of Lithium-Lead/Water Interactions in a Fusion Reactor Design,* Transactions of the ANS, V41, 1983.

40. J.P. Herzog, M.L. Corradini, *Liquid-Metal/Water Reaction Experiments and Analysis*, Fusion Technology, V15, No 2, Part 1a, March 1989.

41. M. Nornberg, C. Forest et al, *Experimental and numerical studies on the role of turbulence in the Madison Dynamo Experiment*, Proc. of MHD Turbulence in Planetary & Stellar Dynamics, 2006.

42. S. Piet, M. Corradini et al, *Liquid Metal Chemical Reaction Safety in Fusion Facilities,* Fusion Engr. Design, V5, p 273-298, 1987.

9.2 Fuels and Materials References

Fuels

1. S. L. Hayes, B. A. Hilton, H. J. MacLean, D. J. Utterbeck, W. J. Carmack and K. O. Pasamehmetoglu (2009), “Status of Transuranic Bearing Metallic Fuel Development”, The Nuclear Fuel Cycle: Sustainable Options & Industrial Perspectives (GLOBAL-2009), Paris, France, 6-11 September 2009, pp. 1997-2005.

2. S. L. Hayes, J. R. Kennedy, B. A. Hilton, R. S. Fielding, T. A. Hyde, D. D. Keiser, Jr. and D. L. Porter (2009), “Metallic Fuels,” in Nuclear Fuels & Materials Spotlight (Ed., K. O. Pasamehmetoglu), Idaho National Laboratory, March 2009, pp. 21-41.

3. G. L. Hofman and L. C. Walters (1994), “Metallic Fast Reactor Fuels,” in Materials Science and Technology, Vol. 10A (Ed., B. R. T. Frost), VHC Publishers Inc., New York, pp. 1-43.

4. G. L. Hofman, S. L. Hayes and M. C. Petri (1996), "Temperature Gradient Driven Constituent Redistribution in U-Zr Alloys," Journal of Nuclear Materials, 227: 277-286.

5. D. D. Keiser, Jr. (2009), “The Development of Fuel Cladding Chemical Interaction Zones in Irradiated U-Zr and U-Pu-Zr Fuel Elements with Stainless Steel Cladding,” in Nuclear Reactors, Nuclear Fusion and Fusion Engineering (Eds., A. Aasen and P. Olsson), Nova Science Publishers, Hauppauge, NY, pp. 163-194.

6. D. D. Keiser, Jr., J. R. Kennedy, B. A. Hilton and S. L. Hayes (2008), “Development of Metallic Nuclear Fuels for Transmutation Applications: Materials Challenges,” JOM—Journal of the Minerals, Metals & Materials Society, 60: 29-32.

7. J. R. Kennedy, B. A. Hilton, S. L. Hayes and D. C. Crawford (2006), “Metallic Fuels for Actinide Transmutation,” *invited* presentation at the Ninth OECD/NEA Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, Nîmes, France, 25-29 September 2006.

8. Y. S. Kim, S. L. Hayes, G. L. Hofman and A. M. Yacout (2006), “Modeling of Constituent Redistribution in U-Pu-Zr Metallic Fuel,” Journal of Nuclear Materials, 359: 17-28.

9. H. J. MacLean and S. L. Hayes (2007), "Irradiation of Metallic and Oxide Fuels for Actinide Transmutation in the ATR," Proceedings of the International Conference on Advanced Nuclear Fuel Cycles and Systems (GLOBAL-2007), Boise, ID, 9-13 September 2007.
10. R. D. Mariani, D. L. Porter, T. P. O'Holleran, S. L. Hayes and J. R. Kennedy, "Lanthanides in Metallic Nuclear Fuels: Their Behavior and Methods for Their Control," Journal of Nuclear Materials, in press.
11. M. K. Meyer, S. L. Hayes, W. J. Carmack and H. Tsai (2009), "The EBR-II X501 Minor Actinide Burning Experiment," Journal of Nuclear Materials.392, 176-183.
12. A. A. Bridges, A. E. Waltar, R.D. Leggett, R. B. Baker and J. L. Ethridge (1993), "A Liquid-Metal Core Demonstration Experiment Using HT-9," *Nuc. Tech*., 102, 353.
13. J. J. Carbajo, G. L. Yoder, S. G. Popov and B. K. Ivanov (2001), "A Review of the Thermophysical Properties of MOX and UO_2 Fuels," *Jour. Nuc. Mat.,* 299, 181-1908.
14. D. C. Crawford, D. L. Porter and S. L. Hayes (2007), "Fuels for Sodium-cooled Fast Reactors: U.S. Perspective," *Jour. Nucl. Mat*., 371, 202-231.
15. S. L. Hayes, private communication (2011).
16. IAEA (1997), "Thermophysical Properties of Materials for Water Cooled Reactors," IAEA-TECDOC-949.
17. J. D. B. Lambert, J. H. Bottcher, R. V. Strain, K. C. Gross, M. J. Lee, R. P. Colburn, S. Ukai, S. Nomura, and T. Odo (1990), "Run-Beyond-Cladding-Breach Oxide Testing in EBR-II," Proc., BNES Conference: Fast Reactor Core and Fuel Structural Behavior, 17-23, BNES, London.
18. J. D. B. Lambert and R. V. Strain (1994), "Oxide Fuels," Vol. 10A (Ed. B. R. T. Frost), VHC Publications Inc., New York, 109-190.
19. K. Morimoto, M. Kato, M. Ogasawara and M. Kashimura (2008), "Thermal Conductivities of Hypostoichiometric (U,Pu, Am)O_{2-x} Oxide," *Jour. Nuc. Mat.* 374, 378-385.
20. R. O. Meyer (1974), *Jour. Nuc. Mat*., 50, 11-24.
21. L. D. Noble, S. Derby, B. Sarma, M. Plummer, N. Nielsen, G. R. Pflasterer and M. L. Johnson (1972), "SEFOR Core II Doppler and Fuel Temperature Measurements at Power," *Trans. ANS*., **15**, 500.
22. NRC (2011), "Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4, and MATPRO," NUREG/CR-7024.
23. T. H. Bauer and J. W. Holland, "In-Pile Measurement of the Thermal Conductivity of Irradiated Metallic Fuel," Nucl. Technol. 110 (1995) 407.
24. L. A. Lawrence, F. E. Bard, N. S. Canon. "A Decade of LMR Progress and Promise." Proc. ANS Conf. La Grange Park, IL: ANS, pp.170-175 (1990).

In-Core Materials

25. F. A. Garner, Chapter 6: "Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors," Vol. 10A of Materials Science and Technology: A Comprehensive Treatment, VCH Publishers, 1994, pp. 419-543.
26. F. A. Garner, "Void Swelling and Irradiation Creep in Light Water Reactor Environments", Chapter 10 in Understanding and Mitigating Ageing in Nuclear Power Plants: Materials and Operational Aspects of Plant Life Management (PLiM), Woodhead Publishing Series in Energy: Number 4, Edited by Philip G. Tipping, Woodhead Publishing Limited, 2010 Woodhead Publishing Series in Energy: Number 4; Woodhead Publishing: Cambridge, UK, 2010;Chapter 10, pp 308–356.
27. F. A. Garner, "Radiation Damage in Austenitic Stainless Steels", Chapter 65 in Comprehensive Nuclear Materials, in press.
28. F. A. Garner, d. J. Edwards, S. M. Bruemmer et al., Proceedings of Fontevraud 5, Contribution of Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors, 2002, paper #22.
29. B. J. Makenas, S. A. Chastain and B. C. Gneiting, Proceedings of LMR: A Decade of LMR Progress and Promise, ANS: La Grange Park, IL, 1990, pp.176-183.
30. V. S. Neustroev, Z. E. Ostovsky, A. A. Teykovtsev, V. K. Shamardin and V. V. Yakolev, Proceedings of Sixth Russian Conference on Reactor Materials Science: 11-15 September 2000, Dimitrovgrad Russia.

31. F. A. Garner, M. B. Toloczko and B. H. Sencer, "Comparison of Swelling and Irradiation Creep Behavior of fcc-Austenitic and bcc-Ferritic/Martensitic Alloys at High Neutron Exposure", J. of Nuclear Mater. 276 (2000) 123-142.
32. R. L. Klueh and D. R. Harries, High-Chromium ferritic and Martensitic Steels for Nuclear Applications", ASTM Monograph 33, 2001.
33. Y. K. Bibilashvili, V. I Kuznetsov, A. V. Medvedev, A. V. Kozlov, L. P. Sinelnikov, V. N. Golovanov, A. V. Povstyanko, A. E. Fedoseev. "Development of Work to Increase Fuel Burn-up in the Core of the BN-600 Fast Reactor", Proceedings of Technical Meeting of International Atomic Energy Agency on The Current Status and Future Prospects for the Fuel Cycle of the Liquid Metal Coolant Type Reactors, LMFR, Obninsk, Russia, November 21-23, 2005.
34. A. V. Povstyanko, A. Ye. Fedoseev, O. Yu. Makarov, V. I. Prokhorov, "A Study of Four Experimental Fuel Subassemblies using EP-450 Ferritic-Martensitic Pin Claddings and Hexagonal Ducts after Irradiation to 108-163 dpa in the BOR-60 Reactor", Proceedings of ANS Winter Meeting, Las Vegas, NV, 2010.
35. V. N. Voyevodin, O.V. Borodin, V. V. Bryk, A. S. Kalchenko, Y. E. Kupriyanova, A. V. Permyakov and F. A. Garner, "Comparison of ion-induced swelling of EP-450 and EP-450 ODS ferritic-martensitic alloys", Proceedings of First International Workshop on Dispersion Strengthened Steels for Advanced Nuclear Applications, April 4-8, 2011, Aussois, France, in press.
36. J.J. Laidler, R.J. Jackson, Trans. ANS 49, 90.
37. R.D. Leggett, L.C. Walters, J. Nucl. Mater. 204 (1993) 23.
38. B. H. Sencer, J. R. Kennedy, J. I. Cole, S. A. Maloy, F. A. Garner, , J. Nucl. Mater. 393 (2009) 235-241.
39. F. A. Garner, M. B. Toloczko and B. H. Sencer, J. Nucl. Mater. 276 (2000) 123-142.
40. G. R. Odette, M. J. Alinger, B. D. Wirth, Ann. Rev. Mater. Res. 38, (2008) 471.
41. G. R. Odette and D. T. Hoelzer, "Recent Progress in Developing Irradiation Tolerant Nanostructured Ferritic Alloys: Transforming Helium Accumulation From a Liability to an Asset", J. of Metals, in press.
42. M. J. Alinger, G. R. Odette and D. T. Hoelzer, Acta Mater. 57, 392 (2009)
43. M. Ohnuma ,T. Suzuki, S. Ohtsuka et al., Acta Mater. 57 (2009) 5571
44. M. K. Miller, K. F. Russell, D. T. Hoelzer, J. Nuc. Mat., 351(2006) 261
45. H. Sasasegawa, L. Chaffron , L. Legendre et al., J. Nucl. Mat. 384-2 (2009) 115
46. M. B. Toloczko, D. S. Gelles, F. A. Garner, R. J. Kurtz and K. Abe, J. Nucl. Mater. 329-333 (2004) 352-355.
47. "Experimental Facilities for Sodium Fast Reactor Safety Studies," 2011 OECD report NEA/CSNI/R(2010)12

Fabrication

48. D.E. Burkes, R. S. Fielding, D. L. Porter, D. C. Crawford and M. K. Meyer, A US Perspective on Fast Reactor Fabrication Technology and Experience. Part I: Metal Fuels and Assembly Design, J. of Nuclear Materials, 389 (2009) p. 458-469.
49. D.E. Burkes, R. S. Fielding, D. L. Porter, M. K. Meyer and B. J. Makenas, A US Perspective on Fast Reactor Fabrication Technology and Experience. Part II: Ceramic Fuels, J. of Nuclear Materials, 393 (2009) p. 1-11.

Fuel Performance Codes

50. M.C. Billone, et al., Proc. Int. Conf. Reliable Fuels for Liquid Metal Reactors, Tucson, Arizona, September 7-11, 1986 (American Nuclear Society).
51. ANL-IFR-169, Integral Fast Reactor Program Annual Progress Reports FY 1991, June 1992.

52. V.Z. Jankus, and R.W. Weeks, Nucl. Eng. Des., 18 (1972) 83.

53. A. Boltax, et. al., Int. Fast Reactor Safety Meeting, Snowbird, 1990, Vol. II, p. 427-436, Hinsdale, IL, American Nuclear Society.

54. Y.Y. Liu, S. Zawadzki, M.C. Billone, Development of LIFE-4CN, Mathematical/Mechanical Modeling of Reactor Fuel Elements, (ed.) Y. R. Rashid, ISSN 0172-0465, p. 59 (1979).

55. Y.I. Chang, Y. I., "The Integral Fast Reactor," Nucl. Technol., 88 (1989) 129.

56. M.C. Billone, unpublished work, Argonne National Laboratory, 1994.

57. D.C. Crawford, D. L. Porter, and S. L. Hayes, "Fuels for sodium-cooled fast reactors: US perspective," J. Nucl. Mater., 371 (2007) 202.

58. A.M. Yacout, Y. Tsuboi, and N. Ueda, "Development of 4s and Related Technologies(2): Long Life Metallic Fuel," Proceedings of ICAPP '09, Tokyo, Japan, May 10-14, 2009, paper 9195.

59. ARC reference.

60. R. B. Baker and D. R. Wilson, "SIEX3 — A Correlated Computer Code for the Prediction of Fast Reactor Mixed-Oxide Fuel and Blanket Pin Performance," HEDL-TME 85-18, Hanford Engineering Development Laboratory, Richland, WA, January, 1986.

61. U. Imke, D. Struwe, and W. Pfrang, "DEFORM-4C: Steady State and Transient PreFailure Pin Behavior," Forschungszentrum Karlsruhe Draft Report, October, 1995.

62. T.H. Hughes and J.M. Kramer, Proc. Conf. on the Science and Technology of Fast Reactor Safety, vol. 2 (BNES, Guernsey, UK, 1986) p. 427.

63. N.S. Cannon, F.H. Huang and M.L. Hamilton, Effects of Radiation on Materials: 14th Int. Symp., Vol. II, eds. N.H. Packen, R.E. Staller and A.S. Kumar, ASTM-STP 1046 (1991) p. 729.

64. D. C. Crawford, D. L. Porter and S. L. Hayes (2007), "Fuels for Sodium-cooled Fast Reactors: U.S. Perspective," Journal of Nuclear Materials, 371: 202-231.

65. D. C. Crawford, D. L. Porter, S. L. Hayes, M. K. Meyer, D. A. Petti and K. Pasamehmetoglu (2007), "An Approach to Fuel Development and Qualification," Journal of Nuclear Materials, 371: 232-242.

66. J. M. Kramer, Y. Y. Liu, M. C. Billone, and H. C. Tsai, "Behavior of Metallic Fast Reactor Fuels During Extended Transients," Proc. ANS/ENS Int'l Conf., November 15-20, 1992, Chicago, IL (Conf. No. 921102), 208-209.

67. A. E. Wright, D. S. Dutt, and L. J. Harrison, "Fast Reactor Safety Testing in TREAT in the 1980s," Proc. 1990 Int'l Fast Reactor Safety Meeting, Snowbird, UT, August 12-16, 1990 (Conf. No. 990804), Vol. II, pp. 233-244.

Structural Materials

68. O.K. Chopra and K. Natesan, "Structural Materials for Advanced Burner Reactors" Unpublished Report, March 2007

69. K. Nateson, M. Li, S. Majumdar, R. Nanstad, T.-L. Sham, "Preliminary Assessment of code Qualification for ABR Structural Materials," ANL-AFCI-244, September 2008.

9.3 Accident Analysis References

1. Sackett, J. I., Singer, R.M., Amorosi, A. "Design Features to Maximize Simplicity, Operability and Inherent Safety of LMFBRs". Trans. ANS, Vol. 38, June 1982

2. Sackett, J.I., "Safety Philosophy in Upgrading the EBR-II Protection System", proceedings of the International Meeting on Fast Reactor Safety and Related Physics, Chicago, Illinois, November, 1976

3. Singer, R.M., et al., “Decay Heat Removal and Dynamic Plant Testing at EBR-II,” Second Specialists’ Meeting on Decay Heat Removal and Natural Convection in LMFBRs, BNL, Upton, New York, April 17-19, 1985
4. Lambert, J.D.B., et al., “performance of Breached LMFBR Fuel Pins During Continued Operation,” BNES Conference on Nuclear Fuel Performance, March 25-29, 1985
5. Walters, L.C., Seidel, B.R., Kittel, J.S., “Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeder Reactors, “Nuclear Technology, Vol. 65, May 1984
6. Sackett, J.I., Buschman, H.W., “Safety Implications from 20 Years of Operation at EBR-II”, International Topical Meeting on Fast Reactor Safety, Knoxville, Tennessee, April 21-25
7. Chang, Y.I., Konomura, M., Lo Pinto, P., “A Case for Small Modular Fast Reactor, Journal of Nuclear Science and Technology, Vol. 44, No. 3, p. 264-269 (2007)
8. Fistedis, S.H. (ed.), “The Experimental Breeder Reactor-II Inherent Safety Demonstration,” Nucl. Eng. Des., 101(1),1 (1987)
9. Chang, Y.I., “Advanced Nuclear Energy System for the Twenty-First Century,” PHYSOR 2002, Seoul, Korea, Oct. 7-10 (2002)
10. Dostal, V., “A supercritical carbon dioxide cycle for next generation nuclear reactors,” Ph.D. Dissertation, MIT (2004)
11. Seidel, B.R., Walters, L.C., Chang, Y.I. “Advances in metallic nuclear fuel,” J. Metals, 39[4], 10 (1987)
12. Chang, Y.I., “The Integral Fast Reactor,” Nucl. Technol., 88[11], 129 (1989)
13. Walters, L.C., Hoffman, G.L., *Metallic Fast Reactor Fuels: A Comprehensive Treatment*, Material Science and Technology, Vol. 10A, eds. R.W. Cahn et al. (1994)
14. U.S. Nuclear Regulatory Commission, *Safety Evaluation* Report *related to Operation of the Fast Flux Test Facility*, NUREGE-0358,August, 1978
15. U.S. Nuclear Regulatory Commission, *Safety Evaluation Report related to the Construction of the Clinch River Breeder Reactor Plant*, NUREG-1368, February,1994
16. U.S. Nuclear Regulatory Commission, *Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor(SAFR)Liquid Metal Reactor,* NUREG-1369, December, 1991
17. U.S. Nuclear Regulatory Commission, Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid Metal Reactor, NUREG-1368, February 1994
18. U.S. Nuclear Regulatory Commission, *Staff Plan to Make a Risk-Informed and Performance-Based Revision to 10CFR Part 50,* SECY-06-0007, January 9, 2006
19. U.S. Nuclear Regulatory Commission, *Approaches to Risk-Informed and Performance-Based Requirements for Nuclear Power Reactors,*10 CFR Parts 50 and 53, RIN 3150-AH-81, May 4, 2006
20. Koch, L.J., Experimental Breeder Reactor-II, *An Integrated Experimental Fast Reactor Nuclear Power Station, Argonne National Laboratory*
21. Simpson, D.E., *Resolution of Key Safety-Related Issues in FFTF Regulatory Review*, Proc. Intl. Mtg. Fast Reactor Safety and Related Physics, CONF-761011, Vol. II, pp. 400-41, American Nuclear Society, Chicago Illinois, Oct. 5-8, 1976
22. Strawbridge, *Safety-Related Criteria and Design Features in the Clinch River Breeder Reactor Plant,* Proc. Fast Reactor Safety Meeting, CONF-740401-P1, pp. 72-92, American Nuclear Society, Beverly Hills, California, April 2-4, 1974
23. Schade, A.R., Simpson, D.E., *FFTF Regulatory Review for Operating Authorization.,* Proc. Intl. Mtg. Fast Reactor Safety Technology, Vol. 5, pp. 2425-2430, American Nuclear Society, Seattle, Washington, August 19-23, 1979

24. Slember, R.J., *Safety-Related Design Considerations for the Clinch River Breeder Reactor Plant,* Proc. Intl. Mtg. Fast Reactor Safety and Related Physics, CONF-761001, Vol. I, pp. 112-125, American Nuclear society, Chicago, Illinois, October 5-8, 1976

25. Strawbridge, L.E., Clare, G.H., *Exclusion of Core Disruptive Accidents from the Design Basis Accident Envelope in CRBRP,* Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp. 317-327, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

26. Gitter, J.G., Akhtar, M.W., *An Assessment of the Clinch River Breeder Reactor Core and Containment Response to Core Disruptive Accidents,* Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp. 463-470, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

27. Theofanous, T.G., Bell, C.R., *An assessment of CRBR Core Disruptive Accident Energetics,* Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1 pp. 471-480, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

28. Magee, P.M., *U.S. ALMR Licensing Status,* Proc. Intl. Topical Mtg. Advanced Reactor Safety, Vol. 2, pp. 1011-1017, American Nuclear Society, Pittsburg, Pennsylvania, April 17-21, 1994

29. American National Standards Institute / American Nuclear Society, *General Safety Design Criteria for a Liquid Metal Reactor Nuclear power Plant,* ANSI/ANS-54.1-1989, American Nuclear Society, La Grange Park, Illinois

30. Chang, Y.I., Finck, P.J., Grandy, C., *Advanced Burner Test Reactor: Preconceptual Design Report*, ANL-ABR-1 (ANL-AFCI-173), ANL Nuclear Engineering Division, Sept. 5, 2006

31. Herzog, J.P., et al., *Code Validation With EBR-II Test Data* ,ANL/CP-74826, ASME/AICHE National Heat Transfer Conference, August 9-12, 1992, San Diego, California

32. Golden, G.H., et al., *Evolution of Thermal-Hydraulics Testing in EBR-II,* Nuclear Engineering and Design, 1987, vol. 101, 3-12

33. Planchon, H.P., et al., *The Experimental Breeder Reactor II Inherent Shutdown and Heat Removal Tests – Results and Analyses,* Nuclear Engineering and Design, 1986, vol. 91, 287-296

34. Mohr, D., et al., *Loss-of-Primary-Flow-Without-Scram Tests: Pretest Predictions and Preliminary Results,"* Nuclear Engineering and Design, 1987, vol. 101, 45-56

35. Feldman, E.E., et al., *EBR-II Unprotected Loss-of-Heat-Sink Predictions and Preliminary test Results,* Nuclear Engineering and Design, 1987, vol. 101, 57-66

36. Mohr, D., Feldman, E.E., *A Dynamic Simulation of the EBR-II Plant during Natural Convection with the NATDEMO code,* Decay Heat Removal and Natural Convection in Fast Breeder Reactors, Agrawal K. and Guppy G (eds.) Hemisphere Publication Corp., Washington D.C., 1981, 207-223

37. Mohr D., Chang, L.K., Planchon, H.P., *Validation of the HOTCHAN Code for Analyzing the EBR-II Core Following and Unprotected Loss of Flow,* Transactions of the American Nuclear Society, 1988, vol. 57, 318-319

38. Mohr, D., Chang, L.K., *Perturbations of Reactor Flow and Inlet Temperature for EBR-II Reactivity-Feedback Validation,* Proceedings of International Topical Meeting on Fast Reactor Safety, 1985, vol. 2, April, 617-627

39. Lehto, W.K., et al., *Safety Analysis for Loss-of-Flow and Loss-of-Heat-Sink without Scram Tests in EBR-II,* Nuclear Engineering and Design, 1987, vol. 101, 35-44

40. Herzog, J.P., *SASSYS Validation with the EBR-II Shutdown Heat Removal Tests,* Fifth Proceedings of Nuclear Thermal Hydraulics, 1989, November, 416-423

41. Chang, L.K., et al., *Demonstration of EBR-II Power Maneuvers without Control Rod Movement,* Nuclear Engineering and Design, 1989, vol. 113, 141-148

42. Lehto, W.K., Dean, E.M., Fryer, R.M., *Reactor Safety Implications of Pump Run-up Tests in EBR-II,* Nuclear Engineering and Design, 1988, vol. 110, 47-53

43. Dean, E.M., Lehto, W.K., Larson, H.A., *System Modeling and Simulation at EBR-II*, Proceedings of the Sixth Power Plant dynamics, Control and Testing Symposium, 1988, vol. 2, April, 47.01-47.14

44. Betten, P.R., et al., *Verification of Steady State Temperature Predictions in and Instrumented LMFBR Driver Subassembly,* Third International Meeting on Reactor Thermal Hydraulics, 1985, vol. 2, October, 17.H1-17.H.6

45. Betten, P.R., et al., *Conceptual Design Basis and Temperature Predictions for a Simulated Instrumented LMFBR Blanket Subassembly,* Third International Meeting on Reactor Thermal Hydraulics, 1985, vol.2, October, 17.G1-17.G6

46. Gopalakrishnan, A., Gillette, J.L., *EBRFLOW – A Computer Program for Predicting the Coolant Flow Distribution in the Experimental Breeder Reactor – II,* Nuclear Technology, 1973, vol. 17, March, 205-216

47. Chang, L.K., et al., *Experimental and Analytical Study of Loss-of-Flow Transients in EBR-II Occurring at Decay Power Levels*, Alternative Energy Sources VIII, Veziroglu, T.N. (ed.), Hemisphere Publication Corp., Washington D.C., 1981, vol. 3, 471-484

48. Chang, L.K., Mohr D., Planchon, H.P., *Loss-of-Flow-Without-Scram Tests in Experimental Breeder Reactor-II and Comparison with Pretest Predictions*, Nuclear Safety, 1987, vol. 28-2, April-June, 199-211

49. Dunn, F.E. et al., *The SASSYS-1LMFBR Systems Analysis Code,* Proceedings of the International Topical Meeting on Fast Reactor Safety, 1985, May, 999-1006

50. Till, C.E., Chang, Y.I., *The Integral Fast Reactor Concept,* Proceedings of the American Power Conference, 1986, vol. 48, 688-693

51. Hill, D.J., *SASSYS Validation Studies,* Proceedings of the International Topical Meeting on Safety of Next Generation Power Reactors, 1988, May, 686-692

52. Tentner, A.M., et al., *The SAS4A LMFBR Whole Core Accident Analysis Code,* Proceedings of the International Topical Meeting on Fast Reactor Safety, 1985, May, 989-997

53. Lehto, W.K., et al., *Experimental Breeder Reactor-II Dynamic Modeling and Code Verification,* Transactions of the American Nuclear Society, 1983, vol. 44, 310-311

54. Madell, J.T., et al., *Dynamic Analysis of the EBR-II Plant with the DNSP-ND Simulation Package,* Safety, Status and Future of Non-Commercial Reactors and Irradiation Facilities, 1990, vol. 2, September, 705-716

55. Singer, R.M., et al., *Studies of Thermal-Hydraulic Phenomena in EBR-II,* Nuclear Engineering and Design, 1980, vol. 62, 219-232

56. George, T.L., Masterson, R.E., Basehore, K.L., *A Modified Version of COBRA for Whole-Core LMFBR Transient Analysis,* Transactions of the American Nuclear Society, 1979, vol. 32, 531-532

57. Ragland, W.A., Feldman, E.E. *EBR-II In-Vessel Natural Circulation Experiments on Hot and Cold Pool Stratification,* International Fast Reactor Safety Meeting, 1990, vol. 3, August, 513-522

58. Larson, H.A., et al., *Computer Programs for Describing the Dynamic Behavior of EBR-II,*EBR-II Project, Argonne National Laboratory, Argonne Illinois, Sept. 1973

59. Cheung, A.C., et al., *A comparison of Results Between the DEMO 1-D and the COMMIX 3-D Analysis of the CRBRP-IHX under a Natural-Circulation Event,* CRBRP, Sept. 8, 1973

60. Domanus, H.M., Sha, W.T., *Numerical Simulation of Combined Natural and Forced Convection During thermal-Hydraulic Transients*

61. Sha, W.T., *et al.*, COMMIX-1: *A Three-Dimensional, Transient, Single-Phase Component Computer Program for Thermal-Hydraulic Analysis*, NUREGE/CR-9785, ANL-77-96 (Sept 1978)

62. Domanus, H.M., et.al, *Numerical Results Obtained from the Three-Dimensional Transient, Single-Phase Version of the COMMIX Computer Code*, NUREG/CR-0355, ANL-CT-78-3, (Oct. 1977)

63. Domanus, H.M., Sha, W.T., *Numerical Results for a Hexagonal Fuel Assembly with a Planar Blockage using the COMMIX-1A Computer Code,* NUREG/CR-0483, ANL-CT-79-8 (Sept. 1978)

64. Domanus, H.M., et al., *Analysis of a Loss of Piping Integrity Transient in an LMFBR Fuel Assembly*, Trans. Am. Nucl. Soc. 32, pp. 827-829

65. Domanus, H.M., et al., *Computational Results for a 7-Pin Hexagonal Fuel Assembly during a Flow rundown Transient using the COMMIX-1A Computer Code*, NUREG/CR-1285, ANL-CT-80-10 (Jan 1980)

66. Sha, W.T., et al., *A New approach for Rod-Bundle Thermal-Hydraulic* Analysis, Proceedings of the International Meeting on Nuclear Power Reactor Safety, Brussels, Belgium (Oct. 16-19, 1978)

67. Sha, W.T., *An overview of Rod Bundle Thermal Hydraulics*, NUREG/CR-1825, ANL079-10 (Nov. 1980)

68. Bordner, G.L., et al., *Operation and Evaluation of the SLSF-P2 Loss-of-Flow Safety Experiment*, Presented at the ASME Winter Meeting on Fluid Transient and Acoustics in the Power Industry, San Francisco, California, (Dec. 10-15, 1978)

69. Lorenz, J.J., Howard, P.A., *A Study of CRBR Outlet Plenum Thermal Oscillation during Steady State Conditions*, ANL-CT-76-36 (July 1976)

70. Howard, P.A., Lorenz, J.J., *A Study of CRBR Outlet Plenum Thermal Oscillation during Steady State Conditions*, ANL-CT-76-36 (July 1976)

71. Howard, P.A., Lorenz, J.J., *CRBR Outlet Plenum Thermal Behavior during Transient Conditions*, ANL-CT-76-49

72. Schaefer, R.W., *A Probabilistic Method for Evaluating Reactivity Feedbacks and Its Application to EBR-II, Advances in Mathematics, Computations, and Reactor Physics*, April 28-May 1, 1991, Pittsburgh, Pa

73. Wade, D.C., Chang, Y.I., *The Integral Fast Reactor (IFR) Concept: Physics of Operation and Safety,* Proc. Int. Top. Mtg. on Advances in Reactor Physics, Mathematics and Computation, Paris, France, April 27030, 1987

74. Hill, D.J., *An Overview of the EBR-II PRA*, Proc. 1990 Int. Fast Reactor Safety Meeting, Snowbird, UT, August 12-16, 1990, Vo. IV, P. 33 American Nuclear Society (1990)

75. Mueller, C.J., Wade, D.C., *Probabilities of Successful Inherent Shutdown of Unprotected Accidents in Innovative Liquid Metal Reactors*, Proc. Int. Top. Mtg. of Next Generation Power Reactors, Seattle, WA, May 1-5, 1988, p. 704 American Nuclear Society (1988)

76. Iman, R.L., Helton, J.C., *A comparison of Uncertainty and Sensitivity Analysis Techniques for Computer Models*, NUREG/CR-3904, SAND84-1461, Sandia National Laboratories (1985)

77. Khatib-Rahbar, M., et al., *A probabilistic approach to Quantifying Uncertainties in the progression of Severe Accidents*, Nucle. Sci. Eng., 102, p. 219 (1989)

78. May, R.S., et al., *Probabilistic Methods for Evaluating Operational Transient Margins and Uncertainties*, Nucl. Sci. Eng., 103, p. 81 (1989)

79. Meneghetti, D., Kucera, D.A., *Delineations of Per, Power-to-Flow and Inlet-Temperature Feedbacks of EBR-II*, Ann. Nucl. Energy, Vol. 17, No.1, p. 19 (1990)

80. Chang, L.K., Mohr, D., *The Effect of Primary Pump Coast down Characteristics on Loss-of-Flow Transients Without Scram in EBR-II*, Nucl. Engrg. Des. 97. P 49 (1986)

81. Bergeonneau, Ph., et al., *Uncertainty Analysis on the Measurements and Calculation of Feedback Reactivity Effect in LMFBRs, Application of Super-Phenix-1 Startup Experiments*, 30th NEACRP Meeting Helsinki, Finland, Sept. 1987, NEACRP-A-833 (1987)

82. Meneghetti, D., Kucera, D.A., *Comparisons of PRD Components For Various EBR-II Configurations,* Ann. Nucl. Energy, 14, No. 9, p. 473 (1987)

83. Schaefer, R.W., *Critical Experiment Tests of Bowing and Expansion Reactivity Calculations for Liquid-Metal-Cooled Fast Reactors*, Nucl. Sci. Eng., 103, p. 196 (1989)

84. Beck, C.L., et al., *Sodium Void Reactivity in LMFBRS: A physics Assessment*, Proc. Top. Mtg., Advances in Reactor Physics and Core Thermal Hydraulics, Kiamesha Lake, NY, Sept. 22-24, 1982 NUREG/CP-0034, Vol. 1. P. 78 (1982)

85. Sackett, J.I., et al., *EBR-II Test Program*, Proc. 1990 Int. Fast Reactor Safety Meeting, Snowbird UT, August 12-16, 1990, Vol. III, p. 181, American Nuclear Society

86. Van Tuyle, G.J., et al., *Summary of Advanced LMR Evaluations – PRISM and SAFR*, NUREG/CR-5364, BNL-NUREG-52197, 1989

87. Baumeister, E.B., et al., *Inherent Safety Features and Licensing Plan of the SAFR Plant*, Proceedings of the International Conference on Fast Breeder Systems, 1, p. 3.4-1, Sept. 13-17, 1987

88. Berflund, R.C., et al., *Design of PRISM, an Inherently Safe, Economic, and Testable liquid Metal Fast Breeder Reactor Plant*, Proceedings of the ANS/ENS International Conference on Fast Breeder Systems, Volume 2, Pasco, WA, Sept. 13-17, 1987

89. Brown, N.W., et al., *Inherent Safety and Design Certification of PRISM*, Proceedings of the International Conference of Fast Breeder Systems, 1, p. 3.5-1, Sept. 13-17, 1987

90. Cheng, H.S., Van Tuyle, G.J., *A Simple Model for Radial Expansion Reactivity in LMRs*, Safety of Next Generation Power Reactors, Seattle, WA, May 1988

91. Gluekler, E.L., et al., *Safety Characteristics of a Small Modular Reactor*, Proceedings of the international Topical Meeting on Fast Reactor Safety, 1,pp. 69-74, 1985

92. Guppy, J.G., et al., *Super system Code (SSC Rev. 2) An Advanced Thermo-hydraulic Simulation Code for Transients in LMFBRs*, Brookhaven National Laboratory Report, NUREG/CR-3169, BNL-NUREG-51650, April, 1983

93. Hofman, G.L., et al., *Metallic Fuels Handbook*, Argonne National Laboratory Report, ANL-IFR-29, November 1985

94. Krueger, P.G., Van Tuyle, G.J., *Evaluation of Advanced Liquid Metal Reactor Passive Air Cooling Systems*, Transactions of the American Nuclear Society, Low Angeles Winter Meeting, November 1987, also BNL-NUREG-39957

95. Lancet, R.T., Marchaterre, J.F., *Inherent Safety of the SAFR Plant*, Proceedings of the int. Topical Meeting on Fast Reactor Safety, 1, pp. 43-50, 1985

96. Padilla, A., Hill, D.J., *Comparison of Reactivity Feedback Models for the FFTF Passive Safety Tests*, Proceedings of the Safety of Next Generation Power Reactors Conference, Seattle, WA, May 1988

97. Robinson, W.R., et al., *First TREAT overpower Tests on U-Pu-Zr- Fuel: M5 and M6,* Transactions of the American Nuclear Society, 55, Los Angeles, CA, November 1987

98. Slovik, G.C., et al., *Analysis of Postulated Unscrammed Loss-of-Flow in SAFR Using SSC*, Transactions of the American Nuclear Society, Vol. 57, Washington, DC, November 1988

99. Slovik, G.C., et al., *Evaluating Advanced LMR Reactivity Feedbacks Using SSC,* Proceedings of the 1988 ANS Topical Meeting on Safety of Next Generation Power Reactors, May 1-5, 1988, Seattle, WA

100. Van Tuyle, et al., *Comparing the Inherent Safety of the Modula LMRs and HTGRs and the PIUS Concept*, Proceedings of the 1988 ANS Topical Meeting on Safety of Next Generation Power Reactors, May 1-5. 1988, Seattle, WA

101. Van Tuyle, et al., Initial Pipe Break Analyses for Advanced LMR Concepts Using MINET, Transactions of the American Nuclear Society, Low Angeles Winter Meeting, November 1987, also BNL-NUREG-39956

102. L. E. Strawbridge and G. H. Clare, Exclusion of Core Disruptive Accidents from the Design Basis Accident Envelope in CRBRP, Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp.317-327, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

103. E. L. Gluekler, et al, Safety Characteristics of a Small Modular Reactor, Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp. 69-74, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

104. R. A. Doncals, et al, The Westinghouse Approach to an Inherently Safe Liquid Metal Reactor Design, Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp. 59-68, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

105. S. Kondo, Roles of Passive Safety Features in Fast Reactor Design and Operation in Japan, Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp. 115-122, American Nuclear Society, Snowbird, Utah, August 12-16, 1990

106. D. Broadley, et al, Passive Design Features of EFR, , Proc. Intl. Mtg. Fast Reactor Safety, Vol. 1, pp. 123-130, American Nuclear Society, Snowbird, Utah, August 12-16, 1990

107. T. R. Beaver, et al, Transient Testing of the FFTF for Decay Heat Removal by Natural Convection, Proc. LMFBR Safety Topical Meeting, Vol. II, pp. 525-534, European Nuclear Society, Lyon, France, July 19-23, 1982

108. B. W. Spencer, et al, Results of Recent Reactor-Material Tests on Dispersal of Oxide Fuel from a Disrupted Core, Proc. Intl. Mtg. Fast Reactor Safety, Vol. 2, pp. 877-882, American Nuclear Society, Knoxville, Tennessee, April 21-25, 1985

109. Y. Fukano, et al., CAFÉ Experiments on the Flow and Freezing of Metal Fuel and Cladding Melts (1), Test Conditions and Overview of the Results, FR09, Kyoto, Japan, December 2009.

110. A. E. Wright, et al., CAFÉ Experiments on the Flow and Freezing of Metal Fuel and Cladding Melts (2), Results, Analysis, and Applications, FR09, Kyoto, Japan, December 2009.

111. Y. Tobita, et al, Analytical Study on Elimination of Severe Recriticalities in Large Scale LMFBRs with Enhancement of Fuel Discharge, Nucl. Eng. Des., 238, pp. 57-65, 2008

112. J.E. Cahalan, R.A. Wigeland, G. Friedel, G. Kussmaul, J. Moreau, M. Perks, and P. Royl, "Performance of Metal and Oxide Fuels During Accidents in a Large Liquid Metal Cooled Reactors," Proceedings of the International Fast Reactor Safety Meeting, Snowbird, UT, Vol. IV, p. 73 (August 1990)

113. R.A. Wigeland, R.B. Turski, and P.A. Pizzica, "Impact of Reducing Sodium Void Worth on the Severe Accident Response of Metallic-Fueled Sodium-Cooled Reactors," Proceedings of the International Topical Meeting on Advanced Reactors Safety, Pittsburgh, PA, Vol. II, p. 1062 (April 1994)

114. R.A. Wigeland, R.B. Turski, and P.A. Pizzica, "Tradeoff on Sodium Void Worth and Reactivity Burnup Swing: Impacts on Balanced Safety Position in Metallic-Fueled Cores," International Topical Meeting on Sodium-Cooled Fast Reactor Safety, Obninsk, Russia (October 1994)

115. Wigeland, R., Cahalan, J., Mitigation of Severe Accident Consequences Using Inherent Safety Principles, FR09, Japan, December 2009

116. Pedersen, D., et al., Experimental and Analytical Studies of Passive Shutdown Heat Removal from Advanced LMRs, Argonne National Laboratory, Dec. 1988

117. Lahm, C.E., et al., EBR-II Driver Fuel Qualification for Loww-of-Flow and Loss-Heat-Sink Tests Without Scram, Nuclear Engineering and Design

118. Gopalakrishnan, A., Gillette, J.L., EBRFLOW – A Computer Program for Predicting the Coolant Flow Diestribution in the Experimental Breeder Reactor II, Nucl. Tech., Vol. 17, March 1973

119. Bishop, A.A., et al., Review of Pertinent Thermal-Hydraulic Data For LMFBR Core Natural Circulation Analysis, CRBRP, ASME/AICHE 19th National Heat Transfer Conference, Orlando, Florida, July 27-30, 1980

120. Alliston, W.H., et al., Clinch River Breeder Reactor Plant-LMFBR DEMO Plant Simulation Model (DEMO), Topical Report, CRBRP-ARD-005

121. George, T.L., et al., A Modified version of COBRA for Whole Core LMFBR Transient Analysis, Trans. Amer. Nucle. Soc., 32, pp. 531-532 (June 1979)

122. Miller, J.V., Coffield, R.D., Clinch River Breeder Reactor Plant, FORE-2M: A Modified Version of the FORE-II Computer Program for the Analysis of LMFBR Transients, Topical Report, CRBRP-ARD-0142, May, 1976.

123. Engel, F.C., et al., Laminar, Transition and Turbulent Parallel Flow Pressure Drop Across Wire-Wrap-Apaced Rod Bundles, Nucle. Sci. And Eng., 69, pp. 290-296 (1979)

124. Chiu, C., et al., Pressure Drop Measurements in LMFBR Wire-Wrapped Blanket Assemblies, COO-2245-42TR, July 1977

125. Hofmann, H., Experimental Investigations on Coolant Cross Mixing and Pressure Drop in Rod Bundles with Helical Spacers, EURFNR-1161, Dec. 1973

126. Chiu, C., et al., Subchannel and Bundle Friction Factors and Flow Split Parameters for Laminar Transition and Turbulant Longitudinal Flows in Wire-Wrap Spaced Hexagonal Arrays, ANS Topical Meeting on Reactor Thermal-Hydraulics, Oct. 1980

127. Pederson, D.R., et al., Experimental Investigation of the Hydrodynamic Entrance Length and Subchannel Velocity Profiles in a 91-Element Wire-Wrapped Subassembly, ANL/RAS 74-5, April 1974

128. Collingham, R.E., et al, 217-Pin Wire Wrapped Bundle Coolant Mixing Test, HEDL-TME 71-146, Nov. 1971

129. Ginsberg, T., Lorenz, J.J., Experimental Mixing Studies in Simulated Wire-Wrap Fuel Assemblies, International Meeting on Reactor Heat Transfer, pp. 1`3-38, Gesellschaft fur Kernforschung MbH, Karlsruhe, Germany, 1973

130. Chen, Y.B., et al., Velocity Measurements in Edge Channels of Wire-Wrapped LMFBR Fuel Assemblies, Trans. Amer. Nucle. Soc., 19, pp. 323-324

131. Lorenz, J.J., et al., Experimental Mixing Studies and Velocity Measurements with Simulated 91-Element LMFBR Fuel Assembly, ANL-CT-74-09, March 1974

132. Bartholet, T.G., et al., Clinch River Breeder Reactor Plant. 11:1 Scale Wire-Wrapped Rod Bundle Air Flow Test, Interior Subchannels, CRBRP-ARD-0108, January 1977

133. Freeman, D.D., SEFOR Experimental Results and Applications to LMFBRs, GEAP-13929, Jan. 1973

134. Coffield, R.D., Planchon, H.P., LMFBR Natural Circulation Verification Program (NCVP) Review of Experimental Facilities and Testing Recommendations, WARD-NC-3045-1, July 1977

135. Spencer, B.W., et al., Thermal-Hydraulic Development of a Small, Simplified, Proliferation-Resistant Reactor, ANL Report, Oct. 1999

136. Milllburg, M.L., et al., Natural Circulation Heat Transfer Testing with a Simulated Full-Scale LMFBR 217-Pin Electrically Heated Fuel Assembly, HEDL-TME-77-3 June 1977

137. Marklely, R.A., Engel, F.C., LMFBR Assembly Heat Transfer and Hydraulic Test Data Evaluation, International Working Group on Fast Reactors Specialists' Meeting on Thermodynamics of FBR Fuel Subassemblies Under Nominal and Non-Nominal Operating conditions, IWDFR/29, PP. 229-253, Feb. 1979

138. Engel, F.C., et al., Buoyancy Effects on Sodium Coolant Temperature Profiles Measured in an Electrically Heated Mock-Up of a 61-Rod Breeder Reactor Blanket Assembly, ASME Paper 78-WA/HT-25

139. Gillette, J.L., et al., Experimental Observations of Inter-Sub-Assembly Heat Transfer Effects During Natural Circulation, Trans. Amer. Nucl. Soc., 29, pp. 537-538 (November 1978)

140. Chang, L.K., Lee, M.J., Thermal and Structural Behavior of EBR-II Plant During Unprotected Loss-of-Flow Transients, 8th International Conference, Structural Mechanics in Reactor Technology, August 19-23, 1985 Brussels, Belgium

141. Forrester, R.J., et al., EBR-II High-Ramp Transients under Computer Control, American Nuclear Society Annual Meeting, June 12-17, 1983, Detroit, Michigan

142. Morris, E.E., et al., Analysis of the TREAT Loww-of-Flow Tests L6 and L7 Using SAS3D, ANL Report

143. Simms, R., et al., Loss-of-Flow Transient Reactor Test Facility Tests L6 and L7 with Irriadiated Liquid-Metal Fast Breeder Reactor Type Fuel, Nuclear Technology, Vol. 52, pp. 331-341 (March 1981)

144. De Volpi, A., et al., Fast-Neutron Hodoscope at TREAT: Data Processing, Analysis and Results, Nuclear Technology, Vol. 30, pp. 398-421 (Sep. 1976)

145. Bowers, C.H.., et al., Analysis of TREAT Tests L7 and L8 with SAS3D, LEVITATE, and PLUTO2, Specialists' Workshop on Predictive Analysis of Material Dynamics in LMFBR Safety Experiments, pp. 242-256, Los Alamos Scientific Laboratory, LA-7938-C (March 1979)

146. Simms, R., et al., TREAT Test L7 Simulating an LMFBR Loss-of-Flow Accident with FTR-Type Fuel, Argonne National Laboratory, ANL-80-112 (Nov. 1980)

147. Gruber, E.E., Randklev, E.H., Comparison of 'Fission Gas Effects in a Transient Overpower Test (HUT 5-7A) to FRAS3 Code Predictions, Proceedings of the International Meeting on Fast Reactor Safety Technology, pp. 2050-2059, Seattle, WA (August 1979)

148. Simms, R., An Evaluation of Fuel Motion in Recent TREAT Experiments with Liquid-Metal Fast Breeder Reactor Fuel, Nuclear Technology, Vol. 50, pp. 257-266 (Oct. 1980)

149. Bauer, T.H., et al. Behavior of Metallic Fuel in TREAT Transient Overpower Tests, Proc. Int'l Topical Meeting on Safety of Next Generation Power Reactors, May, 1988, Seattle, WA, p. 857

150. Bauer, T.H., et al., Cladding Failure Margins for Metallic Fuel in the Integral Fast Reactor, Invited Lecture in the SMIRT-9 Conference, August 1987

151. Wright, A.E., et al., Fast Reactor Safety Testing in TREAT in the 1980s, 1990 International Fast Reactor Safety Meeting, August, 12-16, 1990, Snowbird, Utah

152. Pitner, A.L., et al., Axial Relocation of Oxide Fuel During Overpower Transients, Proc. Int'l Topical Meeting on Safety of Next Generation Power Reactors, May 1988, Seattle, WA

153. Robinson, W.R., et al., First TREAT Transient Overpower Tests on U-Pu-Zr Fuel: M5 and M6, Argonne National Laboratory

154. Robinson, W.R., et al., Integral Fast Reactor Safety Tests M2 and M3 in TREAT, Trans. Am. Nuclear Soc., 50, 352 (Nov. 1985)

155. Doerner, R.C., et al., Fuel Motions in the Multi-Pin PFR/TREAT Test Series, ANL Report

156. Holland, J.W., et al., Post-Test Examination Results of Recent TREAT Tests on Metal Fuel, International Conference on Reliable Fuels for Liquid Metal Reactors, American Nuclear Society, Material Science and Technology Division, Tucson, AZ, Sept. 7-11, 1986

157. Deitrich, L.W., et al., A Review of Experiments and Results From the Transient Reactor Test (TREAT) Facility, ANS Winter Meeting, Washington, DC, Nov. 15-19, 1998

158. Klickman, A.E., et al., Review of Recent ANL Safety Experiments in SLSF and TREAT, ANL Report

159. Kraft, T.E., et al., Simulation of an Unprotected Loss-of-Flow Accident with a 37-pin Bundle in the Sodium Loop Safety Facility, Proc., International Meeting of Fast Reactor Safety Technology, Seattle, WA (1979)

160. Kraft, T.E., et al., The Response of Low-Burnup Fuel to a Loss-of-Flow Transient in an LMFBR, Proc. Topical Meeting on Reactor Safety of Fuel Behavior, Sun Valley (1981), p. 279

161. Spencer, B.W., et al., Summary and Evaluation of R-Series Loss-of-Flow Safety Tests in TREAT,. Proc. International Meeting on Fast Reactor Safety and Related Physics, Chicago (1976)

162. Rothman, A.B., et al., TREAT Experiments with Irradiated Fuel Simulating Hypothertical Loss-of-Flow Accidents in Large LMFBRs, Proc. International Meeting on Fast Reactor Safety Technology, Seattle (1979) p. 924

163. Tylka, J.P., et al., PFR/TREAT CO1 and LO1 Experiments, Proc. International Topical Meeting on Liquid Metal Fast Breeder Reactor Safety and Related Design and Operational Aspects, Lyon (1982)

164. Deitrich, L.W., Experiments on Transient Fuel Failure Mechanisms – Selected ANL Programs, International Working Group on Fast Reactor Specialists' Meeting on Fuel Failure Mechanisms, Seattle, WA, May 11-16, 1975

165. Rose, S.D., Carbajo, J.J., Core Coolability Limits in LMFBR Loss-of-Heat-Sink Accidents, ANS Anneal Meeting, June 3-8, 1984, New Orleans, Louisiana

166. Strawbridge, L.E., CRBRP Structural and Thermal Margin Beyond the Design Base, ANS International Meeting on Fast Reactor Safety Technology, Seattle, WA, Aug 19-23, 1979

167. Marchaterre, J.F., Overview of Core Disruptive Accidents, International Seminar on Containment of Fast Breeder Reactors, San Francisco, CA, Aug. 22-23, 1977

168. Tentner, A.M., et al., SAS4A A Computer Model for the Analysis of Hypothetical Core Disruptive Accidents in Liquid Metal Reactors, Eastern Computer Simulation Conference, Orland, FL, April 6-9, 1987

169. Marchaterre, J.F., et al., Structural and Containment Response to LMFBR Accidents, International Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, Oct. 16, 1978

170. Dickerman, C.E., et al., Summary of TREAT Experiments on Oxide Core-Disruptive Accidents, ANL-79-13, Feb. 1979

171. Sackett, J.I, "EBR-II Test Experience as the IFR Prototype", Proceedings of Global 2009, Paris, France, Sept. 6-11, 2009

9.4 Source Term References

1. H-J. Allelein, A. Auvinen, J. Ball, S. Guntay, L.E. Herranz, A. Hidaka, A.V. Jones, M. Kissane, D.A. Powers, G. Weber, "State-of-the-Art Report on Nuclear Aerosols", NEA/CSNI/R(2009)5, Nuclear Energy Agency, November 2009.

2. D.J. Alpert, D.I. Chanin, and L.T. Ritchie, "Relative Importance of individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions," NUREG/CR-4467, Sandia National Laboratories, Albuquerque, NM, March 1986.

3. D.R. Armstrong, G.T. Goldfuss, and R.H. Gebner, "Explosive Interaction of Molten UO_2 and Liquid Sodium", ANL-76-24, Argonne, National Laboratory, 1976.

4. M. Berlin, E. de Montaignac, J. Dugresne, and G. Geisse, Evaluation of the Sodium Retention Factors for Fission Products and Fuel, Rapport DSN # 599[e], Conference internationale sur la surete des reacteurs rapides a metal liquide et ses aspects conception et fonctionnement, Lyon France, July 19-23, 1982.

5. R.J. Budnitz, G. Apostolakis, D.M Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell and P.A. Morris, "Recommendations for Probabilistic Seismic Hazard Analysis – Guidance on Uncertainty and Use of Experts", NUREG/CR-6372, 1997.

6. T.Y. Chu, A.G. Beattie, W.D. Drotning, and D.A. Powers, "Medium-scale Melt-Sodium Fragmentation Experiments", International Meeting on Fast Reactor Safety Technology, Volume I, pp. 742-756, 1979.

7. R. Clift, J.R. Grace, and M.E. Weber, "Bubbles, Drops and Particles," Academic Press, 1978.

8. J.J. DiNunno *et al*., "Calculation of Distance Factors for Power and Test Reactor Sites", Technical Information Document TID-14844, U.S. Atomic Energy Commission, 1962.

9. DOE, "Nuclear Energy Research and Development Roadmap – Report to Congress", U.S. Department of Energy, Washington, D.C., April 2010.

10. G.L. Fiorini *et al*., "The Collaborative Project on European Sodium Fast Reactor (CP ESFR), European Commission 7th Framework Programme," FISA 2009.

11. R.O. Gauntt, "Synthesis of VERCORS and Phebus Data in Severe Accident Codes and Applications", SAND2010-1633, Sandia National Laboratories, April, 2010.

12. R.O. Gauntt, R.K. Cole, C.M. Erickson, R.G. Gido, R.D. Gasser, S.B. Rodriquez, and M.F. Young, MELCOR Computer Code Manuals, NUREG/CR-6119, Rev.2, SAND2000-2417, prepared for NRC by Sandia National Laboratories, Albuquerque, NM, December 2000.

13. F. Gelbard, "MAEROS User's Manual", NUREG/CR-1391, 1982.

14. J.A. Gieseke, *et al*., "Source Term Code Package: User's Guide," NUREG/CR-4587 (BMI-2138), prepared for NRC by Battelle Memorial Laboratory, Columbus, Ohio, July 1986.

15. A-G Guézennec, J-C. Huber, F. Patisson, P. Sessiecq, J-P. Birat, D. Ablitzer, "Dust formation in Electric Arc Furnace : Birth of the particles", Powder Technology, *157* (2005) 2-11.

16. R.E. Henry, C.Y. Palik, and M.G. Plys, "MAAP4 – Modular Accident Analysis Program for LWR Power Plant – Code Structure and Theory, Computer Code Manual", Fauske and Associates, Inc., May 1994.

17. S. Jordan, "Release of Fission Products from Contaminated Sodium Fires", Proceedings International Meeting on Fast Reactor Safety and Related Physics, Chicago, 1976.

18. S. Jordan and Y. Ozawa, "Fuel Particle and Fission Product Release from LMFBR core Catcher", Proceedings International Meeting on Fast Reactor Safety and Related Physics, Chicago, 1976.

19. T.K. Kim and W.S. Yang, "Preliminary Estimation of Isotopic Inventories of 2000 MWt ABR", Argonne National Laboratory, 2006.

20. M. Koch, U. Brockmeier, and H. Unger, Erstellung Eines Mechansitischen Transportmodells für Spaltproduckte, Institute for Energy Technology, Bochum university of Bochum, April, 1990.

21. H. Lamb, **Hydrodynamics**, Sixth edition, Cambridge University Press, 1994.

22. J.Y. Lee, "Implementation of Reactor Accident Alternative Source Term," Trans. ANS, *94* (2006) 159.

23. D.A. Lochbaum, "An Assessment of Potential Pathways for Releases of Gaseous Radioactivity Following Fuel Damage During Run14 at the Sodium Reactor Experiment", Prepared for the Santa Susana Field Laboratory Advisory Pane, October, 2006

24. K.K. Murata, *et al*., "CONTAIN LMR/1B-Mod1, A Computer Code for Containment Analysis of Accidents in Liquid-Metal Cooled Nuclear Reactors", SAND91-1490, Sandia National Laboratories, Albuquerque, NM, 1993.

25. E.M. Page, "The Fuel Melting Incident" Chapter 11 in "FERMI-I New Age for Nuclear Power", E.P. Alexanderson and H.A. Wagner, editors, American Nuclear Society, 1979.

26. H. Sauter and W. Schütz, "Aerosol and Activity Releases from a Contaminated Sodium Pool into an Inert Gas Atmosphere", KfK 3504, Kernforschungszentrum Karlsruhe, Germany, 1983.

27. W. Schütz, "UO_2 and Fission Product Release from Sodium Pools", KfK3010, Kernforschungszentrum Karlsruhe, Germany, 1980.

28. L. Soffer, S.B. Burson, C.M. Ferrell, R.Y. Lee, and J.N. Ridgely, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, U.S. Nuclear Regulatory Commission, February 1995.

29. M. Tomaides and K.T. Whitby, "Generation of Aerosols by Bursting of Single Bubbles", in Fine Particles, B.H.Y. Liu, ed., Academic Press, 1976.

30. E.T. Turkdogen, P. Grieveson, and L.S. Darken, J. Metals, (July 1962) 521.

31. U.S. Nuclear Regulatory Commission, "Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors", Regulatory Guide 1.3, Revision 2, June 1974a.

32. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", Regulatory Guide 1.4, Revision 2, June 1974b.

33. U.S. Nuclear Regulatory Commission, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, June 1981.

34. U.S. Nuclear Regulatory Commission, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, December 1990.

35. U.S. Nuclear Regulatory Commission, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Regulatory Guide 1.183, July 2000.

9.5 Codes and Models References

General References

1. W. B. Oberkampf, M. Pilch, T. G. Trucano, Predictive Capability Maturity Model for Computational Modeling and Simulation, SAND2007-5948, Sandia National Laboratories, Albuquerque, NM, 2007.
2. D.A. Powers, B. Clément, R. Denning, S. Ohno, R. Zeyen, “Advanced Sodium Fast Reactor Accident Source Terms: Research Needs,” SAND-2010-5506, Sandia National Laboratories, Albuquerque, NM, 2010, September 2010.
3. J. Sackett, R. Wigeland, R. Bari, R. Budnitz, J. Cahalan, C. Grandy, D. Wade, M. Corradini, R. Denning, G. Flanagan, S. Wright, “Advanced Sodium Fast Reactor Accident Initiators/Sequences Technology Gap Analysis”, FCRD-REAC-2010-000126, U.S. Department of energy Fuel Cycle Research and Development Program, March 2010.
4. M. Corradini et. al., “Advanced Burner Reactor Sodium Technology Gap Analysis,” FCR&D-REAC-2010-000034, U.S. Department of energy Fuel Cycle Research and Development Program, February 2010

Computer Code References (In Alphabetical order)

ANL Nuclear Engineering Division Software

5. http://www.ne.anl.gov/codes/index.html

ANSYS and ANSYS-AUTODYN

6. www.ansys.com

APPLOHS

JAEA code. No public references found

AQUA

7. I. Maekawa, “Numerical Diffusion in Single-Phase Multi-Dimensional Thermal-Hydraulic Analysis,” Nucl. Eng. Des., 120, 323 (1990).

AQUA-SF

8. T. Takata, A. Yamaguchi, I. Maekawa, "Numerical Investigation of Multi-dimensional Characteristics in Sodium Combustion," Nucl. Eng. Design, 220, pp.37-50 (2003).

ASFRE

9. H. Ohshima, H. Ninokata, “Thermal-Hydraulic Analysis of Fast Reactor Fuel Subassembly with Porous Blockages,” Proc. 4th Int. Seminar on Subchannel Analysis (ISSCA-4), pp323-333, Tokyo (1997).

BAMBOO

10. T. Uwaba, M. Ito, S. Ukai, M. Pelletier, "Development of a FBR Fuel Bundle-duct Interaction Analysis Code-BAMBOO: Analysis Model and Verification by Phenix High Burn-up Fuel Subassemblies," J. Nucl. Sci. Tech., Vol.42, pp.608-617, 2005.

BISHOP

11. Y. Okano, A. Yamaguchi, "Direct Numerical Simulation of a Combustion Experiment of a Free-falling Liquid Sodium Droplet," Proc. 8th Int. Conf. on Nucl. Engineering (ICONE 8), ICONE-8210, Baltimore (2000).

CAST3M

12. http://www-cast3m.cea.fr/cast3m/xmlpage.do?name=presentation
13. http://www-cast3m.cea.fr/cast3m/html/ManuelCastemEnsta_en/castem3.html

REPSO/CALHYPSO

14. T. Desmas, N. Kong, J.P. Maupre, P. Schindler, D. Blanc "Hydrogen detection systems leak response codes", IAEA Specialists' meeting on steam generator failure and failure propagation experience, Aix-en-provence, France, Sep. 26-28, 1990.

CATHARE V2.5

15. G. Geffraye, O. Antoni, M. Farvacque, D. Kadri, G. Lavialle, B. Rameau, A. Ruby, CATHARE 2 V2.5_2: A single version for various applications, Nuclear Engineering and Design, In Press, Corrected Proof, Available online 22 October 2010, ISSN 0029-5493, DOI: 10.1016/j.nucengdes.2010.09.019.

CONTAIN LMR/1B-Mod1

16. K.K. Murata, et al., “CONTAIN LMR/1B-Mod1, A Computer Code for Containment Analysis of Accidents in Liquid-Metal Cooled Nuclear Reactors”, SAND91-1490, Sandia National Laboratories, Albuquerque, NM, 1993.

CONTAIN-LMR-J

17. O. Miyake, H. Seino, T. Takai, H. Hara, “Development of CONTAIN Code for FBR Severe Accident Analysis,” Proc. Vol. IV, International Conference on Design and Safety of Advanced Nuclear Power Plants (ANP'92), p41.4-1 - 41.4-5, Tokyo (1992).

CONTAIN-LMR-K

KAERI version of CONTAIN-LMR code. No public references found

DEBIDO

AREVA code. No public references found

DIF3D

18. R.D. Lawrence, "Progress in Nodal Methods for the Solution of the Neutron Diffusion and Transport Equations," Prog. Nucl. Energy, 17, 3, 271 (1986).
19. K. L. Derstine, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," Argonne-82-64, Argonne National Laboratory (April 1984).
20. R. D. Lawrence, "The DIF3D Nodal Neutronics Option for Two- and Three-Dimensional Diffusion Theory Calculations in Hexagonal Geometry," Argonne-83-1 Argonne National Laboratory (March 1983).
21. http://www.ne.anl.gov/codes/dif3d

ERANOS2

22. G. Rimpault et al. 'The ERANOS Code and Data System for Fast Reactor Neutronic Analyses', Proc. Int. Conf. PHYSOR 2002, Seoul, Korea, October 7-10, 2002
23. http://www.nea.fr/tools/abstract/detail/nea-1683
24. J. Tommasi et al., “Status of the ERANOS-2 code system validation for Sodium Fast Reactor Applications, PHYSOR 2010, Pittsburgh, PE, USA, May 9-14, 2010
25. G. Rimpault, “Algorithmic Features of the ECCO Cell Code for Treating Heterogeneous Fast Reactor Assemblies”, ICAPP 2006, Reno, Nevada, USA, June 4-8, 2006

EUROPLEXUS

26. http://europlexus.jrc.ec.europa.eu/
27. http://wwww.samtech.com/en/pss.php?ID=32&W=products

FEUMIX

28. J. Dufresne et al., “Sodium Concrete Interaction”, LMFBR Topical Meeting, Lyon, France, 1982

FINAS

29. FINAS version 13.0 Users' Manual, PNC TN9520 95-014 (1995) [in Japanese]

FLICA-4

30. I. Toumi, A. Bergeron, D. Gallo, E. Royer, D. Caruge, FLICA-4: a three-dimensional two-phase flow computer code with advanced numerical methods for nuclear applications, Nuclear Engineering and Design, Volume 200, Issues 1-2, August 2000, Pages 139-155, ISSN 0029-5493, DOI: 10.1016/S0029-5493(99)00332-5.

GALILEE

31. Coste-Delclaux, M. (2008), "GALILEE: A Nuclear Data Processing System for Transport, Depletion and Shielding Codes", Proc. of Int. Conf. on the Physics of Reactors: Nuclear Power: A Sustainable Resource (PHYSOR2008), Interlaken, Switzerland, 14-19 September. 2008

GERMINAL-V1

32. L. Roche and M. Pelletier, "Modelling of the Thermo-Mechanical and Physical Processes in FR Fuel Pins Using the GERMINAL Code," *Proc. Int. Symp. MOX Fuel Cycle Technologies for Medium and Long Deployment*, IAEA-SM-3580 25, Vienna, Austria, May 17–20, 1999, International Atomic Energy Agency ~1999

GVNOV

AREVA code. No public references found

MACCS

33. Rollstin, J.A., D. I. Chanin, and H.-N. Jow (1990), MELCOR Accident Consequence Code System (MACCS), Programmer's Reference Manual, NUREG/CR-4691, SAND86-1562, Sandia National Laboratories, Albuquerque, NM.
34. Chanin, D., J. Rollstin, J. Foster, and L. Miller (1993), MACCS Version 1.5.11.1: A Maintenance Release of the Code, NUREG/CR-6059, SAND92-2146, Sandia National Laboratories, Albuquerque, NM.

MACCS2

35. Chanin, D., and K.L. Young (1997), Code Manual for MACCS2 Volume 1, Users Guide, NUREG/CR-6613, SAND97-0594, Sandia National Laboratories, Albuquerque, NM.

MARS-LMR

36. Kwi-Seok Ha, Hae-Yong Jeong, Chungho Cho, Young-Min Kwon, Yong-Bum Lee, and Dohee Hahn, "Simulation of the EBR-II Loss-of-Flow tests using the MARS code," Nucl. Technol., Vol. 169, No. 2, pp. 134-142, 2010.

MATRA-LMR/FB

37. Hae-Yong Jeong, Kwi-Seok Ha, Won-Pyo Chang, Young-Min Kwon, and Yong-Bum Lee, "Modeling of flow Blockage in a Liquid Metal-cooled Reactor Subassembly with a Subchannel Analysis Code," Nucl. Technol., Vol. 149, No. 1, pp. 71-87, 2005.

MC2-2

38. http://www.ne.anl.gov/codes/mc2-2/
39. L. C. Leal, C. G. Stenberg, and B. R. Chandler, "Completion of the Conversion of the MC22/ SDX Codes from IBM to SUN (#3)," Argonne National Laboratory Internal Memorandum, May 27, 1993.
40. H. Henryson, II, B. J. Toppel, C. G. Stenberg, " $MC^2$2: A Code to Calculate Fast Neutron Spectra and Multigroup Cross Sections," Argonne-8144, June 1976.

MCNP

41. http://mcnp-green.lanl.gov/

42. F. B. Brown, et al. "MCNP5-1.51 Release Notes," LA-UR-09-00384, Los Alamos National Laboratory, 2009

MECTUB

43. F. Baque "MECTUB: A thermomechanical model for overheating studies of LMFBR Steam Generator tubes during a sodium/water reaction", IAEA Specialists' meeting on steam generator failure and failure propagation experience, Aix-en-provence, France, Sep. 26-28, 1990.

MELTSPREAD

44. M. T. Farmer, J. J. Sienicki, B. W. Spencer and C. C. Chu, "Status of the MELTSPREAD-1 Computer Code for the Analysis of Transient Spreading of Core Debris Melts," 2nd CSNI Specialist Meeting on Core Debris-Concrete Interactions, Karlsruhe, Germany, April 1-3, 1992.
45. M. T. Farmer, “Melt Spreading Code Assessment, Modifications, and Applications to the EPR Core Catcher Design,” ANL-09/10, Nuclear Engineering Division, Argonne National Laboratory, March 2009.

MELT-III

46. A.E. Walter et al., MELT-III: A neutronics, thermal-hydraulics computer program for fast reactor safety analysis, Vol. 1, HEDL-TME 74-47, Hanford Engineering Development Lab., December, 1974.

NACOM

47. S. S. Tsai, “The NACOM Code for Analysis of Postulated Sodium Spray Fires in LMFBRs”, NUREG/CR-1405 , BNL-NUREG-51180, 1980. (Note: this report is available online at the following link http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr1405/#pub-info)

NERGAL

48. K. Ito, T. Kunugi, H. Ohshima, T. Kawamura, “Formulations and Validations of a High-precision Volume-of-fluid Algorithm on Non-orthogonal Meshes for Numerical Simulations of Gas Entrainment Phenomena”, Journal of Nuclear Science and Technology, Vol. 46, No. 4, p. 366–373 (2009).

ORIGEN-2

49. http://www-rsicc.ornl.gov/codes/ccc/ccc3/ccc-371.html
50. ORIGEN-2, Isotope Generation and Depletion Code, ORNL TM-7175, Oak Ridge National Laboratory, Oak Ridge, TN. July 1980.
51. I. C. Gauld and B. D. Murphy, “Updates to the ORIGEN-2 Data Libraries Using ENDF/B-VI, FENDL-2.0, and EAF-99 Data,” ORNL/TM-2003/118, Oak Ridge National Laboratory, Oak Ridge, TN, 2004.

PERKY

52. Susumu Iijima et. al., "Calculation Program for Fast Reactor Design (Multi-dimensional Perturbation Theory Code based on Diffusion Approximation :PERGY)," JAERI-M 6993, Feb. 1977.

PROPANA

53. G. Thomine, JP. Maupre, FN Remy, Ph. Hobbes “Modelling of water leakage effects in fast breeder reactor steam generator”, Proc. 13th Int. Conf. on Structural Mechanics in reactor Technology (SMiRT 13), Porto Alegre, Brazil, August 13-18, 1995.

PULSAR

IRSN code. No public references found

PYROS-1

CEA code. No public references found

REACNOV

AREVA code. No public references found

REBUS-3

53. B. J. Toppel, "The Fuel Cycle Analysis Capability REBUS-3," Argonne-83-2 Argonne National Laboratory (March 1983).

55. http://www.ne.anl.gov/codes/rebus/

RELAP5

56. USNRC, RELAP5/MOD3.3 Code Manual, vol. 1–8, Information Systems Laboratories, Rockville, Maryland, Idaho Falls, Idaho, USA, prepared for USNRC, 2006.

57. The RELAP5 Development Team, RELAP5-3D© Code Manuals Volume 1-5, INEEL-EXT-98-00834 Revision 2.4, Idaho National Engineering Laboratory, June 2005 (available online at http://www.inl.gov/relap5/)

RESSORT

58. JC Malet et al., "Sodium Concrete Interaction Experimental Studies and Modeling", 4th International Conference (LIMET) 1988, Avignon, France.

SAS4A/SASSYS-1

59. The SAS4A/SASSYS-1 LMR Analysis Code System, Volumes 1–5, ANL-FRA-1996-3, Argonne National Laboratory, August 1996.

60. J. E. Cahalan et al., "Advanced LMR Safety Analysis Capabilities in the SASSYS-1 and SAS4A Computer Codes," *Proceedings of the International Topical Meeting on Advanced Reactors Safety*, Pittsburgh, PA, April 17–21, American Nuclear Society, 1994.

61. A. M. Tentner, et al., "The SAS4A LMFBR Whole Core Accident Analysis Code," *Proc. International Meeting on Fast Reactor Safety*, pp 989-998, Knoxville, TN (April 1985).

62. K. J. Miles, et al., "DEFORM-4: fuel pin characterization and transient response in the SAS4A accident analysis code system," *Proc. International Mtg. on Fast Reactor Safety*, Vol.2, p51, Guernsey, England (May 1986).

63. A. M. Tentner, et al., "Fuel relocation modeling in the SAS4A accident analysis code system," *Proc. International Mtg. on Fast Reactor Safety*, Vol.2, p85, Guernsey, England (May 1986).

64. P. L. Garner et al., "Development of a Graphical User Interface Allowing Use of the SASSYS-1 LMR Systems Analysis Code as an EBR-II Interactive Simulator," *Proceedings of International Topical Meeting on Advanced Reactors Safety*, Pittsburgh, PA, April 17-21, American Nuclear Society, 1994.

65. J. E. Cahalan and T. Wei, "Modeling Developments for the SAS4A and SASSYS Computer Codes," *Proceedings of the International Fast Reactor Safety Meeting*, Snowbird, UT, August 12-16, American Nuclear Society, 1990.

66. F. E. Dunn, "Decay Heat Calculations for Transient Analysis," *Trans. Am. Nucl. Soc.*, **60**, 633, 1989.

67. J. P. Herzog, "SASSYS Validation with the EBR II Shutdown Heat Removal Tests," *Trans. Am. Nucl. Soc.*, **60**, 730, 1989.

SERAPHIM

68. "T. Takata, A. Yamaguchi, A. Uchibori and H. Ohshima, "Computational Methodology of Sodium-Water Reaction Phenomenon in Steam Generator of Sodium-Cooled Fast Reactor," J. Nucl. Sci. Tech., Vol.46, pp.613-623, 2009."

SE2

69. W. S. Yang and A. M. Yacout, "Assessment of the SE2-ANL Code using EBR-II Temperature measurements", ANL/RA/CP--83070; CONF-9509042, 7th International Meeting on Nuclear Reactor Thermal Hydraulics, Saratoga Springs, NY, Sept. 10-15, 1995 (http://www.osti.gov/bridge/product.biblio.jsp?osti_id=78595)

70. http://www.ne.anl.gov/codes/se2anl/

SIMMER-III

71. H. Yamano, Y. Tobita, S. Fujita and W. Maschek, "First 3-D Calculation of Core Disruptive Accident in a Large Scale Sodium-Cooled Fast Reactor," Annals of Nuclear Energy, Vol. 36, No. 3, pp. 337-343 (Apr. 2009).

72. W. Maschek, A. Rineiski, M. Flad, P. Liu, X.N. Chen, Y. Tobita, H. Yamano, T. Suzuki, S. Fujita, K. Kamiyama, S. Pigny, T. Cadiou, K. Morita and G. Bandini, "SIMMER Safety Code System and its Validation Efforts for Fast Reactor Application," Proc. 2008 International Conference on Reactor Physics (PHYSOR'08), Interlaken, Switzerland, FP056 (Sep. 14-19, 2008).

73. H. Yamano, Y. Tobita, S. Fujita, T. Suzuki, K. Kamiyama,, K. Morita, W. Maschek and S. Pigny, "SIMMER-III: A Coupled Neutronics-Thermohydraulics Computer Code for Safety Analysis," Proc. 15th International Conference on Nuclear Engineering (ICONE15), Nagoya, Japan, ICONE15-10462 (April 22-26, 2007).

74. Y. Tobita, Sa. Kondo, H. Yamano, K. Morita, W. Maschek, P. Coste and T. Cadiou, "The Development of SIMMER-III, An Advanced Computer Program for LMFR Safety Analysis and Its Application to Sodium Experiments," Nuclear Technology, Vol. 153, No. 3, pp. 245-255 (March 2006).

75. H. Yamano, S. Fujita, Y. Tobita, K. Kamiyama, Sa. Kondo, K. Morita, E. A. Fischer, D. J. Brear, N. Shirakawa, X. Cao, M. Sugaya, M. Mizuno, S. Hosono, T. Kondo, W. Maschek, E. Kiefhaber, G. Buckel, A. Rineiski, M. Flad, T. Suzuki, P. Coste, S. Pigny, J. Louvet and T. Cadiou "SIMMER-III: A Computer Program for LMFR Core Disruptive Accident Analysis, Version 3.A Model Summary and Program Description," JNC TN9400 2003-071 (August 2003).

SIMMER-IV

76. W. Maschek, A. Rineiski, T. Suzuki, S. Wang, Mg. Mori, E. Wiegner, D. Wilhelm, F. Kretzschmar, Y. Tobita, H. Yamano, S. Fujita, P. Coste, S. Pigny, A. Henriques, T. Cadiou, K. Morita, and G. Bandini, " SIMMER-III and SIMMER-IV Safety Code Development for Reactors with Transmutation Capability," Mathematics and Computation, Supercomputing, Reactor Physics and Biological Applications, Palais de Papes, Avignon, France, September 12-15, 2005, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2005)

77. H. Yamano, S. Fujita, Y. Tobita, K. Kamiyama et al., SIMMER-IV: A Three-Dimensional Computer Program for LMFR Core Disruptive Accident Analysis, Japan Nuclear Cycle Development Institute, JNC TN9400 2003-070, 2003.

SORBET

78. JC Malet et al., "Sodium Concrete Interaction Experimental Studies and Modeling", 4th International Conference (LIMET) 1988, Avignon, France

SPHINCS

79. A. Yamaguchi, Y. Tajima, "Validation Study of Computer Code SPHINCS for Sodium Fire Safety Evaluation of Fast Reactor," Nucl. Eng. Design, 219, pp.19-34 (2003).

SPIKE

80. J. H. Park, J. H. Choi, T. J. Kim, K. C. Jeong, "Development of the SPIKE Code for Analysis of the Sodium Water Reaction," KAERI/TR-1123/98, Korea Atomic Energy Research Institute, 1998.

SPIRAL

81. H. Ohshimma, Y. Imai, "Validation Study of Thermal-Hydraulic Analysis Program "SPIRAL" for Fuel Pin Bunfle of Sodium-cooled Fast Reactor," 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, 423, 2005.

SSC-K

82. W.P. Chang et al., “Model Development for analysis of the Korea Advanced Liquid Metal Reactor,” Nucl. Engrg. Des., Vol. 217, No. 1-2, pp. 63-80, 2002.

Super-COPD

83. F. Yamada, H. Ohira, “Numerical Simulation of Monju Plant Dynamics by Super-COPD using Previous Startup Tests Data”, Proc. ASME 2010 3rd Joint US-European Fluids Engineering Summer Meeting and 8th International conference on Nanochannels, Microchannels, and Minichannels, FEDSM-ICNMM2010-30287, Montreal (2010).
84. Nakai, S., et al., “Development of Module Integrated Plant Dynamics Analysis Code -Development of Super-COPD Code-,” PNC Technical review No. 68-3 (1988) (in Japanese)."

SUPERENERGY-2

85. K.L. Basehore and N.E. Todreas, “SUPERENERGY-2: A Multiassembly Steady-State Computer Code for LMFBR Core Thermal-Hydraulic Analysis,” PNL-3379, Pacific Northwest Laboratory (August 1980)

SWACS

86. H. Hamada, M. Suzuki, Y. Himeno, "Modification of the Large Leak Sodium-Water Reaction Analysis Code, SWACS/REG4, for Large Steam Generators Having Non-Cover Gas Space (User's Manual)", PNC Technical report, PNC TN9520 89-016 (in Japanese)

SWAMM-II

87. Y.W. Shin, A.H. Wiedermann, T.V. Eichler, C.K. Youngdahl, and C.E. Ockert, "An Analytical Model for Dynamics of a Sodium/Water Reaction Bubble in an LMFBR Steam Generator and the Coupled Response of the Intermediate Heat Transport System," Nucl. Engr. Design, 106, 221-230 (1988)
88. C.. Gerardi, C. Youngdahl, and C. Grandy, "SWAAM-MF: More than a Sodium-Water Reaction Code," 2011 American Nuclear Society Annual Meeting, June 26-30 (2011; submitted)

TACT

Under development at JAEA. No references available at present time.

TRIO-U

89. Ulrich Bieder and Estelle Graffard, Qualification of the CFD code Trio_U for full scale reactor applications, Nuclear Engineering and Design, Volume 238, Issue 3, Benchmarking of CFD Codes for Application to Nuclear Reactor Safety, March 2008, Pages 671-679, ISSN 0029-5493, DOI: 10.1016/j.nucengdes.2007.02.040.
90. Thomas Hohne, Soren Kliem, Ulrich Bieder, Modeling of a buoyancy-driven flow experiment at the ROCOM test facility using the CFD codes CFX-5 and Trio_U, Nuclear Engineering and Design, Volume 236, Issue 12, June 2006, Pages 1309-1325, ISSN 0029-5493, DOI: 10.1016/j.nucengdes.2005.12.005.
91. Höhne, TH;, Kliem, S., Bieder, U. : « Numerical modelling of a buoyancy-driven flow experiment at the ROCOM test facility using the CFD codes CFX and Trio_U », Nuclear Engineering and Design, 236 (12), pp 1309-1325, 2006.

TRIPOLI4

92. C.M. Diop, O. Petit, E. Dumonteil, F.X. Hugot, Y.K. Lee, A. Mazzolo and J.C. Trama, “TRIPOLI-4: A 3D Continuous-Energy Monte Carlo Transport Code” PHYTRA1: First International Conference on Physics and Technology of Reactors and Applications. Marrakech (Morocco), March 14-16, 2007, GMTR (2007) (available online at http://citeseerx.ist.psu.edu/viewdoc/summary?doi=10.1.1.111.8175)
93. J. P. Both, Y.K. Lee, A. Mazzolo, O. Petit, Y. Peneliau, B. Roesslinger, M. Soldevila: TRIPOLI-4 - A Three Dimensional Polykinetic Particle Transport Monte Carlo Code, SNA'2003, Paris Sept. 2003.
94. J.P. Both, H. Derriennic, B. Morillon and J.C. Nimal, Proceedings of the 8th International Conference on Radiation Shielding, 1, 373,1994.

VARI3D

95. http://www.ne.anl.gov/codes/vari3d

96. VARI3D: C. H. Adams, "Specifications for VARI3D - A Multidimensional Reactor Design Sensitivity Code," FRA-TM-74, Argonne National Laboratory, 1975.

VIBUL

97. A. Yamaguchi, A. Hashimoto, “A Computational Model for Dissolved Gas and Bubble Behavior in the Primary Coolant System of Sodium-Cooled Fast Reactor,” 11th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, 477, 2005.

VENUS-II

98. J. F. JACKSON and R. B. NICHOLSON, “VENUS-II : An LMFBR Disassembly Program,” ANL-7951, Argonne National Laboratory, 1972.

VOLUME I – SAND2012-4260

DISTRIBUTION

External Distribution

1 Tanju Sofu
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60439

1 George Flanagan
PO Box 2008
MS 6165
Oak Ridge, Tennessee 37831-6165

1 Robert Bari
Brookhaven National Laboratory
Building 130
PO Box 5000
Upton, NY 11973-5000

1 Roald Wigeland
Idaho National Laboratory
PO Box 1625
Idaho Falls, ID 83425-3860

1 Tom Wei
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60439

1 Robert N. Hill
Nuclear Energy Division
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60439

1 Matthew Hutmaker
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

1 Craig Welling
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

1 Brian Robinson
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

1 Jason Tokey
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

1 Timothy O'Conner
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

1 Sal Golub
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

1 John Kelly
U.S. Department of Energy
NE-74/GTN
1000 Independence Avenue SW
Washington, DC 20585

Internal Distribution

1	MS0748	Jeffery L. LaChance	06231
1	MS0736	Evaristo Bonano	06220
1	MS0736	Susan Y. Pickering	06230
1	MS0747	Matthew Denman	06231
1	MS0899	Technical Library	9536 *(electronic copy)*

Sandia National Laboratories