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Sodium Fast Reactor Safety and Licensing Research Plan – Volume II

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

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J. LaChance, Steve Wright, A. Suo-Anttila,
J. Hewson, T.J. Olivier, J. Phillips, M. Denman,
D.A. Powers, and R. Schmidt
Sandia National Laboratories
Albuquerque, NM 87185-0748

J. Sackett, R. Wigeland, S. Hayes, and D. Porter
Idaho National Laboratory
Idaho Falls, ID 83415

R. Bari, H. Ludewig
Brokhaven National Laboratory
Upton, NY 11973

R. Budnitz
Lawrence Berkeley National Laboratory
Berkeley, CA 94720

J. Cahalen, C. Grandy, D. Wade, M. Farmer,
J. Lambert, K. Natesan, A. Wright, A. Yacout,
T. Sofu, T. Wei, and J. Thomas
Argonne National Laboratory
Argonne, IL 60439

M. Corradini
University of Wisconsin, Madison
Madison, WI 53706

R. Denning
Dept. of Mechanical Engineering
Ohio State University
Columbus, OH 43210-1142

G. Flanagan, L. Ott, and J. Carbajo
Oak Ridge National Laboratory
Oak Ridge, TN 37831

S. Miyhara, S. Ohno, H. Ohshima, and Y. Tobita
Japan Atomic Energy Agency
4002 Narita-cho, O-arai-machi, Ibaraki, 311-1393, JAPAN

L. Walters
Advanced Reactor Concepts
Los Alamos, NM 87544

F. Garner
Radiation Effects Consulting
Richland, WA 99354

B. Clément
IRSN/DPAM.SEMIC Bt 702
BP3 13115 Saint-Paul-lez-Durance, FRANCE

R. Zeyen
Institute for Energy Petten
P.O. Box 3, Building 250, F-13115 Saint-Paul-lez-Durance, FRANCE

H. Jeong
Korea Atomic Energy Research Institute
Daejeon, 305-352 KOREA

F. Serre
Centre d'études nucléaires de Cadarache – CEA
FRANCE

ABSTRACT

Expert panels comprised of subject matter experts identified at the U.S. National Laboratories (SNL, ANL, INL, ORNL, LBL, and BNL), universities (University of Wisconsin and Ohio State University), international agencies (IRSN, CEA, JAEA, KAERI, and JRC-IE) and private consultation companies (Radiation Effects Consulting) were assembled to perform a gap analysis for sodium fast reactor licensing. Expert-opinion elicitation was performed to qualitatively assess the current state of sodium fast reactor technologies. Five independent gap analyses were performed resulting in the following topical reports:

- Accident Initiators and Sequences (i.e., Initiators/Sequences Technology Gap Analysis),
- Sodium Technology Phenomena (i.e., Advanced Burner Reactor Sodium Technology Gap Analysis),
- Fuels and Materials (i.e., Sodium Fast Reactor Fuels and Materials: Research Needs),
- Source Term Characterization (i.e., Advanced Sodium Fast Reactor Accident Source Terms: Research Needs), and
- Computer Codes and Models (i.e., Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety).

Volume II of the Sodium Research Plan consolidates the five gap analysis reports produced by each expert panel, wherein the importance of the identified phenomena and necessities of further experimental research and code development were addressed. The findings from these five reports comprised the basis for the analysis in Sodium Fast Reactor Research Plan Volume I.

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Advanced Sodium Fast Reactor Accident Initiators/Sequences Technology Gap Analysis

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
Reactor Campaign

John Sackett, Chair, Roald Wigeland
Idaho National Laboratory
Robert Bari
Brookhaven National Laboratory
Robert Budnitz
Lawrence Berkeley National Laboratory
Jim Cahalan, Chris Grandy, Dave Wade
Argonne National Laboratory
Michael Corradini
University of Wisconsin, Madison
Richard Denning
Ohio State University
George Flanagan
Oak Ridge National Laboratory
Steve Wright
Sandia National Laboratories

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SUMMARY

An advanced Sodium-Fast-Reactor (SFR) is being evaluated by DOE to provide the capability to transmute actinides and enhance the long-term fissile fuel-supply for fission reactors. An essential element in this evaluation is whether an adequate technology base exists to support the safety case for an SFR.

The panel concluded that there are no major technology gaps in preparing a safety case for an advanced SFR, so long as one stays with known technology. Defining the current state of knowledge was therefore an important activity of the panel, along with the context in which it can be used for licensing. Significant potential departures from known technology were identified, such as development of fuel containing high concentrations of minor actinides, which will require further investments in R&D both to develop the technology and to develop an adequate safety case.

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TABLES

ACRONYMS

ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transients Without Scram
BDBA	Beyond Design Basis Accident
BOP	Balance of Plant
CRBRP	Clinch River Breeder Reactor Plant
DBA	Design Basis Accident
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor-II
FFTF	Fast Flux Test Facility
GDC	General Design Criteria
GEMS	Gas Expansion Modules
HCDA	Hypothetical Core Disruptive Accident
IFR	Integral Fast Reactor
IHX	Intermediate Heat Exchanger
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk-Assessment
PRISM	Power Reactor Inherently Safe Module
R&D	Research and Development
SAFR	Sodium Advanced Fast Reactor
SFR	Sodium Fast Reactor
TREAT	Transient Reactor Test Facility
ULOF	Unprotected Loss of Flow
ULOHS	Unprotected Loss of Heat Sink
UTOP	Unprotected Transient Over Power

REACTOR CAMPAIGN/ADVANCED SODIUM FAST REACTOR ACCIDENT INITIATORS/SEQUENCES TECHNOLOGY GAP ANALYSIS

1. INTRODUCTION

Sodium-cooled fast reactor design, safety analysis and licensing were central themes of reactor development for the U.S. Department of Energy from the early 1960's through the mid 1990's. The U.S. Nuclear Regulatory Commission supported parallel programs to develop their regulatory capabilities to license these designs. These programs involved major nuclear test facilities, extensive analytic support and international cooperation. In addition to the US programs, there is an active international program of fast reactor development, building upon decades of research and development. Since sodium-cooled fast reactors are now being considered as one option for potential future nuclear fuel cycle systems, it is appropriate to evaluate the state of knowledge for such reactors with regards to the potential risk posed to the public. The evaluation can be used to help define research and development (R&D) that would be desirable or required to support the design and licensing of such reactors.

1.1 Objectives

The task of this panel is to assess the completeness and relevance of existing information, specifically as follows:

"Assess the status of knowledge pertaining to accident phenomena important to safety-analysis and licensing of a sodium-cooled fast reactor (SFR) and identify knowledge or capability gaps."

In doing so, the panel is to take into account any and all phenomena that may occur in response to an accident, realizing that it may be possible to make design choices that would either alter the importance of any given phenomenon, or eliminate it entirely from consideration. In order to make the study as general as possible, the activities pursued to accomplish the objective included identifying all important safety-relevant phenomena, evaluating the state of knowledge for each phenomenon, and identifying any gaps in knowledge or technology that would require R&D effort.

1.2 Analysis and Evaluation Approach

Since the mission of this working group is to identify knowledge or capability gaps important to safety analysis and licensing of future sodium fast reactors (SFR), the panel identified general reactor transient and accident sequences that are important for establishing the overall safety characteristics of a particular reactor design. Next, drawing on the expert knowledge base of the panel members, the physical phenomena that are judged to be guiding or ruling in each accident sequence were identified. After this step was completed, consideration was given to the safety approach that could be taken with an advanced sodium-cooled fast reactor, including use of design features that are beneficial in preventing accidents or mitigating their consequences. This allowed the relative importance of various types of accidents to be considered in the evaluations. Finally, the listed phenomena were assessed for a) importance to the safety case, b) knowledge for phenomenological modeling for analysis, and c) adequacy of supporting experimental data.

As a result, a broad set of safety-relevant phenomena was addressed to identify potential gaps in the ability to license a modern SFR. The scope of these considerations extended to secondary systems and balance of plant interactions with accident events where appropriate. However, accidents not directly associated with reactor operation, such as fuel handling or fuel storage accidents, were considered to be outside the scope of the panel's consideration.

2. GENERIC EVENTS, ACCIDENTS AND PHENOMENA IDENTIFICATION

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

- protected accidents - an accident initiator occurs, such as a component failure, failure of a safety grade system (other than the reactor protection systems), or an external event, followed by successful activation of the plant protection systems to shut down the reactor
- unprotected accidents – an accident initiator occurs as in the case for protected accidents, but the reactor protection systems fail to function. Such accidents may result in fuel damage, fuel melting, and fuel pin failures. For the purposes of these evaluations, accidents where fuel melting and fuel pin failures are widespread throughout the reactor core are treated in the severe accident category
- severe accidents with core melting – typically an unprotected accident where the failure of the reactor protection system results in conditions within the reactor such that widespread melting and failure of the reactor fuel occurs

Given these general categories, the three general types of upset conditions were considered, (a) reduction or loss of core cooling, (b) addition (or insertion) of reactivity to the reactor core, and (c) reduction or loss of heat removal capability from the reactor. For these general accident initiators, Table 1 identifies the systems, subsystems, or components that may be involved in the accident and the phenomena that could potentially occur.

Table 1: Event Descriptions and Relevant Phenomena

Event Description	Key Systems Involved	Relevant Phenomena
Protected Events		
Loss of Core Cooling		
Equipment Failure electrical faults loss of site power controller failures internal flow blockage mechanical faults pump mechanical failure loss of piping integrity ***Operator Error*** turning off pump power opening breakers to power supplies ***External Events*** earthquakes, fire, flood, tornado, terrorist	**Component or System** primary pump power supplies shaft/ bearing/ impeller off-site power connection primary piping and vessel system core and assembly coolant flow channels fuel cladding reactor control and protection systems shutdown heat removal systems reactor containment electrical-magnetic pump power leads	**Thermal-hydraulics** single phase transient sodium flow thermal inertia pump-coast down profiles sodium stratification transition to natural convection core cooling core flow redistribution in transition to natural convection decay heat generation **Reactivity Effects Prior to Scram** mechanical changes in core structure intact fuel and fuel pin motion fuel/coolant/structure temperatures **Material Behavior** structure behavior at elevated temperatures cladding integrity margin leak-before-break behavior of piping primary coolant boundary integrity margin containment building integrity margin thermal shock to structures

Table 1: Event Descriptions and Relevant Phenomena (continued)

Event Description	**Key Systems Involved**	**Relevant Phenomena**
Protected Events		
Reactivity Addition		
Equipment Failure uncontrolled control-rod motion overcooling from pump speed increase Balance of Plant (BOP) system pressure loss gas bubble entrainment ***Operator Error*** control-rod movement error coolant pump control error actuation of BOP pressure relief valve ***External Events*** Earthquakes	**Component or System** reactor control system and control rod drives primary pumps BOP heat removal systems shutdown heat removal primary and intermediate cooling systems reactor protection systems BOP control systems reactor containment	**Reactivity Effects Prior to Scram** reactivity feedback at high power end-of-life prediction of reactivity feedback burnup control swing / control rod worth reactivity effects of gas bubble entrainment integrity of fuel with breached cladding integrity of fuel with load following
Loss of Normal Heat Rejection		
Equipment Failure steam generator failure intermediate heat transport system failure supercritical CO_2 system failure loss of electric grid load flow blockage in heat transfer loop ***Operator Error*** stopping intermediate loop flow steam generator blow down isolating plant from the grid ***External Events*** earthquake, fire, flood, tornado, terrorist	**Component or System** secondary sodium pumps secondary system piping steam generators sodium-CO_2 heat exchanger turbine-generators shutdown heat removal systems intermediate heat exchanger reactor protection systems reactor containment	**Thermal-hydraulic effects:** sodium-steam chemical reaction CO_2-sodium chemical reaction pressure-pulse impacts from chemical reaction decay heat generation **Material Behavior:** long-term performance of structures at elevated temperatures

Table 1: Event Descriptions and Relevant Phenomena (continued)

Event Description	Key Systems Involved	Relevant Phenomena
Unprotected Events (Anticipated Transients – Without Scram, ATWS)		
Loss of Core Cooling (ATWS)		
Reactor shutdown system failure following: electrical faults mechanical faults loss of site power loss of piping integrity internal flow blockage	**Component or System** primary pump power supplies pump mechanicals off-site power primary piping system core and assembly coolant flow channels core structure fuel and subassemblies primary coolant system Inherent and passive safety systems flow coast down extenders	**Same as for protected events plus:** **Thermal-hydraulics** thermal inertia pump-coast down profiles sodium stratification margin to boiling at peak temperature core thermal and structural effects heat removal path and capacity **Reactivity Effects** core reactivity feedback fuel motion in intact fuel pins core restraint system performance reactor shutdown mechanism **Material Behavior** long-term performance of structures at elevated temperatures fuel cladding integrity at elevated temperatures
Reactivity Addition (ATWS)		
Reactor shutdown system failure with: uncontrolled withdrawal of a single control rod overcooling from pump speed increase	**Component or System** reactor shutdown systems control rod drive system fuel and subassemblies primary pumps BOP heat rejection system	**Same as for protected events plus:** **Thermal-hydraulics** heat removal path/capacity **Reactivity Effects** reactivity feedback at high power coolant heating and margin to boiling core reactivity feedback core thermal and structural effects **Material Behavior** fuel cladding structural integrity at elevated temperatures cooling systems structural integrity at elevated temperatures containment structure integrity
Loss of Normal Heat Rejection (ATWS)		
Reactor shutdown system failure with: steam generator failure intermediate heat transport failure supercritical CO_2 system failure decay heat removal system failure	**Component or System** secondary sodium pumps secondary system piping and intermediate heat exchangers (IHX) steam generators decay heat removal systems sodium-CO_2 heat exchanger	**Same as for protected events plus:** **Thermal-hydraulics** thermal inertia core thermal / structural effects **Reactivity Effects:** core reactivity feedback fuel motion in intact fuel pins core restraint system performance reactor shutdown mechanism **Material behavior** long-term performance of structures at elevated temperatures fuel cladding structural integrity at elevated temperatures containment structure integrity

Table 1: Event Descriptions and Relevant Phenomena (continued)

Event Description	Key Systems Involved	Relevant Phenomena
Severe Accidents – Substantial Core Melting		
Severe loss of core cooling event **Severe reactivity addition event** **Severe loss of heat rejection capability**	**Component or System** core fuel and assemblies core grid and restraint structure primary coolant system containment building support structure seismic isolation	**Same as for above plus:** **Fuel and Core Behavior:** sodium voiding effects temporal and spatial incoherence fuel pin failure fuel dispersal and coolability re-criticality potential for energetic events primary vessel thermal and structural integrity radiation release and transport

3. APPROACH TO SAFETY FOR THE SFR

The events and accidents that could occur in a modern sodium-fast-reactor (SFR) provide the context in which the panel made its assessments of the status of knowledge about accident phenomena. In particular, since a large class of Anticipated Transients - Without Scram (ATWS) events can be accommodated in modern SFR designs without resulting in serious damage to the reactor, the question arises about appropriate accidents to be considered that challenge those barriers. It has been shown one can choose to design a system that will accommodate the Unprotected Loss of Flow (ULOF), Unprotected Transient Over Power (UTOP) and Unprotected Loss of Heat Sink (ULOHS) ATWS events with no or only minor fuel damage.

It must be recognized that the NRC has limited experience with licensing a sodium-cooled fast reactor, confined to the CRBR license application, the review of the FFTF design that was ultimately licensed by DOE, and the pre-application discussions for the Power Reactor Inherently Safe Module (PRISM) reactor and the Sodium Advanced Fast Reactor (SAFR). (The FFTF review resulted in a letter of approval from the Advisory Committee on Reactor Safeguards, verifying the adequacy of the safety analysis.) At this time, the General Design Criteria (GDC) for an SFR do not exist, with previous approaches being based on examining existing GDCs for other reactors to evaluate their applicability to the SFR and to identify the need for new GDCs. Based on recent regulatory trends, it is likely that the approach used to license an SFR would involve the use of best-estimate analyses with quantified uncertainties. It is also very likely that probabilistic risk-assessment (PRA) will play an important role. The NRC has also indicated that new reactors would be held to more stringent risk requirements than the current generation of plants.

It is expected that accidents analyzed for the license application will cover the entire range of probability, including accident initiators of very low probability. For the purposes of the evaluations performed in this study, it is possible to consider the events that might occur in an SFR as listed in Table 1 using the general guidelines that the NRC has established for such events.

3.1 Designs for Safety and Defense-in-Depth

In addition to strict standards for safety-relevant equipment and systems, reactors traditionally have diverse and redundant safety features to provide defense-in-depth against any potential event. Plant functions that are important to safety are designed according to the defense-in-depth principle, which provides multiple layers of safety assurance. Multiple, diverse, and independent structures, systems, or components are provided, each capable of achieving the defined safety function. Redundancy, diversity, and independence assure that loss of all safety functions due to a single failure, either internal (equipment failure, operator action) or external (earthquake, fire, flood), is extremely unlikely. Safety grade systems, components, and structures are designed and maintained to criteria that assure their reliable operation, with special attention to quality assurance and provisions for inspection, testing, and repair. Examples of such an approach are:

- *Containment* of radioactive material is assured by multiple physical barriers; the fuel cladding, the primary coolant system boundary, and the containment structure
- *Reactor shutdown* is assured by multiple redundant and diverse safety systems capable of independently providing shutdown
- *Residual heat removal* is assured by multiple heat transport paths and systems: the normal heat removal system (steam generator, condenser), and dedicated emergency shutdown heat removal systems

The result is that higher probability events are accommodated with no challenges to the reactor system, and that events that could pose a threat to any of the barriers to the release of radiation are possible only at extremely low probabilities of occurrence. More recently, the incorporation of self-protective features in the design which rely on passive inherent response to provide protective margins for very low probability events, including Anticipated Transients - Without Scram (ATWS), allows for even these accidents to result in potentially benign consequences. The passive inherent features include favorable reactivity feedback in response to upsets in normal reactor core conditions, and the ability to provide adequate cooling using natural convective flow, i.e., without requiring forced pumping. These characteristics can be used to increase safety margins and reduce risks, in essence reflecting another aspect of defense-in-depth philosophy that involves both prevention and mitigating measures. For the purposes of the evaluations in this study, it is assumed that an SFR would take advantage of such design features.

3.2 Proposed Safety Performance

There is a broad spectrum of potential accidents that can occur in a nuclear power plant depending on the occurrence of initiating events and the assumed failure of safety-related or protective features. The traditional licensing process is designed to address a range of accidents in terms of their likelihoods and potential consequences. A familiar classification scheme is shown in Table 2, typical of what is proposed in the NRC's Standard Review Plan used for reactor licensing. The frequency and allowable consequences in this table reflect safety standards that are an order of magnitude more stringent than were established for existing plants, reflecting the NRC's desire that any new reactor provide enhanced levels of safety.

Table 2: Classification of Events and Consequences for Reactor Licensing

Events	Frequency	Current NRC Allowable Consequences
Anticipated Operational Occurrences (AOOs)		
Operational events	Expected during the lifetime of the plant (> 10^{-2} per reactor year)	None; maintain margin to fuel damage
Postulated Accidents		
Design Basis Accidents (DBAs), typically failure of one safety grade system	Not expected to occur during the lifetime of the plant, but anticipated in the design; probability > 10^{-5} per reactor year	Minor fuel damage permissible at lower probability (< 10^{-4} per reactor year); allowable individual exposure < 25 rem
Beyond Design Basis Accidents (BDBAs), e.g., multiple failures of safety grade systems, including ATWS events	Accidents of very low probability not considered as part of the design basis for the plant; probability < 10^{-5} per reactor year	Substantial fuel damage permissible; allowable exposure > 25 rem to public at lower probability (<10^{-6} per reactor year)

For the sodium-cooled fast reactor (SFR) considered in this study, a significantly enhanced safety performance is proposed, partly to reflect the use of passive inherent safety concepts. The proposed safety performance that was assumed in this study is listed in Table 3. One major difference between the events and consequences listed in Table 2 and those in Table 3 is that the ATWS events and other similar events with a frequency of occurrence between 10^{-5} and 10^{-7} per reactor year are now anticipated to have no significant consequences beyond limited fuel damage and few, if any, fuel pin failures. There would be no uncontrolled release of radioactive materials for accidents in this range of frequency. As a result, substantial fuel damage, fuel pin failures, and any challenge to primary system or containment integrity may only be expected to occur at frequencies of occurrence less than 10^{-7} per reactor year. The likelihood of severe consequences from such low frequency events and the magnitude of the consequences, including any uncontrolled release of radioactive materials, would depend on the details of the specific accident scenario as well as the choices made in developing the design.

Table 3: Modified Event Consequences for Licensing a Sodium-Cooled Fast Reactor with a Higher Level of Safety

Events	Frequency	Proposed Anticipated Consequences
Anticipated Operational Occurrences		
Operational Events	Expected during the lifetime of the plant (> 10^{-2} per reactor year)	None, maintain margin to fuel damage
Postulated Accidents		
Design Basis Accidents (DBAs) , typically failure of one safety grade system	Not expected to occur during the lifetime of the plant, but anticipated in the design; probability > 10^{-5} per reactor year	Minor fuel damage possible at lower probability (< 10^{-4} per reactor year); no uncontrolled releases of radioactive materials, no significant exposure
Beyond Design Basis Accidents (BDBAs), e.g., ATWS events (AOO with failure to scram)	Accidents of very low probability not considered as part of the design basis for the plant; 10^{-5} > probability > 10^{-7} per reactor year	Minor fuel damage possible at lower probability; no uncontrolled releases of radioactive materials, no significant exposure
Beyond Design Basis Accidents (BDBAs), e.g., multiple failures of safety grade systems, more severe than ATWS events	Accidents of extremely low probability not considered as part of the design basis for the plant; probability < 10^{-7} per reactor year	Substantial fuel damage may occur; uncontrolled release of radioactive material may result in exposure > 25 rem to public

Consistent with NRC expectations, the first aspect of safety performance is that Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBAs) do not result in fuel failure. The second aspect is that there is high confidence that the higher probability Beyond Design Basis Accidents (BDBAs) will result in at most limited fuel damage, possibly isolated fuel pin failures, and minimal threats to primary system integrity. Depending on design details, the Anticipated Transients - Without Scram (unprotected accidents) are expected to have very small frequencies of occurrence and could result in little or no fuel pin damage. The third aspect addresses what have been historically considered as severe accidents that involve substantial core melting. Such accidents may have the potential for large releases depending on the details of the accident sequence, including the potential for re-criticalities as core materials relocate from their original locations within the core. These accidents are also known as hypothetical core disruptive accidents (HCDAs), and the reactor is designed so that their frequencies are below 10^{-7} per reactor year in our formulation (Table 3).

The overall result is that the probability of loss of barriers to radiation release can reasonably be shown to be extremely low. The panel considered whether there were gaps in understanding the underlying phenomena important to achieving this result, as well as design features important to reducing uncertainties. However, because of the very low probability of BDBAs and ATWS events, the requirements for understanding and accurate modeling of phenomenology in this area of investigation are not as demanding and greater uncertainties are acceptable, which is reflected in the evaluation of the adequacy of knowledge for the accident phenomena.

3.3 Margins Beyond the Design Basis

In addition to safety margins provided by design features belonging to the design basis, the NRC also expects information on the performance of the design in events that exceed the normal safety design envelope. The BDBAs have been described above, and analyses of such events are needed to provide estimates of plant performance. However, unlike AOOs and DBAs, such analyses are typically done on a 'best estimate' basis rather than a conservative basis, and higher uncertainties are both expected and allowed.

In past U.S. regulatory reviews of the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant (CRBRP), ATWS events were identified as *precursors for severe accident progression (core melting).* In particular, the ULOF accident sequence (total loss of reactor coolant flow beginning from full power and flow) was identified as an enveloping event for containment margin assessment. Following the FFTF/CRBRP era, further research and development in the U.S. led to safety enhancements. The approach developed from findings of the Advanced Liquid Metal Reactor (ALMR) program in the 1980's, and in particular the Integral Fast Reactor (IFR) program at Argonne National Laboratory (ANL). Testing performed at Experimental Breeder Reactor-II (EBR-II) in 1986 demonstrated benign reactor behavior (no cladding failures, no coolant boiling, and no fuel melting) in full-scale ULOF and ULOHS accident conditions. These tests showed that the key design features for *prevention of severe accident progression* are 1) inherent reactivity feedbacks to shut down fission power in unprotected accidents, and 2) natural circulation shutdown heat removal. The EBR-II tests demonstrated that with appropriate design selections, the consequences of accident initiators that would have resulted in core melting in FFTF and CRBRP could be limited to an elevated coolant temperature only slightly above the normal operating temperature and without fuel failure. These design features were employed in the Rockwell International SAFR and General Electric PRISM concepts developed in the ALMR program that was supported by DOE.

With the use of passive inherent safety concepts and natural convection cooling, it is possible to quantify safety margins with greater certainty for BDBAs such as ATWS events when no serious consequences occur. Accidents where higher uncertainties are unavoidable due to the severity of the consequence can be relegated to much lower probabilities where higher uncertainties in the estimation of safety margins can be acceptable in the analyses.

4. EVALUATION OF STATE OF KNOWLEDGE

The approach used to evaluate sodium-cooled Fast Reactor Safety phenomenology assumes that the safety and licensing of future plant designs will combine traditional deterministic methods with risk-informed and performance-based methods, and will incorporate passive safety features in the design.

4.1 Importance Ranking

The importance ranking categories are qualitative levels of High (H), Medium (M), and Low (L) which have been found in previous studies to provide adequate resolution and be consistent with an expert opinion process. The general descriptions of these importance ranking levels are:

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

4.2 Figures of Merit

The highest level evaluation criterion or figure of merit (FOM) is radioactive material released to the public, which is common to all of the gap analysis panels. Evaluation criteria for the SFR accident sequence panel are:

- *Radiological release* resulting in a dose at the site boundary.
  This criterion is applied to all accident types. The potential doses from anticipated operational occurrences (AOOs) and design basis accidents (DBAs) are limited to well below regulatory limits. Depending on design choices and accident probability, the potential doses from BDBAs and severe accidents may be significantly higher, although they would still within the applicable regulatory limits depending on the probability of occurrence.

- *Challenges to barriers to radiological release*.
  For AOOs and DBAs, the criteria generally used in licensing a reactor are deterministic, reflecting a significant safety margin to failure of the barriers and defense-in-depth philosophy. For AOOs, it is required that there are no barrier failures and that there are diverse and redundant means to shut down the reactor and provide cooling of the reactor. For DBAs, it is required that the core remains coolable, the vessel and containment remain intact, and at least one system to shut down and cool the core remains operable. For beyond design basis accidents (BDBAs), there are no functional operability requirements except for those needed to reduce the radiological release to acceptable levels, although all safety-grade systems are assumed to function except for those whose failure created the accident initiator. While the only requirement that the NRC Commission has stated is that future plants be safer, this is normally interpreted to imply that the core damage frequency be less than 10-5 per reactor year.

4.3 Knowledge Based Ranking

Expert Opinion used in evaluating the state of knowledge of a phenomenon involved the assessment of both the modeling capabilities and the database to validate the model. General criteria for each level of the assessment are as follows:

High (H)

- A physics-based or correlation-based model is available that is believed to accurately represent the phenomenon over the parameter space of interest.

- A database adequate to validate relevant models exists, or the data are available to make an assessment, consistent with the level of accuracy required for the analyses given the importance of the phenomenon.

Medium (M)

- A candidate model or correlation is available that addresses most of the phenomenon over at least some portion of the parameter space, or represents the phenomenon over the parameter space with a higher degree of uncertainty.
- Data are available but are not necessarily complete or of high fidelity, allowing only moderately reliable assessments, sufficient for scenarios and phenomena of medium or low importance

Low (L)

- Model applicability has large uncertainty or speculative aspects.
- Little or no existing database, assessments have large uncertainty, but may be adequate for phenomena of low importance.

Major Assumptions:

1. It is assumed in the evaluations that information on known technology is available and accessible to a fast reactor project, and that a full-fledged knowledge management effort is in place to avoid repetition of past R&D.
2. Verification of predicted reactor system response to upsets as part of plant qualification testing can be used to reduce uncertainties in expected reactor performance based on modeling.

4.4 Evaluation of Gaps

Identification and ranking of phenomena are grounded in the defense-in-depth safety principle applied to the three basic safety design functions: 1) reactor shutdown and control, 2) reactor shutdown cooling, and 3) containment. In a conventional reactor design, these three functions are accomplished with multiple, diverse, and independent design features that yield a very high level of safety reliability and public protection, and reduce the probability of public risk to a very low level.

4.4.1 Loss of Core Cooling Events

The most important phenomena are those ensuring adequate core cooling with or without reactor scram. In particular, reactor cooling to maintain fuel cladding integrity (the first containment barrier) is of paramount importance. When reactor shutdown and primary cooling systems operate as designed, as occurs for AOOs and DBAs, cladding integrity is guaranteed by design. However, if active shutdown and primary cooling systems should fail, an event of very low probability, SFRs are capable of inherent reactor power shutdown and natural circulation decay heat removal. As such, much attention has been paid in past R&D to developing models which accurately predict the transition to natural convection cooling, the temperatures that occur during that transition and the reactivity changes that result.

There is an extensive reference base (Appendix A) that details the work done in the U.S. and addresses these phenomena. Generally, the level of knowledge is quite good (medium to high). There is a significant body of test data, both from single phenomenon tests in water and sodium and from integral tests in both EBR-II and FFTF. In particular, the EBR-II tests were conducted over a period of 12 years and generated a great deal of important data which has been used in benchmarking codes for predicting response of advanced reactor designs. These tests included characterization of the natural convective flow of sodium coolant for a wide range of reactor conditions, including dynamic transition to natural convective flow from a variety of initial reactor powers and temperatures. Also done were extensive measurements of reactivity feedbacks from all sources, using techniques such as rod drops and power oscillations to separate individual components. The culminations of these tests were the Inherent Safety

Demonstration Tests which demonstrated inherently safe response of the reactor to Unprotected Loss-of-Flow and Loss-of-Heat-Sink. In addition, tests were conducted to demonstrate safe reactor response to overcooling and load following characteristics in response to changes in power demand. These data have been made available as part of the knowledge management program at EBR-II and have been captured in digital form to support development of advanced modeling and simulation.

Extensive tests were also conducted at the FFTF, including full characterization of transition to natural convective flow and characterization of reactivity feedbacks in the reactor. Inherent Safety Demonstration tests were conducted from partial power, assisted by Gas Expansion Modules (GEMS). In combination, these two series of tests thoroughly explored the inherent behavior of sodium-cooled fast reactor systems to ATWS events. In addition, extensive out-of-pile tests have been conducted to characterize natural convective flow in sodium systems, involving both water and sodium loops.

A benefit of a system which takes full advantage of passive inherent features in the design is that the behavior of convective flow, either forced flow or by natural convection and its impact on reactivity feedback can be verified in the actual plant once built. This ability is important to evaluating the importance of uncertainty in modeling and is an important consideration for the gap analysis panel in judging the adequacy of information to support these designs.

4.4.2 Reactivity Addition Events

Investigation of transient-overpower events, both protected and unprotected has also been an important topic of investigation (see references Appendix A). For the less severe overpower transients, there is significant data from operational transient tests in EBR-II that provide significant confidence in the ability to model fuel performance and the consequences of failure. (Such tests included transients on fuel with breached cladding to determine the potential for fuel loss to the coolant). Reactivity effects of mechanical changes in core structure, sodium density effects and changes in fuel structure have also been extensively studied starting with the investigation of fuel pin bowing in EBR-I. An important aspect of the safe response to events with the uncontrolled withdrawal of single control-rod is limiting the available reactivity in the control rod itself. The excess reactivity required for reactor startup and to accommodate burnup of the fuel is a function of design choices and is under the control of the designer.

The severe-overpower and under-cooling transients received a great deal of attention in the 1970s and 1980s, especially given the potential for leading to Hypothetical Core Disruptive Accidents (HCDAs), focusing on transient tests in the Transient Reactor Test facility (TREAT) and other similar facilities. For such transients, analysis uncertainties will always be higher than for less benign conditions because of the complexity of the events, the rapidity with which numerous different phenomena occur, and the difficulties of performing and instrumenting experiments. Larger uncertainties are acceptable for the more complex phenomena because they are less probable. However, as discussed above, a reasonable objective of the plant designer is to decrease the probability of entering into such severe accident conditions by the use of redundant and diverse safety systems (prevention by defense-in-depth), by using favorable passive inherent design features (accident consequence mitigation), and by providing additional active design features for accident mitigation (e.g. self-actuated shutdown systems or FFTF gas expansion modules). Some SFR designers have also provided features for mitigation of severe accident conditions that could occur with substantial melting of the reactor core. By properly using these features in reactor design, it is possible to greatly reduce the probability of such events in an SFR to a very low level, and potentially avoid having this type of accident become a focus of regulatory concern.

Table 4 collects the phenomena under general categories appropriate to accident consequences, and then identifies the importance of these phenomena to the safety case, and the level of knowledge (data, analysis methods, etc) currently available to model these phenomena under accident conditions, consistent with the level of knowledge required based on frequency of occurrence.

Table 4: Evaluation of Phenomena and Their Importance

Modeling Issue	Underlying Phenomenon	Importance To safety Case	Knowledge Adequacy	
			Modeling	Experimental data
DBAs and BDBAs not leading to fuel failure				
Reactivity Feedbacks in Transients (High Importance)				
Mechanical changes in core structure	expansion of core grid structure	High	High	High
	expansion of control rod drives	High	High	High
	mechanical changes in core structure over life (swelling, etc)	High	High	High
	bowing of fuel assemblies and blanket	High	High	High
	core restraint system performance	High	High	High
	axial thermal expansion of fuel and cladding			
	Metal	High	High	High
	Oxide	High	Medium	High
	reactivity feedback coefficients from mechanical changes	High	High	High
Intact fuel and fuel changes	fission product impacts on fuel structure and properties	High	High	High
	Doppler feedback as a function of fuel composition	High	High	High
	cross section information for minor actinides	Low	Medium	Low
	end-of-life power distribution and control rod position	High	High	High
	end-of-life fuel composition	High	High	High
	end-of-life prediction of reactivity feedback	High	Medium	Medium
	burnup control swing	High	Medium	Medium
	control rod worth	High	High	High
	reactivity feedback at high temperature	High	High	High
	axial growth of fuel with irradiation			
	Metal	High	High	High
	Oxide	Low	High	High
Sodium density effects	sodium temperature coefficient of reactivity	High	High	High
	sodium void coefficients	High	High	High
Margin to Fuel Cladding Failure (High Importance)				
Fuel cladding failure	fuel cladding failure mechanisms			
	metal	High	High	High
	Oxide	High	High	High
	metal fuel cladding failure time and location	High	High	High
	oxide fuel cladding failure time and location	High	Medium	Medium
Fluid Flow and Heat Transfer (High Importance)				
Steady-state and transient forced convection	single phase sodium forced flow	High	High	High
	sodium convective heat transfer	High	High	High
	fuel pin heat removal	High	High	High
Transition to natural convective cooling	single phase transient sodium flow	High	High	High
	pump-coast down profiles	High	High	High
	sodium stratification	High	Medium	High
	core flow redistribution in transition	High	High	High
	coolant heat up profile and margin to boiling	High	High	High
Thermal response of structures	thermal shock to structures	High	High	High
	thermal striping	High	High	Medium
	structure heat conduction	High	High	High
	radiation heat transfer from vessels	High	High	Medium
Decay heat rejection	convective heat transfer	High	High	High
	cooling systems structural integrity over time	High	High	High
	natural circulation heat removal	High	High	High

Table 4: Evaluation of Phenomena and Their Importance (continued)

Modeling Issue	Underlying Phenomenon	Importance To safety Case	Knowledge Adequacy: Modeling	Knowledge Adequacy: Experimental data
Power conversion	steam-sodium reactions	High	High	High
	pressure pulse migration	High	High	High
	CO2-sodium chemical interaction (supercritical CO2 cycle)	High	Low	Low
	High pressure CO2 release and impact (advanced cycle)	High	Low	Low
Fuel Transient Behavior (High Importance)				
	Evolution of fuel and cladding over life	High	High	High
	Cladding structural integrity (margin)	High	High	High
	Length effects on fuel performance during transients			
	Metal	Medium	High	Low
	Oxide	Medium	Medium	Medium
	Fuel-pin behavior with breached cladding:			
	Metal	Low	High	High
	Oxide	Medium	High	High
	High-minor-actinide content fuel performance	High	Low	Low
	source term is different			
	physics are different			
	chemistry is different			
Material Interactions and Chemistry (High Importance)				
	Sodium vapor condensation and plate out (system degradation)	High	High	High
	structural material corrosion	Low	High	High
	sodium purity control	High	High	High
Structural Mechanics (High Importance)				
Seismic response	Seismic response of reactor core and coolant system	High	High	High
	Seismic response of containment	High	High	High
DBAs and Beyond DBA Phenomenology With Fuel Pin Failures				
Localized core damage (Low Importance)				
	Local flow blockage			
	fission product transport and delayed neutron detection	High	High	High
	extent of fuel melting within affected subassemblies	High	High	High
	propagation of fuel melting across subassemblies			
	Metal	High	High	High
	Oxide	High	High	High
Severe Core Damage (Medium Importance)				
	Sodium voiding effects			
	temporal and spatial incoherence	High	High	High
	bubble growth at boiling temperature	High	High	High
	thermal-hydraulic effects	High	High	High
	Fuel failure			
	failure mode and location			
	Metal	High	High	High
	Oxide	High	High	High
	fuel motion, dispersal, morphology			
	Metal	High	Medium	Medium
	oxide (including fuel-coolant-interaction)	High	Medium	High

Table 4: Evaluation of Phenomena and Their Importance (continued)

Modeling Issue	Underlying Phenomenon	Importance To safety Case	Knowledge Adequacy	
			Modeling	Experimental data
	Pre-existing radionuclide distribution in the pin (ST)			
	metal (including bond)	High	High	High
	Oxide	High	High	High
	Coolability of rubble/debris bed			
	Metal	High	High	High
	Oxide	High	High	High
	Pressure sources/primary system loads (ST)	High	High	High
	Primary system response to loads (ST)	High	High	High
Challenges to Containment (Medium Importance)				
	Pressure sources/containment loads	High	High	High
	Containment response to loads	High	High	High
	Sodium-concrete interactions (Sodium group)			
	Sodium fire with contaminated sodium (ST) (sodium group)			
	Ultimate heat removal path/capacity	High	High	High
HCDA (Low Importance)				
	Re-criticality	High	High	High
	Energetic dispersal/reactivity shutdown			
	sodium voiding timing and coherence	High	Medium	Medium
	Fuel vaporization	High	Medium	Medium
	Mechanical energy generation	High	Medium	Medium
	Response of primary system to HCDA loads	High	Medium	Medium
	Response of containment to HCDA loads	High	Medium	Medium
	ultimate shutdown mechanisms	High	Medium	Medium
	ultimate heat removal path/capacity	High	Medium	Medium
	Hydrodynamics	High	Medium	Medium

As can be seen in Table 4, the state of knowledge about safety phenomena relevant to the SFR is extensive, judged to be 'high' in all cases for phenomena of high importance to the safety case. For phenomena of medium or low importance to the safety case, the judgment on the adequacy of knowledge reflects the acceptable higher uncertainty for less important issues. An example is for the HCDA phenomena, where the knowledge status is judged to be high, even though there can be significant uncertainties about the phenomena, because the importance of the HCDA events to the safety case is low.

Overall, no significant gaps have been identified that need to be targeted by R&D prior to proceeding with the development of an SFR and the assembly of the safety case, at least for more conventional technologies. There are a few relatively minor gaps in data and modeling, such as length effects for metal fuel in transient conditions and for fuel dispersal for both oxide and metal that could be addressed with transient fuel testing and additional modeling effort, as noted in Table 4. These could be addressed as part of an ongoing R&D program.

5. CONCLUSIONS

1. **There are no major technology gaps which 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 are no major 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 the planned reactor fuels and to complete severe accident testing to support safety analyses and NRC licensing discussions.

2. **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.

3. **Passive inherent features to provide 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.

4. **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 the pedigree nor have been collected and reviewed/evaluated in a manner needed to support the licensing of an SFR. There is a real possibility that this firsthand knowledge of the data and interpretation of the data may be lost if a knowledge management program is not implemented to capture this information base. This would include collecting and cataloging information which exists in log and data books, especially from facilities that are destined for D&D.

5. **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 which could be studied or developed to determine if they offer opportunities to improve the economics, safety and security of a sodium-cooled fast reactor. Although not needed to proceed with an advanced sodium-cooled fast reactor, as such they may be considered as “gaps” for further advances in development of these specific technologies. These are:

1. Advanced simulation of coupled neutronic/fluid flow dynamics.
2. Supercritical CO2 power conversion.
3. High minor-actinide content fuel.

Appendix A

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Advanced Burner Reactor Sodium Technology Gap Analysis

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy

Reactor Campaign

Michael Corradini
Chair, University of Wisconsin, Madison
Ahti Suo-Anttila, John Hewson,
Tara Jean Olivier, Jesse Phillips
Sandia National Laboratories
Mitch Farmer
Argonne National Laboratory
George Flanagan
Oak Ridge National Laboratory
Shinya Miyhara
Japan Atomic Energy Agency

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DISCLAIMER

SUMMARY

An Advanced Burner Reactor (ABR) based on liquid metal cooled fast reactor technology is being evaluated by DOE to provide the capability to transmute actinides and enhance the long-term fissile fuel supply for fission reactors. An essential element in this evaluation is the development of the safety case and appropriate ABR licensing approaches. ABR safety will be integral to the reactor system design. Development of the safety case for the ABR requires evaluation of the status of the existing technology base and identifying where gaps exists and additional information is required. This report focuses on identifying Sodium Technology gaps. 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 a) defining the relevant accident scenarios and the safety relevant features and components relevant to sodium technology phenomena, b) identifying the key phenomena active in the scenarios, c) assessing the importance of those phenomena to the ABR safety case, d) assessing the knowledge level currently available to address these issues for licensing. The technology areas of 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 that must be considered in the evaluation of ABR reactor safety, when a sodium leak occurs. This work focuses on the phenomena that would exist after a leak occurs and does not focus on SFR inspection and leak detection technologies. Although these elements are part of the initiation of any sodium phenomena, this work assumes that location and extent of the sodium leak will be provided from the plant analysis. 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 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 (~1MPa) cause a dispersed sodium spray in a containment compartment, whereas leaks at lower pressures (0.1MPa) could be characterized with a jet-pouring mode of contact within a compartment.

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.

- 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 (FOM) 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 ABR gap analyses. Two refined evaluation criteria were identified by this expert panel:

- Radiological consequence criteria: dose at the site boundary, worker dose, radioactive inventory;
- 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:

<u>Sodium Spray and Jet Dynamics:</u> 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 prior to impact on a surface, in closing knowledge gaps for spray dynamics 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.

<u>Sodium-Fluid Interactions:</u> Carbon-dioxide, CO_2, is being considered for the power conversion fluid in advanced supercritical cycles for the Gen-IV sodium fast reactor. 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. Both supporting research and understanding for the fluid interaction between sodium-CO_2 is meager for operational as well as safety issues. Experiments and supporting analysis for Sodium - CO_2 interactions is needed to determine their safety significance given such advanced power conversion systems.

<u>Sodium Surface Pool-Fire on Inert Substrate:</u> Substantial research has already been carried out to quantify the gross behavior of sodium pool fires at a variety of scales ranging up to 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 is needed such as i) radiation heat flux from a burning pool, ii) overall pool mass burning rate with oxide crust present, iii) oxide crust behavior, iv) source term for sodium aerosols.

<u>Sodium-Cavity Interactions:</u>

<u>*Sodium-Liner Interactions*:</u> Experiments focused upon steel liner corrosion with various ratios of sodium metal, oxide, hydroxide, 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.

<u>*Sodium-Concrete Interactions*:</u> 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 model 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 might 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 knowledge existed, although two areas were identified as uncertain. This may be especially true when considering the effect of sodium aerosols on the mechanistic source term. The uncertainty in the agglomeration process of sodium aerosols and other aerosols coming from fuel and cladding can result in uncertainties in the mechanistic source term. The degree of importance of these uncertainties are better defined by the source term gap analysis team.

CONTENTS

FIGURES

TABLES

ACRONYMS

ABR	Advanced Burner Reactor
AFCI	Advanced Fuel Cycle Initiative
DOE	Department of Energy
FOM	Figure of Merit
FPHE	Formed Plate Heat Exchanger
LWR	Light Water Reactor
NRC	Nuclear Regulatory Commission
PCHE	Printed Circuit Heat Exchanger
PIRT	Phenomena Identification and Ranking Table
R&D	Research and Development
SFR	Sodium-cooled Fast Reactor
SNL	Sandia National Laboratories

REACTOR CAMPAIGN

ADVANCED BURNER REACTOR SODIUM TECHNOLOGY GAP ANALYSIS

1. INTRODUCTION

The U.S. Department of Energy (DOE) is developing the next generation nuclear power reactors as a long-term component for future US energy supply. This effort is part of a research and development (R&D) program that includes the Generation IV Nuclear Energy Initiative and the Advance Fuel Cycle Initiative (AFCI). The Generation IV Nuclear Energy Initiative is based around the development of new reactor systems to be deployed during the next 20 years. AFCI's mission is to close the nuclear fuel cycle, optimize the use of fissile resources, and minimize the volume and longevity of the spent fuel waste. The safety of the next generation of nuclear power plants and associated facilities is essential in this R&D program and is the focus of this current short-term study.

Closing the fuel cycle requires the capability to reprocess the spent fuel to recycle the remaining fissile materials as well as transmute the long-lived transuranics that dominate the nuclear waste long-term radio toxicity. These transuranic bearing fuels can be burned in a fast reactor spectrum to not only extract the additional energy available, but to convert the long-lived radioactive species to fission products with much shorter half lives. An Advanced Burner Reactor (ABR) based on liquid metal cooled fast reactor technology is being evaluated by DOE to provide this capability. An essential element in this evaluation is the development of the safety case and appropriate licensing approaches for the ABR. ABR safety will be integral to the design of the reactor systems since safety should be an integral part of the design. Development of the safety case for the ABR requires evaluation of the status of the existing technology base (both experimental and simulation) and identifying where gaps exists and additional information is required.

2. STUDY OBJECTIVES

The technology base (both experimental and modeling) needed to demonstrate the safety, safeguards and security of commercial fast reactors for licensing is being assessed in a series of focused ‘Gap Analyses’. The work is aimed at evaluating the existing experimental and modeling databases (both domestic and international) and conduct an analysis to identify areas requiring augmentation of key test data or models.

A Gap Analysis is focuses on a specific subset of safety technology knowledge and experience with a goal of identifying and prioritizing knowledge gaps that require work. The Gap Analyses for the ABR are being organized in the following topic areas:

- Accident initiators/sequences,
- Methods and data, fuels and materials,
- Source term, and
- Sodium technology phenomena.

An expert elicitation process is used as the primary tool in this assessment.

For this Sodium Technology Gap Analysis, the panel experts gathered and reviewed the available pertinent information, prioritized the dominant phenomena, identified and ranked gaps in the database, and proposed approaches to close these gaps. The functional objective of the 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 information developed in the work should inform ABR safety evaluations including:

- Evaluation of the safety implications of ABR design options
- Identification of the high priority R&D needs to support ABR safety evaluation
- Inform the process of fully integrating safety in the ABR design activities.

3. GAP ANALYSIS APPROACH

The ABR is still in the early stages of development, and specific designs and fuel types have not yet been selected. It is therefore important to understand the range of safety relevant features and phenomena for these design options at the earliest possible stage, and determine if there are important gaps in our ability to analyze the safety case needed to license an ABR. To accomplish this objective, DOE initiated a series of sodium technology and fast reactor related Gap Analyses. The approach taken incorporates familiar features of a traditional Phenomena Identification and Ranking Table (PIRT) process incorporated to identify ABR safety relevant phenomena, evaluate the knowledge base, and rank potential gaps related in four specific areas of ABR safety technologies.

A PIRT is a systematic way of gathering information from experts on a specific subject, and ranking the importance of the information, in order to meet some objective, such as what has highest priority for research. PIRT processes are used in many fields where complex phenomena must be understood and modeled. It is an effective tool to consistently use expert assessments of safety relevant phenomena and identify areas of R&D needs. The PIRT process formalizes the approach for identifying the key phenomena that control the progression of safety significant events, and evaluating the state of capabilities needed to assess these phenomena. The PIRT process essentially draws on relevant existing information and expert opinion to identify the important phenomena that are active in the full range of safety scenarios and qualitatively ranks the importance of these phenomena and the current state of knowledge against a defined set of criteria. The PIRT is then a starting point for the R&D activities that will provide the information needed to inform development of the ABR safety case and licensing strategy.

PIRT’s are generally structured to address the scope and level of detail appropriate to the system being assessed. Evaluation of well developed designs or specific scenarios can be more narrowly focused, while assessment of more generic designs can be used to evaluate the safety implications of candidate options. Our gap analysis for the ABR safety areas, like Sodium Technology, focuses on the latter since it is relatively insensitive to ABR design details.

Liquid metal reactor technology has been sufficiently developed in earlier programs (e.g., Experimental Breeder Reactor, Fast Flux Test Facility, Clinch River Breeder Reactor, Power Reactor Innovative Small Module, in addition to foreign liquid-metal cooled fast reactor programs) to provide a partial basis for the identification of the important safety phenomenology, the status of our understanding of these phenomena, and gaps in our understanding for the ABR. Several design options for the ABR are being developed that include new features that can reduce the accident potential and improve behavior. Thus, a structured process should take into account any effects of the proposed ABR design options on identified accident phenomena. Gaps in the state of knowledge pertaining to each phenomenon will be identified and R&D programs can then be prioritized to address any open issues and provide the necessary data and analytical capabilities to define and support licensing.

It is important to the eventual licensing of an ABR that the gap analysis as well as further PIRT activities and results be accepted by the Nuclear Regulatory Commission (NRC). For that reason the gap analyses utilized for the ABR follow the general PIRT process utilized in the NRC Next Generation Nuclear Plant process [1], Fig. 1 below.

This effort required preparatory activities for each technology area to coordinate the compilation of the existing knowledge base, define specific panel study objectives and criteria, identify expert participants, and organize and coordinate the panels. Our Gap Analysis panel evaluation then involved a) defining the relevant accident scenarios and the safety relevant features and components relevant to sodium technology phenomena, b) identifying the key phenomena active in the scenarios, c) assessing the importance of those phenomena to the ABR safety case, d) assessing the knowledge level currently available to address these issues for licensing and e) documenting our work. Gaps or areas of inadequate understanding will be identified to define safety related R&D needs.

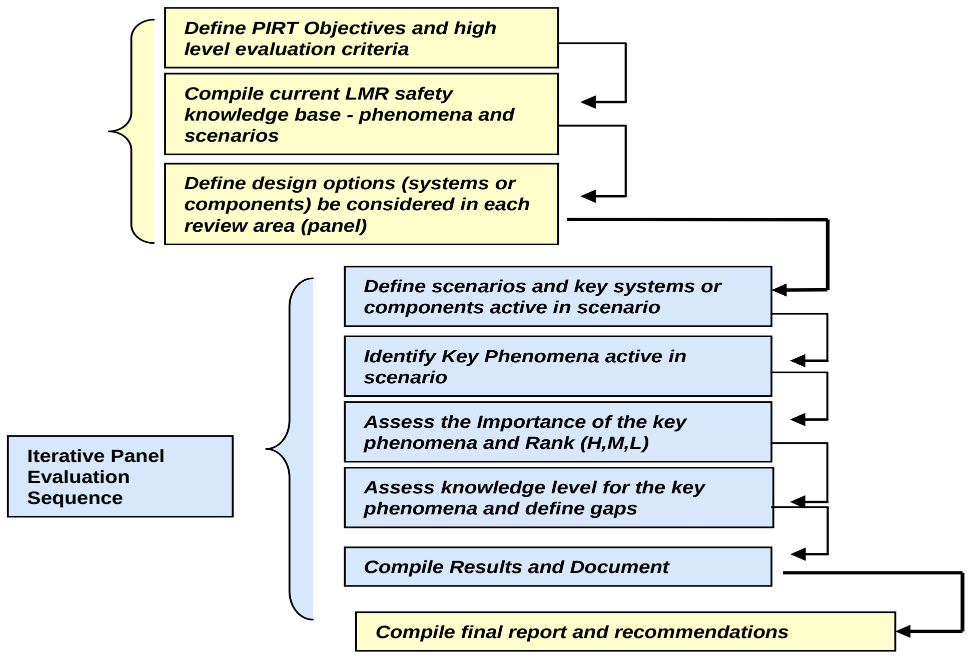

Figure 3-1. Sequence of gap analysis activities and panel process.

3.1 ABR Evaluation Criteria (Figures of Merit)

The most important evaluation criteria or figure of merit (FOM) for the phenomena being considered in these studies is radioactive material (fission products and other sources in the plant) release to the public. This must be a common element to all PIRT panel areas. Expanded or refined evaluation criteria were discussed by the Sodium Technology panel of experts and two general criteria were finally considered:

- Radiological consequence criteria: dose at the site boundary, worker dose, primary or secondary radioactive material inventory criteria;
- Functional criteria: potential impact on system or component operability or functionality under the accident scenario of interest (time at temperature, peak temperatures, neutron fluence levels, stress levels, boundary integrity, etc.)

3.2 ABR PIRT Importance Ranking

The importance ranking of ABR phenomena are evaluated according to the set of evaluation criteria (figures of merit) noted above. The importance ranking categories are qualitative levels of High (H), Medium (M), Low (L), and Uncertain (U). These rankings have been found in previous studies to provide adequate resolution and be consistent with an expert opinion process. The gap analysis results will be summarized in the form of a table, which includes comment sections for each ranking. In addition, the subsequent section of the report will provide details of the rationale or justification for the panel ranking. The general descriptions of these importance ranking levels based on the evaluation criteria are:

- High (H) – phenomena is of first order (fundamental) importance.
- Medium (M) – phenomena is of secondary (contributing) importance.
- Low (L) – phenomena not important for the scenario and the criteria considered.
- Uncertain (U) - potentially important, but insufficient information available to evaluate. Importance should be investigated further.

3.3 Knowledge Base Ranking

Evaluating the state-of-knowledge of a phenomenon generally involves the assessment of both the modeling capabilities and the database to validate the model. The panel discussed each phenomenon extensively during the evaluation with the general criteria for state-of-knowledge for each level of the assessment 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 adequate to validate relevant models exists, or the data is available to make an assessment.

Medium (M)

- A candidate model or correlation is available that addresses most of the phenomenon over at least some 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 or model applicability is uncertain or speculative.
- No existing database, assessments cannot be made reliably.

Uncertain (U)

- There is not adequate knowledge (available to the panel) to make an assessment.

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.

4. SODIUM TECHNOLOGY ACCIDENT SCENARIOS

The primary mission of the ABR program is to demonstrate the transmutation of transuranics recovered from spent fuel, and demonstrate the benefits of closing the fuel cycle as part of a larger effort in nuclear waste management as well as more efficient use of available fissile material. The transmutation or burning of the transuranics is accomplished by fissioning and this is most effectively done in a fast spectrum system. The ABR reactor and its system design should incorporate inherent safety features as well as utilize passive safety systems to the maximum extent possible. A prototype ABR will be developed as the first step in the commercialization of the ABR. Four industry groups are currently developing different design options for the prototype. A goal of the prototype will be to successfully demonstrate the major design goals of transmutation of transuranics in a fuel consistent with closed fuel cycle composition and requirements, and demonstration of the key safety features of liquid-metal-cooled fast reactors through the full range of reactor operations.

This gap analysis centers on the fast-spectrum sodium-cooled reactors. Sodium coolants add an additional dimension of chemical compatibility and reactivity phenomena that must be considered in the overall evaluation of ABR reactor safety.

Although the sodium compatibility/reactivity phenomena is not design specific, the likelihood and location of a sodium leak from the primary or the intermediate loop piping is dependent on the reactor design as well as dependent upon the leak-detection and inspection systems employed in sodium-cooled fast reactor (SFR). Our focus is on the phenomena that would exist after a leak occurs and does not focus on SFR inspection and leak detection technologies. Although these elements are part of the initiation of any sodium phenomena, this panel assumes that location and extent of the sodium leak will be provided from the plant analysis, either via a deterministic or risk-based assessment.

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. 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 into a compartment of the reactor containment;
- Sodium leakage from the primary or intermediate cooling system at low pressure into a compartment of the reactor containment;
- Coolant leakage (either water or supercritical CO_2) from the steam generator or heat exchanger into the sodium-cooling loop.

The distinction between high-pressure and low-pressure leakage was qualitative but centered on the recognition, that high-pressure leaks (~1MPa) could cause a dispersed sodium spray in the containment atmosphere whereas low-pressure leaks (0.1MPa) could be characterized with a jet-pouring mode of contact within the containment system.

5. SODIUM TECHNOLOGY GAP ANALYSIS RESULTS AND DISCUSSION

Given this classification of the accident scenarios we identified a group of general phenomena that were then subdivided into specific phenomena for discussion and ranking of their importance and their knowledge base. Seven general phenomena were identified by the panel:

- 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

Sodium spray phenomena include modeling and experiments related to the plume and spray dynamics (thermal-fluid dynamics, spray characteristics, including droplet size and velocity distributions, chemical combustion kinetics, and agglomeration phenomena). Sodium jet dynamics include phenomena such as jet breakup and mixing. Sodium-fluid interactions consider leakage events that take place in the intermediate heat exchanger where water or supercritical CO_2 may contact sodium. Sodium pool fire phenomena that are considered in this panel include radiation heat fluxes between the pool surface and environment, convection at the surface, development of the oxide crust, and sodium flow (spreading) issue as well as aerosol generation and dynamics, which is considered separately here. Sodium interactions with the containment cavity-liner and/or with concrete result in hydrogen production and aerosol generation. In severe accident sequences, sodium related phenomena may also interact with fission product chemistry and other accident phenomena, and therefore are to be accounted for in the overall accident analysis.

The complete tabular results of the gap analysis are provided in the appendix. In order to understand the panel rankings on importance and knowledge, we first summarize all the topical areas where gaps exist and then discuss the key gaps for each general phenomenon separately below. For our purposes here, a key gap is identified when the specific phenomenon is judged to be of 'high' importance for either high-pressure or low-pressure leakage events (or heat exchanger leakage), while being judged of having 'low' or 'medium' knowledge levels either in modeling or in experimental data. A summary of the key gaps identified by the panel is provided in Table 5-1.

5.1 Sodium Spray Dynamics

When a sodium leak occurs in a location elevated from horizontal floor-like surfaces, the leaked sodium may develop into a spray. The spray develops through the breakup of a liquid jet emanating from a leak. Spray scenarios are important because the breakup process exposes a relatively large surface-to-volume ratio for the leaked sodium. This large surface-to-volume ratio leads to greater exposure to the ambient atmosphere. In an oxidizing atmosphere, this will lead to faster oxidation and increased rates of associated heat release. In an inert atmosphere, the rate of sodium evaporation will be enhanced. Sodium oxidation also leads to the formation of oxide aerosols that can be transported substantial distances. These oxides are generally corrosive, might contain radionuclides and might allow electrical breakdown to occur where air-gaps would otherwise provide protection. The phenomena associated with the fate and consequence of aerosols is discussed primarily in the section on Aerosol Dynamics. The consequences of sodium leaks have been analyzed in terms of their potential thermal hazard associated with the damage of reactor-related equipment in the vicinity of the leak and in terms of the potential distribution of oxide aerosols that might be dispersed more substantially.

Table 5-1. Summary of key gaps identified by expert panel.

General Phenomena	Specific Phenomena	Importance		Knowledge	
		Hi-P	Lo-P	Mod.	Expt.
Sodium Spray Dynamics	Single Drop Particle Average Size	H	M	M	M
	Single Drop Particle Size Distribution	H	M	L	M
	Pre-Ignition Phase Dynamics	H	M	M	M
	Basic Evaporation/Combustion Models	H	M	H	M
	Crust Formation on Droplets	H	H	L	L
	Source of Sodium Aerosols	H	M	L	L
	Radiation Transfer with/from Aerosols	H	H	H	L
	Inertial Impact of Molten Sodium	L	H	M	L
	Burning of Droplet on Surface Sodium Pool	H	H	L	L
Sodium-Fluid Interactions	Fluid (CO2) Jet Leak into Sodium in Ht.Ex.	H	NA	L	L
Sodium Surface Pool Fire on an Inert substrate	Radiation Net Heat Flux	NA	H	H	L
	Mass Burning Rate	NA	H	H	L
	Oxide Crust Behavior on Pool Surface	NA	H	L	L
	Near Surface Aerosol Size/Distribution	NA	H	L	L
	Surface Aerosol Production	NA	H	L	L
Aerosol Dynamics	Sodium Aerosol Source Term	NA	H	L	L
	Hydrolysis of Peroxides	NA	H	M	L
Cavity Liner	Liner Failure Pressure or Thermal	NA	H	M	M
	Reaction Product Swelling Behavior	NA	H	L	L
	Corrosion of Liner	NA	H	M	M
Sodium-Concrete Interactions	Aerosol Source Term	NA	H	L	L
	Inert Concrete – Sodium Interactions	NA	H	L	M
	Basaltic Concrete – Sodium Interactions	NA	H	L	M
	Limestone (Carbonate) Concrete – Sodium Interactions	NA	H	M	M
	Sodium-Concrete Reaction with Sodium Fire	NA	H	L	M
Sodium-Melt-Concrete Interactions	Fission Product Dissolution and Partitioning in Melt and Gases	NA	H	M	M

5.1.1 Relevant Physics

Before addressing the specific scientific and technical gaps relevant for sodium spray dynamics, a brief overview of the expected characteristics is provided to put the discussion of hazards and technical requirements into perspective. Sodium leaks were considered to originate from pipes with moderate overpressures on the order of several atmospheres. At these moderate pressures, the leaking sodium is expected to break up gradually into droplets of millimeter scale. These droplets will be referred to as the primary droplets to differentiate them from aerosols of micrometer scale formed from evaporation, oxidation and re-condensation. Millimeter-scale droplets are expected to follow ballistic trajectories depending on their initial velocity and gravitational acceleration; they are not expected to be appreciably influenced by moderate drafts or by buoyancy. The micrometer-scale aerosols are only slightly affected by initial velocity and gravity; they are expected to be strongly dispersed by drafts and buoyant plumes.

The millimeter-scale primary droplets evaporate and oxidize, but completion of this oxidation process is only expected after droplets fall on the order of meters (3-30 m would be reasonable distances to complete this process). Thus, it is expected that this is only partially completed before the droplet impacts a surface. A substantial leak will lead to the formation of a pool of sodium on a surface, and the phenomena associated with that pool must also be considered. This connects the phenomena identified in this section with that described in the sections on Sodium Surface Pool Fire on Inert Substrate and Sodium-Concrete Interactions.

5.1.2 Identified Gaps

Gaps identified by the panel relevant to sodium sprays are described in this section. It is assumed in the discussion that sodium releases occur in oxidizing environments, so pure evaporation is not directly addressed.

The size of droplets that occur in a sodium leak is identified as having high importance because the time for complete droplet oxidation (or evaporation) depends to leading order on the droplet diameter squared (the so-called d^2-law); equivalently, the burning rate is proportional to the droplet diameter [2-3]. The ability to predict average droplet sizes is rated at medium due to the existence of Weber-number based criteria for droplet breakup [4], but there are deficiencies for irregular crack geometries. More significant is the lack of knowledge in predicting the full distribution of droplet sizes. It is empirically known that droplet sizes can vary over an order of magnitude or more about an average droplet size [4-7]. A lack of knowledge about droplet size distribution has been identified as a major challenge in using existing data sets for sodium spray fires to test predictive capabilities. This lack of knowledge also impacts the ability to predict possible accident scenario consequences once predictive capabilities have been validated. It is recommended that this gap be addressed through an experimental program that focuses on characterizing the full droplet size distribution for sodium-spray-fire tests and for irregular crack geometries that might be relevant to accident scenarios. It is expected that droplet-size-distribution experiments could be conducted primarily with less reactive simulant liquids (i.e. water) because the knowledge exists to relate results through non-dimensional analysis.

Once droplets have formed, they undergo oxidation. Oxidation can be split into two phases: a "pre-ignition" phase, before a flame is visible, where surface oxidation predominates and a "combustion" phase where gas-phase oxidation predominates [3, 8-10]. This oxidation process is highly important because it drives the heat release and aerosol production in the spray mode and determines the quantity of sodium that ends up in a pool. The state of knowledge for oxidation models is medium to high [2] and data for model comparison is of medium quality [3. 11-12]. One exception to this occurs in the area of oxide-crust formation on droplets during burning. The importance of crust formation was identified as high by the panel and the state of knowledge and availability of data were both identified as low. The importance was called out by the possible crust inhibition of reaction between the sodium and oxidizers. While crusts are observed and the literature results point to changes in droplet heating rates there is also evidence that this does not significantly affect the burning rates [8-9]. The low Piling-Bedforth ratio (denoting a porous oxide crust) is often identified as mitigating the importance of crust formation. At the same time, the available data is sometimes difficult to reconcile, and this difficulty is sometimes attributed to the crust formation. It is recommended by the panel that a series of experiments similar to existing droplet burning measurements [3, 11-12] could be carried out to clarify this knowledge gap. Crust formation is also significant for sodium pool fires and is further discussed in that section.

Once a particle is burning, the products of oxidation are various oxides of sodium. Because sodium has a large heat of vaporization with a heat of combustion only a couple of times larger [8] oxidation occurs close to the surface. With oxidation occurring close to the surface, a fraction of the oxide product returns to the primary particle, forming oxide particles that are large enough to settle rapidly. The remaining fraction of the oxide products is observed to form an aerosol of micrometer scale particles. The source of aerosols, in terms of mass of aerosol produced per mass of sodium burn is obtained from this latter fraction, and this has a high importance relative to the metric of aerosol dispersion. Available literature measurements of this fraction vary widely (experimentally ranging from 10% to 90%) [8,13] and models to interpret or predict these measurements are lacking resulting in low knowledge adequacy for this phenomenon. In this area, well-controlled experiments are recommended in conjunction with a model-development effort. The panel suggested sample experiments that might advise and validate a useful model would include measuring aerosol sources from single suspended droplets and small pools of varying temperature and in varying oxidizing environments [8,13].

In addition to the aerosol dispersion metric, the potential thermal hazard is employed as an importance metric. Thermal hazards of greatest significance are (1) heat transferred from sodium deposited directly onto surfaces and (2) radiation heat transfer to surfaces not directly impacted by sodium. In the latter case, the aerosol optical properties (i.e. absorption and scattering coefficients) were identified as having high importance because of the potentially large volume fraction of aerosols in the atmosphere around a sodium leak. While models for radiation heat transfer exist [14], the aerosol optical properties required by these models were identified as a knowledge gap (low rating). There are several challenges identified in obtaining suitable data. One challenge is associated with the fact that the oxides are known to be chemically transformed across species including Na_2O, Na_2O_2, NaOH and $NaHCO_3$ with the latter species being the more stable. It is expected that the first two species are formed in flames; NaOH is formed over periods of seconds but probably not within the highest temperature portions of the sodium-air flame. The transformation to the bicarbonate may take days [15]. Therefore, when looking across a sodium flame aerosol, it is likely that a variety of aerosol species are observed. Some thought will be required in developing an experiment to elucidate the required optical properties.

Since the droplet burning times for millimeter-scale sodium droplets are long, it is likely that many droplets will impact surfaces. Further, some leaks will be transitional between a spray and a pour, resulting in larger droplets impacting surfaces. Droplets impacting surfaces are subject to freezing, spreading, bouncing and splashing. For very small droplets associated with higher pressure leaks there are literature data and some models available associated with ink-jet printing and metal-spray manufacturing techniques. Data and models are essentially nonexistent for larger droplets; some recent work in this area on water-slug impact is reported in [16]. Because a large fraction of sodium in many sodium spray scenarios is expected to impact surfaces, this is rated as having high importance with low knowledge adequacy for both models and data for low-pressure sprays and leaks.

After droplets impact surfaces, there is a question as to how much heat and aerosols they release through further oxidation. As above, a substantial fraction of any spray is expected to impact surfaces giving this high importance to the metrics selected. Available measurements and models for pool burning tend to focus on pools that are insulated from surfaces and that have reached an equilibrium temperature [17]. This phenomenon addresses sprays that deposit to a limited thickness for which substantial cooling of the deposited sodium is expected. Under these situations, a smoldering-like oxidation is sometimes observed in which an oxide crust forms and appears to successfully inhibit the rate of oxidation. The panel was unaware of any models or data in the literature making this a knowledge gap with low knowledge adequacy for both model and measurements. This phenomenon area links to the Sodium Surface Pool Fire on Inert Substrate and Sodium-Concrete Interactions sections.

5.2 Sodium-Fluid Interactions

5.2.1 Background

The SFR is the base technology for transuranics recycle and destruction in the ABR fuel cycle component of the AFCI. For the mission one critical SFR issue is development of an economic SFR design. Elements of the SFR system research plan include work on SFR design and safety and component design. Current sodium-cooled fast reactor designs involve an intermediate loop of sodium coupled to a Rankine-power cycle with water as the working fluid. In this design, water/steam leakage in the steam generator into the intermediate sodium loop has been a known operational safety concern since the 1950's. Because of that, sodium-water interactions within the confines of the steam generator is a well known process that has been designed for in all subsequent SFRs with intermediate loops. Considering recent SFR intermediate heat exchangers use of double walled tubes and pressure suppression designs as well as the extensive research into the water-sodium interaction, the safety concern for sodium-water interaction could be interpreted as having medium importance with substantial supporting model and data availability. Carbon-dioxide, CO2, is now being considered for the power conversion fluid in advanced supercritical cycles for the Gen-IV sodium fast reactor. Both supporting research and understanding for the fluid interaction between sodium-CO2 is not as well understood and is considered an important issue lacking in information. The primary focus of this discussion is to investigate the extent and characteristics of the sodium interactions with CO2.

5.2.2 Identified Gaps

To improve the advanced design of SFRs power ratings and cycle efficiency, supercritical CO_2 is being considered as the working fluid for the power conversion unit. There have been several studies that suggest that the operation of CO_2 cycles near its supercritical pressure and at or above its supercritical temperature can result in improved efficiency over traditional Rankine steam-water power cycles or helium Brayton cycles. However, there is a considerable lack of knowledge as well as a corresponding need to better understand accidental contact between sodium and high-pressure CO_2. Elimination of the intermediate loop and using advanced heat transfer equipment could create a more cost competitive design. Printed circuit heat exchangers (PCHE) are being considered for CO_2 heat exchange with a range of fluids and it has been proposed to employ this type of heat exchanger design in the intermediate loop between sodium and CO_2. This type of design can result in a more compact heat exchanger and improved economics in the overall SFR design.

Specifically, PCHE or formed plate heat exchangers (FPHE) are essentially composed of monolithic blocks of alloyed metal, containing embedded narrow flow channels (millimeter length scales). One potential advantage of such a design for a sodium-CO_2 heat exchanger would be the potential elimination of a large tube rupture that may inject the working fluid into the liquid metal. Such a liquid metal chemical reaction had been a concern in liquid metal systems for advanced reactor designs. In these safety studies, the working fluid (e.g., steam-water mixture) may be injected through a tube rupture into a flowing stream of liquid metal (e.g., sodium or lead). Although this has been a safety concern, double-walled intermediate loop heat exchanger have solved the technical problem but have made the sodium-cooled reactor system not as cost-effective as first considered.

However, for the PCHE or FPHE with their small channels, it appears the potential for catastrophic failures involving massive sodium-CO_2 intermixing, extreme local pressurizations and attendant temperature rises with tube-tube failure propagation (such as might occur in a shell-and-tube heat exchanger) may not exist. Nevertheless, localized failure leading to fluid leaks between the flow channels cannot be ruled out and their uncertainty and lack of experimental knowledge remains high. Thus, sodium-CO_2 reactions involved in such inter-channel leaks would likely impact the operational performance as well as the related safety of the sodium-CO_2 heat exchangers and needs to be investigated.

Inter-channel leaks in a PCHE would involve the penetration of high-pressure CO_2 into the low-pressure sodium flow channels and subsequent contact between sodium and the CO_2. Some key technical issues need to be considered for such a postulated event; i.e., heat exchanger degradation and pressurization of the intermediate heat transport system.

The chemical reaction between the liquid sodium and CO_2 would likely produce sodium oxides and carbonates and elemental carbon in an exothermic reaction. These reaction products have extremely limited solubility in the sodium liquid and even if the rate of CO_2 injection was not a safety concern (due to energetics and heat release), the resultant chemical reaction products could be deposited as particulates or solids in the sodium channels, leading to degradation of the heat exchanger flow. This is primarily an operational issue associated with the heat exchanger performance; i.e., given the gas injection and the occurrence of the chemical reaction, the sodium channel flow would be inhibited by local deposits, further reducing the flow and allowing for deposits to be built up, causing a channel blockage.

If the reaction between sodium and CO_2 is limited, the unreacted CO_2 will be entrained in the sodium flow and accumulate in the intermediate heat transport system. This accumulation of residual CO_2 would tend to pressurize the intermediate heat transport system. If the PCHE interchannel leaks are massive, the pressurization and associated heat release could become significant. This would then be a safety issue that would require leak detection, pressure suppression and eventual sodium removal from the system to limit the long-term chemical reaction process and its consequences. Thus, key fundamental information is needed on the nature and extent of the sodium-CO_2 reaction.

Sodium is a Group 1 alkali metal that is chemically reactive with a great variety of other elements and chemical compounds; e.g., oxygen, water vapor, nitrogen, carbon-dioxide. In past designs of liquid metal reactor systems, both fission and fusion applications, safety issues involving liquid metal alloy compatibility have been a consideration when a liquid metal such as sodium is accidentally contacted by water vapor (steam), when it has been used as a working fluid in power conversion systems. This is because these reactions are exothermic and can lead to pressure and/or temperature excursions that may compromise equipment operability and plant system reliability.

Given the possibility of using supercritical CO_2 as a working fluid in the power conversion systems for advanced SFR designs, one needs to understand the thermodynamics and dynamics of possible sodium-CO_2 interactions. 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.

At a recent conference [18] (Global 2005) the possibility of sodium-CO_2 interactions was discussed. Based on that discussion and a review of the literature on liquid sodium chemical reactions, one can identify sodium-carbonate, Na_2CO_3; sodium-oxide, Na_2O; or peroxide, Na_2O_2; and carbon, C, and carbon-monoxide, CO, as potential reaction products. The primary reaction path appears to involve the formation of sodium-oxide and elemental carbon based on the following reaction path:

$$4Na + CO_2 \rightarrow 2Na_2O + C \quad (1)$$

This reaction is favored when the limiting reagent is carbon-dioxide, and analysis indicates that a theoretical adiabatic reaction temperature is in excess of 2000K. However, in the presence of excess CO_2, the sodium-oxide would further react with CO_2 to produce sodium-carbonate,

$$Na_2O + CO_2 \rightarrow Na_2CO_3 \quad (2)$$

Based on thermodynamic conditions, at high temperatures (e.g. higher than 700°C), carbon monoxide (CO) would be formed, possibly from the reaction between CO_2 and elemental carbon.

Very few experimental data on the sodium-CO2 reaction exist. The existing data has recently been reviewed by Latge et al [18]. The available data generally confirms the reaction product species identified by the thermodynamic analysis discussed above. It appears, however, that the extent and rate of the reaction would depend on a number of factors. Some important factors are:

- sodium temperature,
- sodium-CO_2 contact mode including the extent of mixing, and
- stability of the product layer at the sodium-CO_2 interface.

The experimental data (e.g., [19]) generally indicate the reaction rate increases with increasing sodium temperature. Since the reaction is highly exothermic, it is possible that the sodium temperature would increase during the reaction if the rate of heat generation exceeds the rate of heat loss. Such an increase in the sodium temperature would accelerate the reaction [20]. This possibility of reaction acceleration would depend on the sodium-CO_2 contact mode. The product layer at the sodium-CO_2 interface would have an inhibiting effect on the reaction. If the product layer is stable, the reaction would be controlled by diffusion of the reactants through the solid product layer. If the product layer is mechanically disrupted, fresh sodium surface would be exposed to the reacting CO_2 gas, thereby enhancing the reaction rate. The stability of the product layer at the sodium-CO_2 interface would largely depend on the contact mode between the sodium and CO_2 gas.

5.3 Sodium Pool Fire on Inert Substrate

This section of the ABR technology gap analysis addresses the issue of sodium surface pool fires on inert substrates. Pool fires on ‘reactive’ concrete surfaces are addressed separately in the last part of this section of the report. In most plant designs, the possibility of sodium-concrete interaction is mitigated by lining structural concrete surfaces with steel plating in areas where sodium leaks could occur. However, the possibility of sodium locally contacting the underlying concrete in areas where the plating may have failed due to thermal loading (e.g., bending, buckling), and/or at the edge of covered surfaces where the plate terminates, cannot be ruled out. Thus, this section also identifies a few key phenomena that would be relevant to this type of limited sodium-concrete interaction with the liner intact.

5.3.1 Background

A low-pressure leak in a pipe segment or component is identified as the primary initiator that can lead to the sodium pool fires. This type of sequence is differentiated from high pressure leaks that cause the sodium to be dispersed over a wide area in the form of a spray. Depending upon the particulars of the plant design, low-pressure leaks can develop in primary system piping, instrumentation feed through, valves or other ancillary equipment for loop-type designs, and in secondary system equipment of similar design for either loop or pool-type plants. In either case, thermal loading from the fire and the associated aerosol generation can inhibit plant operations, possibly even interfering with the ability to obtain a safe shutdown condition for the reactor. Moreover, for primary system leaks that lead to fires, radiation dose from activated sodium, in addition to the presence of fission products from failed fuel elements, can increase risk to plant personnel and possibly the environment. Thus, from a risk perspective, primary system sodium leaks are deemed to be significantly more challenging due to potential radiation dose coupled with the usual hazards associated with a sodium metal fire.

An example illustrating the effect of a sodium pool fire on plant operations is that which occurred at the Monju [21] prototype fast breeder reactor. As part of the startup procedures for this plant in 1995, a sodium leak occurred through a thermocouple well penetration in the pipe wall on the secondary (non-radioactive) outlet side of the intermediate heat exchanger. This led to a pool fire and subsequent shutdown of the reactor. The fire led to the formation of a ~1 m^3 mound of Na_2O on the steel floor, with aerosol deposited on the walls and floor of the facility. Clean-up and repairs were made using standard industrial techniques, but restart of the reactor was delayed by legal actions, court decisions, appeals, safety reviews, etc [21].

5.3.2 Identified Gaps

The Appendix presents the complete phenomena table, which summarizes the evaluation of phenomena that was carried out as part of the gap analysis for sodium pool fires. At the onset of this analysis, it is worthwhile to note that a significant amount of research has already been carried out to quantify the integral behavior of sodium pool fires at a variety of scales ranging up to cubic meters of sodium. These tests focused on quantifying gross behavior such as overall pool burn rate (viz. kg/m^2-hr) and aerosol concentration (viz. g/m^3) as a function of key parameters that include atmospheric oxygen concentration and relative humidity. In virtually all cases, these tests were minimally instrumented and initial and boundary conditions were not always well characterized. This data may be sufficient for quantifying the overall consequences of sodium pool fire phenomena. However, from the viewpoint of developing and validating advanced computational models that are increasingly being utilized to support reactor licensing applications, this collection of data is lacking in terms of the range and fidelity of information that is provided. This theme underlies a significant portion of the pool fire gap analysis that is outlined below.

The first phenomenon that was evaluated was radiation heat flux from a burning sodium pool. Depending upon the pool temperature and whether or not the pool surface has begun to cake over with a sodium oxide crust, radiation heat transfer can have either a medium or high (M-H) influence in determining heat losses from the pool surface to the surrounding environment. Models for radiation heat transfer are well developed (H). However, surface and aerosol optical properties are required input for these models, and these data are poorly known (L). Thus, this is an area where well-controlled experiments could be conducted to determine surface/aerosol optical properties during the combustion process using modern, commercially available instruments. Variations in surface conditions and the high temperatures involved will make the measurements challenging.

The overall pool mass burning rate is a second phenomena that is key to evaluating the consequences of a sodium pool fire (H). The burning rate is directly proportional to the thermal loading of the equipment enclosure, but also drives aerosol production that constitutes the ‘source term’ in the event of a primary system leak. The aerosols that are produced also can damage equipment, instruments, and cabling within the structure. Under high temperature pool conditions in which the oxide melts and sinks to the bottom of the pool, the knowledge state of the combustion models and supporting data are deemed to be high (H). However, under lower-temperature smoldering-type conditions in which the pool surface cakes over with an oxide crust, the knowledge level of models and supporting data are deemed to be low (L). The treatment of the oxide crust on the pool surface is felt to be of high importance (H), but the knowledge base in terms of modeling and supporting data on the crust behavior is deemed to be low (L/L). The convection and diffusion of oxygen though the crust that controls the burn rate is poorly understood. This is a second area where well-controlled and well-instrumented experiments are needed in which the overall pool burning rate is measured while concurrently documenting the pool surface conditions, including the presence/absence of surface crusts, and if a crust is present, the crust thickness vs. time. The influence/coupling of the crust behavior to the pool combustion kinetics is the knowledge gap in this area.

Closely linked to the pool burning rate is the source term for sodium aerosols. Aerosol production is of high importance in evaluating the consequences of a sodium fire (H). However, the state of mechanistic modeling of aerosol production is quite low (L). In most pool fire models, the aerosol production rate is a model input parameter that is specified on the basis of experimental measurements. The overall experimental knowledge base for aerosol production is deemed to be low (L). The information that is available is mostly order-of-magnitude estimates based on experiments that were minimally instrumented in order to characterize aerosol production. Moreover, the data are not well correlated to surface conditions on the pool as the combustion transient progresses. Thus, for the available data, it is difficult if not impossible to correlate aerosol production rate with pool surface characteristics, and this is the kind of information that is needed to guide model development and validation. This is another area in which well controlled and instrumented experiments are needed that quantify the nature and extent of aerosol release as a function of the pool burning rate and pool surface conditions.

Also closely linked to the aerosol production rate is the near surface size distribution of the aerosols produced, which is also deemed to be of high importance to support modern fire code development (H). Currently, the modeling knowledge base in this area is deemed to be low (L), with models typically requiring average particle size and constants for distribution partitioning as user input. Well instrumented and controlled experiments to characterize the size distribution at the "near-field" (i.e. pool surface) would be highly beneficial. The possibility might exist to use aerosol models and track back to predict the near field aerosol distribution, but this is not the desired solution. As an alternative to measuring the near surface distribution, the relative evolution of the aerosols as they travel away from the pool surface may also be useful.

Another ancillary phenomenon that was identified as part of the gap analysis was degassing product interaction with the sodium pool and the subsequent effect on the pool fire. An example of this type of interaction would be localized sodium contact with concrete through gap(s) in steel plating that covers a concrete floor. This phenomenon can be considered a subset of the broader area involving sodium-concrete interaction that is covered in the following section. Gas sparging during a sodium pool fire may act to break up any oxide crusts that start to form on the pool surface, thereby enhancing the overall pool combustion rate. On this basis, the relevance to pool fire phenomenology is deemed to be high (H). However, the status of knowledge in terms of models and available data is deemed to be low (L/L). In this area, it might be possible to modify interfacial crust formation models that have been developed to address the issue of ex-vessel Molten Core-Concrete Interaction for liquid water reactors (LWRs) [22]. Well-controlled experiments involving the injection of an inert gas into a burning sodium pool would be highly beneficial in terms of providing quantitative data on the effects of the sparging gas on stable oxide crust formation.

The above discussion has summarized areas in the field of sodium pool fires in which data knowledge gaps were identified by the committee. The appendix provides the complete summary of other phenomenology, that were discussed but were determined to be of low significance, and/or the existing knowledge base was adequate for addressing the phenomena. Areas of low significance were: i) conduction/convection on the interior of the sodium pool, ii) sodium to solid (inert) surface heat transfer, and iii) the behavior of gaseous products of metal reactions liberated from the combustion surface. Pool boundary effects (i.e., pool size) were deemed to be medium in importance, but sophisticated methods (e.g., various computational fluid dynamic codes such as STAR-CCM and FLUENT) are available for calculation single phase convective flow patterns in virtually any pool size for Newtonian fluids. The evaluation of film thickness (or spread area) as the material spreads from the leakage point was also deemed to be of medium significance, since the overall burning rate is directly proportional to pool area. However, spreading of molten pools has been extensively studied as a part of LWR severe accident research, and the possibility exists to leverage these models for the analysis of spreading of molten sodium pools [23, 24], if this is deemed to be necessary.

Finally, the effects of complex surface geometries on the pool burning process was identified as a potentially important area, but the committee was unable to determine the overall importance of this item and so it was classified as undetermined (U). An example of this is spreading of molten sodium on floor grating as occurred in the Monju leak [21]. Clearly, sodium spread as a thin film on surfaces can act to increase the overall burning rate simply by increasing the sodium surface area. Conversely, steel structure passing up through the overlying pool to the atmosphere may locally act as a cooling fin, and thereby reduce the rate of combustion.

5.4 Aerosol Dynamics

5.4.1 Background

In case of a sodium leak into the containment, there is the possibility the sodium will react with oxygen and form aerosols. These aerosols have the potential to impact equipment which could in turn have an impact on safety or plant reliability. The dynamic behavior of sodium aerosols will determine where these aerosols eventually end up and which systems are potentially vulnerable. In SFR safety studies, however, the greatest safety impact of sodium aerosols occurs during severe accidents.

In the unlikely event of a severe accident condition in a SFR, some of the metals in an SFR may become vaporized. Aerosols are formed when these vaporized metals (Na, Cs, Mn, Cr, Fe, U, Pu) typically found in an SFR are oxidized. Aerosol behavior is important to the transport of fission products and other highly radioactive materials because they impact the leakage into the environment. In general, the behavior of aerosols is important inside the primary heat transport system, in the containment building, as well as during atmospheric transport outside containment. The effect of aerosols on fission product transport is most apparent in the containment building.

In the event of a severe accident, large amounts of sodium combustion product aerosols will accompany any release of fuel aerosols into the containment. Experiments have shown that fuel and sodium aerosols co-agglomerate and form large non spherical “fluffy” odd shaped particles. The dynamic behavior of these co-agglomerated aerosols has the potential to affect the source term within and exterior to the containment.

5.4.2 Identified Gaps

A number of specific phenomena listed below have been identified that will impact the dynamic behavior of sodium aerosols, the agglomeration rate of these with other aerosols and their transport and interaction with surfaces. None of these phenomena were identified as important gaps in knowledge separate from other phenomena discussed before in sodium spray dynamics or pool fires.

- Source of sodium aerosol: This was determined by the gap analysis panel to be of high importance for any size sodium leak since knowing this will impact the overall aerosol dynamic predictions. The knowledge base for both experimental data and model adequacy was considered low. This specific phenomenon has been considered important and a gap that relates to sodium sprays and pool combustion and is discussed in detail in these sections.

- Themopheric transport: Temperature effects on aerosol dynamics occur principally in pool burning initiated by large leaks. Much experimental data exists both domestically and internationally and models generally are in agreement with each others and with integral experiments run at the Containment Systems Test Facility [25].

- Aerosol charging: Aerosol particle charging is known to affect water aerosols (water vapor effects and cloud formation). However, the degree to which electrostatic behavior affects sodium aerosols is uncertain. It is also uncertain if experiments or models exist to predict these effects. Literature searches have not indicated that such effects have been measured and current models do not account for these phenomena. Most are in good agreement with integral experiments, which indicates that charging is probably not a major contributor to aerosol dynamics.

- Electric Properties: Most data on electrical equipment malfunction and test results after exposure to sodium aerosols indicate that the effect of aerosols on electrical equipment is associated with plugging the equipment while there is no mention of shorting of the equipment [26]. Thus the effect of sodium aerosols due to equipment shorting is uncertain and data and models have not been developed. Experience in actual plants needs to be examined to determine if such effects have been observed. Small scale testing of various forms of sodium aerosols on electrical equipment could be done, as well as measurement of the sodium aerosol electrical properties.

- Turbulent Inertial Deposition: The impact of turbulent behavior of sodium aerosols on surfaces and the amount of deposition that results is not considered a major contributor to the overall behavior of sodium aerosols. At most it may play some role for those resulting from pool fires initiated from large leaks. Generally this data exists from previous sodium fire experiments and is adequately modeled.

- Gravitational Settling: Gravitational settling is the major removal mechanism associated with aerosol dynamics. It has been measured in some of the previous sodium aerosol experiments and most of the models are in agreement with integral experimental data.

- Interception Sweep out: This phenomenon does not appear to have significant effects on the overall behavior of burning sodium aerosols based on the data taken from previous experiments.

- Electrostatic Deposition: Experiments to date have not indicated that this phenomenon plays any significant role in deposition on surfaces.

- Aerosol Agglomeration: Sodium aerosols have been observed to agglomerate readily and to agglomerate with other metallic aerosols such as those resulting from fuel and fission products. The formation of larger and heavier particle leads to gravitational settling and could plays a major role in reducing the source term. Larger particles also tend to plug small cavities and reduce leakage from the containment. This phenomenon has been measured in numerous experiments and models have been developed that give good agreement with integral data.

- Hydrolysis of Peroxides: This process plays a large role in two ways. First is that it releases oxygen back into the systems allowing further oxidation and potential for fire and second the hydroxides tend to settle more readily. This is deemed to be important to predict the behavior of aerosols and some models take this chemical process into account. The process is a function of humidity present since sodium oxides and peroxides tend to grab the hydrogen out of the water molecule to form sodium hydroxide. Some of the experimental data exist where this has been examined under known conditions [26].

- Chemical Transformation of aerosol (hydroxide to bicarbonates): The chemical transformation of aerosols is also dependent on the humidity conditions. The transformation has shown to happen in 260s ranging from 90% at 50% transformation at relative humidity levels between 3% and 20%. This process should be considered for cases where the humidity levels could be high. Some data exists; however, most models do not consider this chemical process in determining aerosol dynamic behavior [26].

- Effective emissivity of a deposit layer: This phenomenon does not appear to play a major role in predicting the overall aerosol dynamics. Results of tests give little indication that this phenomenon is a contributor to overall aerosol dynamic behavior and therefore is not generally considered in most models.

- Re-suspension: This phenomenon does not appear to play a major role in predicting overall aerosol dynamics. Results of tests give little indication that this phenomenon is a contributor to overall aerosol dynamic behavior and therefore is not generally considered in most models

- Condensation/Evaporization to Aerosols: This process is humidity dependent, but based on experimental data it does not appear to be a major contributor to overall aerosol dynamics. Some data exists and some models take this into account.

5.5 Sodium Cavity-Liner Interactions

Cavity liner failure is normally considered a prerequisite for the initiation of sodium concrete reactions. The cavity liner may be composed of mild steel or of corrosion resistant steels depending upon design requirements. The cavity is vented to allow steam to escape in the event of a sodium leak. Normally this may be considered an adequate solution to prevent sodium concrete reactions, however there may be other factors involved that may circumvent the protection provided by the steel liner.

5.5.1 Background

In a large sodium leak scenario, significant heating of the concrete will occur due to conduction heat transfer through the liner and into the concrete. Concrete is composed typically of 7% water by weight, approximately 160 Kg/m^3. Thus there is significant potential to generate tremendous steam pressure behind the steel liner if the venting mechanism does not operate as designed. Long term steel corrosion (rust) effects or other plugging mechanisms may lead to steam venting failure. Failure of the liner due to pressure buildup is considered medium importance because modeling is considered adequate and is a structural-modeling and design specific issue. It is not considered of low importance because it is coupled to poorly understood concrete heat conduction - steam generation phenomenon. In addition if pressure buckling of the liner occurs the geometric deformation makes the structural modeling a highly non-linear effect.

In the event of a sodium fire, the reaction products are of sodium oxide (80%) and sodium peroxide (20%). These reaction products are combined with sodium hydroxide from the sodium-vented steam reactions and can cause steel corrosion effects that may be fast. An investigation of the Monju sodium leak incident revealed very fast steel corrosion effects, demonstrating that such effects cannot be ruled out. (In the Monju incident, the corrosion mechanism was considered “Na-Fe double oxidation corrosion reaction”. This would imply a slow corrosion rate under a low oxygen potential. On the other hand, the investigation revealed a “Molten salt type corrosion” indicative of very fast corrosion rates under high oxygen potential.) There is very little modeling known between steel and sodium mixed with sodium oxides, peroxides, and hydroxides at elevated temperatures. Sodium peroxide (from the fire) reacting with steam (from the concrete) releases oxygen that can corrode steel almost instantaneously due to high oxygen potential. Both sodium peroxide and steam can be present at the same location if the sodium metal is greater that 500C. At those temperatures sodium pool fire researchers have indicated that the oxide-peroxide-hydroxide slag that forms on the surface will sink to the bottom of the pool. If a crack is present in the liner floor, venting steam will be injected directly into the sodium oxide-peroxide-hydroxide mixture which may cause rapid melt through of the liner in that area. There is also the possibility of slow corrosion effects on the backside (concrete side) of the steel liner that could cause unseen cracks and thinning. Since a significant amount of water (160 kg/m^3) is present in concrete, and alkalis are also present in the Portland cement (and its additives), slow corrosion reactions may persist for the life of the reactor. These slow, many-year corrosion effects may cause significant liner degradation to the point where it does not provide an adequate barrier between sodium and concrete.

5.5.2 Identified Gaps

In the event of a crack, either present at the time of a sodium spill, or caused by thermal heating of the liner, sodium may leak into the gap between the liner and concrete surface. In such an event, sodium concrete reactions of at least the sodium hydroxide producing type will occur. The highly exothermic sodium-concrete aggregate reaction may or may not occur. Reaction product swelling, due to silicate and to a lesser extent carbonate reactions can cause significant pressures and forces that are sufficient to completely disrupt the liner and any supporting structures. This phenomenon is rated as high importance because it can cause early liner failure and there is little experimental data. Current sodium-concrete reaction models do not account for the effects of reaction product swelling.

Experiments that address liner failure due to reaction product swelling need to be large scale (>1m) since the mechanical force effects of swelling are proportional to somewhere between the square and cube of experimental scale. The experiments also need to address concretes made with various types of aggregates to establish what types are more prone to failures of this type. Silicate containing aggregates are known to exhibit significant reaction product swelling. Carbonate aggregates are also known to exhibit reaction product swelling though to a lesser degree.

Experiments with flawed liners also need to be made with a sodium fire present since the slag which forms contains an apparently highly corrosive mixture of sodium oxide-peroxide-hydroxide that can react with steam from the concrete resulting in highly corrosive hot oxygen gas. Such experiments need to be conducted at temperatures above the sodium oxide-peroxide-hydroxide melt temperature (500C) such that the oxide will sink and cover the liner crack. Pure sodium oxide Na_2O has a melt temperature of 920C. Pure sodium hydroxide NaOH has a melt temperature of 318C. Pure sodium peroxide Na_2O_2 decomposes at 460C. The chemistry and phase diagrams of a mixture of those components needs to be investigated to better understand some existing experimental results from sodium fires and sodium concrete reactions.

5.6 Sodium-Concrete Interactions

Sodium concrete reactions have been observed both experimentally and operationally at the ILONA facility. The experiments performed at Sandia National Laboratories (SNL) in the late 70’s and early 80’s indicated two phases of the reaction. Phase 1 is a quiescent phase that consists of sodium metal reacting with the water liberated from concrete to form sodium hydroxide and hydrogen gas. Phase 2 is a highly exothermic reaction between sodium and the concrete aggregate resulting in heat, hydrogen gas, and concrete ablation rates in the range 1-4 mm/min. The transition from phase 1 to phase 2 is poorly understood but seems to be related to sodium metal temperature, and to experiment scale. At temperatures above 500C phase 2 reactions have been observed. Phase 2 reactions may not occur or be reaction product limited in small scale experiments (0.3m diameter). Conversely, they have been observed at large scale (1m diameter) under the same conditions and are not limited by reaction product buildup.

5.6.1 Background

The severity of the reaction is related to the type of aggregate in the concrete. Silica (SiO_2) containing aggregates are the most reactive, followed by carbonates ($CaCO_3$), and alumina/magnesia oxides have the lowest chemical reactivity. The severity of the silica containing concretes is related to reaction product swelling, which mechanically disrupts any protective barrier created by reaction product formation. The continued disruption of the reaction product allows the sodium concrete reaction to continue unabated until one of the reactants, either sodium or concrete, is completely consumed. Limestone aggregate reactions are less severe primarily because the degree of reaction product swelling is less than with silica aggregates. Finally, alumina, magnesia, and other oxide ceramics are chemically inert with sodium metal, but may lose structural integrity due to sodium attacking the grain boundaries. The attack seems to be related to the amount and type of impurities in the material.

Sodium concrete reactions are known to produce large quantities of aerosols. The water in the concrete reacts with the sodium to produce hydrogen gas. The hydrogen will bubble through the liquid sodium and become saturated with sodium vapor. The bubbles will agitate the sodium pool and enhance sodium aerosol generation due to hydrodynamic bubble breakup at the pool surface. It can also transport any entrained aerosols due to other sources. Thus sodium concrete reactions will augment any aerosol production due to sodium fires.

Many experimental and code development studies have focused on the thermal and chemical processes of the sodium-concrete reaction. However more recently, a very limited amount of experimental aerosol source term data is available from Japan. Thus the aerosol production is rated as high importance because only limited data is available and there is very little modeling. Any future sodium concrete reaction tests should include aerosol measurements to provide additional data for modeling. Complications of such measurements are that what is collected during the experiment may continue to react and form chemicals that are significantly different from those which were collected.

5.6.2 Identified Gaps

Inert concretes have been developed, such as INTRACAST AS 701 –LaFarge Co., that are resistant to sodium attack. Such concretes are rated as high importance because of their ability to prevent a sodium concrete reaction, e.g. same effect as steel liner. However due to high expense, these concretes may only be a few to 10 cm thick when installed. Experimental data is lacking for such concretes exposed to sodium with an underlying reactive construction concrete. Experiments need to be performed with purposely flawed, cracked, of poorly mixed (fabrication error) concretes of this type with a highly reactive underlying construction concrete (i.e. silica containing) to determine long term structural integrity when exposed to liquid sodium at elevated temperatures.

Sodium concrete interactions with basaltic i.e. silica containing, concretes have been performed at large scale at SNL circa 1980. These tests indicated that reaction product swelling was responsible for cracking of the concrete crucible and continued attack by liquid sodium until the sodium was completely consumed. This type of phenomena is rated as high importance because reaction product swelling is not included in any of the sodium-concrete modeling. Data was obtained, but it was not completely understood.

New experiments should be focused upon Na - SiO_2 reaction rates and swelling effects. These experiments can be done at small scale to establish reaction rate data. Experiments on swelling effects are better at large scale (>1m) since data seem to indicate those effects are proportional to square or cube of experiment scale.

Sodium concrete reactions with limestone aggregates were also tested at SNL circa 1980. These tests indicated highly exothermic reactions do occur at large scale, but the experiments were often not repeatable, and therefore the phenomena was not fully understood. This phenomenon is rated as medium importance because sodium-limestone concrete models do exist that seem to match small-scale experiments.

Sodium concrete reactions with a sodium fire present lead to highly exothermic reactions in large-scale tests performed at SNL. The data indicated that test results were different with and without a crucible cover that excluded air from the sodium concrete test. The test results were not fully understood and may be related to the presence of sodium oxide and peroxide. The transition temperature to phase 2 (highly exothermic) concrete reactions was approximately 500C, which is also the temperature where sodium oxide-peroxide-hydroxide slag will sink or possibly dissolve into the liquid sodium.

In the case of sodium-concrete reaction in an air atmosphere, the sodium pool overlaying the concrete will generate heat due to the sodium pool fire. A part of the combustion heat will be transferred into the sodium pool increasing the pool temperature. As the result, the sodium pool will reach a transition temperature (threshold temperature) leading to severe exothermic sodium-concrete reaction at an earlier stage compared to sodium-concrete reactions in an inert atmosphere. In addition, hydrogen which is originally generated by the sodium-concrete reaction will also burn at the pool surface and resulting in water vapor which can react with sodium vapor (hydrogen-recombination). This additional combustion heat will also play a role in increasing the pool temperature together with the sodium fire combustion heat.

This phenomenon is rated as high importance due to a lack of adequate modeling, and poorly understood data. In addition there are two sodium combustion burning phases, smoldering and flaming. The addition of hydrogen bubbling from underlying concrete reactions to those combustion phenomena is not modeled or experimentally measured.

Sodium concrete reactions combined with a molten core have never been experimentally investigated. Core melt will add significant decay heat to a sodium leak scenario. Sodium concrete tests with electric pool heaters have been performed and they do confirm that concrete reactions do occur an often need auxiliary heat in order to progress. Core melt will almost guarantee steel liner failure due to the high temperatures involved. Once the liner has failed, sodium concrete reactions will occur and the additional decay heat will serve to increase the ablation rate and degree of penetration.

The effect of additional decay heat has been modeled in computer codes such as CORCON; thus that phenomenon is rated as low importance.

The decay heat is due to fission products in the core melt. There are no models that address the migration of these fission products into sodium metal, and sodium-concrete reaction products. Migration is expected to occur because hydrogen is bubbling and stirring the materials, and oxide cores may be soluble in the melted concrete and the sodium-concrete reaction products. This phenomenon is rated as high importance because there is very little data and virtually no modeling. One can partition the fission products based upon thermodynamic potentials, but the actual mixtures and eutectics that form are unknown.

Experiments can be performed at small scale to determine the solubility's of the various materials and fission products.

6. CONCLUSIONS AND RECOMMENDATIONS

The technology base needed to demonstrate the safety, safeguards and security of commercial fast reactors for licensing is being assessed in a series of focused 'Gap Analyses'. The work is aimed at evaluating the existing experimental and modeling databases and to identify areas requiring augmentation of key test data or models. In this work, we conducted a Gap Analysis on Sodium Technology Phenomena, which is a specific subset of fast reactor safety technologies, knowledge in order to identify and prioritize knowledge gaps that require further work. An expert elicitation process was used as the primary tool in this assessment.

For this Sodium Technology Gap Analysis, the panel experts gathered and reviewed the available pertinent information, prioritized the dominant phenomena, identified and ranked gaps in the database, and proposed approaches to close these gaps. A summary table of our gap analysis is provided in the Appendix. We summarize here is key actions as a path forward, for each of the general sodium technology phenomena areas:

- Sodium spray and jet dynamics
- Sodium-fluid interactions
- Sodium-pool fire on an inert substrate
- Aerosol dynamics
- Sodium-Cavity interactions (Sodium-Liner, Sodium-Concrete, plus melt)

6.1 Sodium Spray and Jet Dynamics

When a sodium leak results in the formation of 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 prior to impact on a surface, in closing knowledge gaps for spray dynamics an experimental program to understand relevant droplet size distributions is recommended. A related issue that can be addressed in concert with droplet spray sizing is in the prediction of liquid breakup when very large droplets impact surfaces and splash. Another important related gap to be addressed is related to the formation of oxides, both on droplet surfaces during the ignition phase and as an aerosol source during droplet burning – model development and experimental efforts could both be directed at better understanding the fate of oxides. The optical properties of the oxides are generally unknown and will affect the heat transfer from a spray fire to surrounding equipment; these properties can be also measured during these studies.

6.2 Sodium-Fluid Interactions

Considering recent SFR intermediate heat exchangers use of double walled tubes and pressure suppression designs as well as past extensive research into the water-sodium interaction, the safety concern for sodium-water interactions is adequately known with a significant experimental database. Unlike water, Carbon-dioxide, CO_2, is being considered for the power conversion fluid in advanced supercritical cycles for the Gen-IV sodium fast reactor. 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. Both supporting research and understanding for the fluid interaction between sodium-CO_2 is meager for operational as well as safety issues. We recommend the establishment of experiments and supporting analysis for Sodium - CO_2 interactions to determine their safety significance give such advanced power conversion systems.

6.3 Sodium Surface Pool-Fire on Inert Substrate

At the onset of this analysis, a significant amount of research had already been carried out to quantify the gross behavior of sodium pool fires at a variety of scales ranging up to cubic meters of sodium. This collection of information (i.e., test data and codes that were developed on the basis of that data) may be sufficient to support licensing activities for currently conceived fast reactor designs. However, the committee concluded that existing sodium pool-fire test data is lacking in terms of the range and fidelity of information necessary to support developing and validating advanced computational models that are increasingly being utilized to support reactor-licensing applications.

Specifically, a number of phenomena were judged to be key (of high importance and having a low knowledge level, especially in data) in evaluating the consequences of a sodium pool fire: i) radiation heat flux from a burning sodium pool, ii) overall pool mass burning rate with oxide crust present, including situations when inerting gas is present, iii) treatment of the oxide crust, iv) source term for sodium aerosols, v) near surface size distribution of the aerosols produced, and vi) degassing product interaction with the sodium pool. The phenomena of pool boundary effects, gas phase convective flux, and the evaluation of pool film thickness (or spread area) as the material spreads from the leakage point, were deemed to be of medium significance. Areas of low significance were: i) conduction/convection on the interior of the sodium pool, ii) sodium to solid (inert) surface heat transfer, and iii) the behavior of gaseous products of metal reactions liberated from the combustion surface.

6.4 Aerosol Dynamics

The dynamic behavior of sodium aerosols can have impacts on the safety performance of equipment and the source term associated with severe accidents. The subject of aerosol dynamics has been thoroughly studied by the liquid-metal reactor community and numerous experiments including large integral experiments have been conducted. Several modeling codes have been developed and these have been compared to experimental results. In general the results have been in agreement with experiments and if not, attempts have been made to understand the differences. The gap analysis team identified the important phenomena that affect the aerosol dynamics and concluded that no major gaps in knowledge existed, although two areas were identified as uncertain. It is recognized that improvements in modeling may require additional information in order to reduce uncertainty. This may be especially true when considering the effects of sodium aerosols on the ABR mechanistic source term. The uncertainty in the agglomeration process of sodium aerosols and other aerosols coming from fuel and cladding can result in uncertainties in the mechanistic source term especially as to the extent that plugging accounted for. The degree of importance of these uncertainties will be better defined by the source term gap analysis team.

It is recommended that the results from the Na technology gap analysis be integrated into the results from the source term gap analysis team in order to define any path forward

6.5 Sodium-Cavity Phenomena

6.5.1 Sodium-Liner Interactions

Cavity liner failure is normally considered a prerequisite for the initiation of sodium concrete reactions. Intact liner failure can occur from mechanical forces or possibly corrosion effects. Mechanical forces can arise from steam pressure buildup if the liner venting mechanism fails to perform adequately. Computational modeling of geometric specific designs should be able to determine the conditions where steam pressure buildup will cause liner failure. Corrosion effects may occur due liners coming into contact with corrosive slags produced in sodium fires and sodium steam reactions. Experiments focused upon steel corrosion with various ratios of sodium metal, oxide, hydroxide, and peroxide with steam present is needed to provide data for model development and to understand the complex chemistry. In connection with this work, failure of flawed liners can occur when sodium metal leaks behind the liner and reacts with the underlying concrete. Large scale (> 1 m) 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 due to this type scenario.

6.5.2 Sodium-Concrete Interactions

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 were not always reproducible. These experiments need to be conducted at large scale (>1m) because vigorous reactions were not always or rarely observed at small scale. The experiments need to be conducted with and without sodium fire present and included aerosol production measurements. Such experiments would also aid in further model development. Further model development needs to focus on silica containing concretes, and the effects of reaction product swelling and buildup and aerosol production. An additional series of experiments needs to focus on sodium resistant concretes. There is little data on what conditions could make such concretes likely to fail. Manufacturers have provided data indicating a high-quality product, but not all conditions were evaluated, especially those that may lead to failure such as cracked or incorrectly mixed or installed concrete.

6.5.3 Sodium-Concrete Interactions with Core Melt

Sodium concrete reactions with core melt are expected to enhance the rate of concrete ablation. Experiments could also to 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. These experiments, if deemed appropriate, should coordinate with aerosol source term work.

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APPENDIX A

SODIUM TECHNOLOGY PHENOMENA RANKING TABLE

Table A-1 Sodium technology phenomena ranking table

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
Sodium Spray Dynamics	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Prediction of Droplet Particle Average Velocity	M	L	NA	Affects vaporization rate, distance traveled before ignition and impact. What air flow and surfaces it interacts with is important as well.	H	H	Basic hydraulics are well known
	Prediction of Droplet Particle Velocity Distribution/Range	M	L	NA	Group with prediction of droplet particle average velocity. Possibly more important than average velocity. Burning rate is more dependent on droplet diameter. Velocity prediction is dependent on the system geometry, which affects its importance.	H	H	Basic hydraulics are well known

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Prediction of Single Droplet Particle Average Size	H	M	NA	Droplet size strongly affects the rate of evaporation and oxidation.	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

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Prediction of Droplet Particle Size Distribution/Range	H	M	NA	Possibly more important than average droplet diameter. The distribution is dependent on liquid break-up dynamics (splashing hi/lo pressure, particle collision). Initial jet break-up was separated from splashing during the discussion. The tails of the size distribution are important because small droplets burn rapidly, enhancing aerosol production, while large droplets do not always burn fully leading to surface impact.	L	M	Considerably more difficult to obtain size distribution data than average diameter data. Empirical correlation models exist for some cases. Only recent data is considered "good" data. Models will require data.

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Pre-ignition Phase before Combustion	H	M	NA	Lower pressure sprays are less likely to ignite before impacting floors, though this is dependent on geometry.	M	M	The Makino model is considered acceptable; the Morewitz *et al.* data is good.[a,b] The ignition delay is determined by the droplet surface temperature and drop size distribution.

[a] 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.

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General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Molecular Diffusion Coefficient within and across Diffusion Flame	H	M	NA	This phenomenon is part of the basic evaporation/combustion models. Pool models are also dependent on the diffusion coefficient. For low-pressure sprays molecular diffusion is likely to be less significant because of the delayed ignition.	H	H	
	Basic Evaporation/Combustion Models	H	M	NA	For low-pressure sprays this phenomenon is likely to be less significant because of the delayed ignition.	H	M	There is good data and models for a single droplet but data for interacting sodium droplets (spray) is sparse.

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Crust Formation on Droplets during Burning	H	H	NA	Equal effects for each scenario (Hi/Lo). There was a discussion as to whether the crust alters the burning rate by limiting diffusion through the porous medium or if the crust doesn't limit burning. Crust thickness is important. There are potential humidity effects that will alter the porosity and the burning rate.	L	L	Wick boiling is documented in the literature for sodium pools but data and models are lacking.

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
	Transition to Group Combustion Mode	L	L	NA	This is relevant when there is no oxygen penetration into the spray region due to vaporization of the sodium. For sodium the particles are expected to be large and the ratio of heat of combustion to heat of vaporization is small leading to slow vaporization. Because of this, group combustion should not physically occur during anticipated accidents and would require a fine droplet distribution.	M	M	

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
	Source for Sodium Aerosols (mass of aerosol released per mass of sodium burned)	H	M	NA	This is the mass partitioning for various products of oxidation between aerosols and association with original droplets. The distribution affects how much aerosol is produced and is more important for high pressure sprays (fine droplet distribution increases the oxidation rate). Current literature suggests a 10 to 90 percent range for fraction aerosolized.	L	L	There is no model known by panel and the data is sparse.

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
	Chemical Kinetics of Sodium Combustion	L	L	NA	Production ratios of sodium peroxide and oxide quickly equilibrate based on oxygen availability. Hydroxide and carbonate formation are slower.			
	Gas-Band Radiation from Diffusion Flames	L	L	NA	Emissivity of sodium.			
	Radiation from Aerosols in Diffusion Flame	H	H	NA	More important than the gas-band radiation (linked with the pool fire section).	H	L	Missing absorption/emissivity data.

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Sodium Particle Collision (Sodium-Sodium)	L	L	NA		L	L	
	Inertial Impact of Molten Sodium (splashing on surfaces)	L	H	NA	Particle breakup on impact. High pressure sprays breakup near source; low pressure sprays may not breakup substantially until impact.	M	L	Well known phenomenon. Data exists 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 better. (low-pressure spray has a low ranking for data and the high pressure spray has a high ranking for data)

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Burning on Surfaces (spraying onto surface)	H	H	NA	This phenomenon includes the chemical interactions following surface impact as well as potential cooling due to surface heat transfer. The specific chemical reactions depend on the type of surface. The oxide product layer is important. The deposit could create a product layer on equipment, which could affect its functionality. The majority of the spray may (depending on geometry and droplet sizes) burn on the surface	L	L	Models aren't good in their present form (need to expand) Data is almost non-existent.

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
Sodium Jet Dynamics	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Liquid Splashing on Solid	L	L	NA	Considered a stream more so than a spray would be less important than the pool in terms of the aerosol formation.			Due to the Low importance, – we will not be consider a potential gap as in this section. Geometry may allow for special cases with jet impingement leading to high importance. But sodium doesn't wet surfaces limiting this concern with regard to exposure time and jet duration.
	Liquid into Pool	L	L	NA	The jet could disturb the pool, but would be much less important than the pool in terms of aerosol formed.			

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
	Liquid Jets, Jet Breakup	L	L	NA	Creates more surface area in the gas phase.			
Sodium-Fluid Interaction	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Fluid Jet into Liquid Sodium	NA	NA	H		H/L	H/L	This general phenomenon is considered important but knowledge is good for sodium-water interactions and is lacking for sodium-CO_2 interactions.
	Sodium-fluid interaction	NA	NA	H		H/L	H/L	
	Sodium steam interaction	NA	NA	H		H/L	H/L	

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
Sodium Surface Pool Fire On Inert Substrate	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Radiation Net Flux from Pool Burning Surface	NA	M-H	NA	Dependent on whether smoldering or "burning". Off products quickly cool through conduction. Unknown material properties (john elaborate).	H	L	Models are good parameters are poor, low accuracy. (surface and aerosol optical properties) (opt. prop. linked with sprays)

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
	Mass Burning Rate	NA	H	NA		H/L	H/L	When at high temperature burning, the models are good. When smoldering (burning through the crust) the models are poor. Most experiment are using insulated surfaces. H for High temp L for Low temp (smoldering). Crust behavior is the gap. The mass burning rate is the unknown due to the crust. (refer to the crust treatment)

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Treatment of the Oxide Crust	NA	H	NA	When the crust thickens the burning rate slows.	L	L	Difficult to measure experimentally due to the low residence time of oxide to hydroxides.
	Gas Phase Convective Flux	NA	M-H	NA	Dependent on whether smoldering or "burning". Low temp convection dependency is higher in comparison.	H	H	Once through the crust, well known
	Liquid Phase Conduction/Convective Flux	NA	L	NA	Heat transfer through the liquid sodium. Thermal conductivity is very high.			
	Sodium to solid surface heat transfer	NA	L	NA	Pool to substrate. Pool depth can be limiting. Freezing effects			
	Near Surface Size and Distribution of Aerosol Particles	NA	H	NA		L	L	No good model (could assume a particle size and assume constants for partitioning; but linked to spray size distribution)

General Phenomenon		IMPORTANCE				KNOWLEDGE ADEQUACY		
	Source of Sodium Aerosols	NA	H	NA	(mass rate)	L	L	
	Damaged State (Complex surface geometry effects)	NA	U	NA	Changed to complex surface geometry effects	L	L	
	Film Thickness in Sodium Pool Spreading (Viscosity Issue)	NA	M	NA	Given the leak mass and pool spread will determine the burn rate. Discussed the Monju floor slope with regard to spreading sodium. Is it low volume/low flow rate?	H	H	Data available from LWR oxide and metal spreading experiments

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Degassing product interaction (most probable is steam)	NA	H	NA	Can keep the crust broken allowing continued burning. Depends on the sparging rate and critical velocity. (is there a liner?) can be linked to the liner discussion below	L	L	Interfacial effect at the crust is not well known and may be true for aerosol production
	Pool boundary geometric effects	NA	M	NA	(2d effects at the edge (convection cells near edge)	H	H	
	Thermal inertia effects (pool depth)	NA	H	NA	Pouring rate, initial cooling from surface.	H	H	
	Pressure Effect on Combustion (Vapor)	NA	H	NA		H	H	
Plume Dynamics (hot gas plumes)	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Momentum Transport (i.e. Velocity Field)	NA	H	NA	Where your aerosol goes	H	H	Consequences of aerosol transport

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Turbulence Production	NA	H	NA		H	H	
	Mixing (Turbulence Model), Oxidizer Transport	NA	H	NA		H	H	
	Temperature Distribution (Fluctuations)	NA	H	NA		H	H	
Aerosol Dynamics	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Source of Sodium Aerosols	NA	H	NA	We are assuming we have a defined source	L	L	
	Thermopheric Transport of Aerosols	NA	M	NA		H	H	
	Aerosol Particle Charging	NA	U	NA		U	U	
	Electrical Properties	NA	U	NA	Aerosols shorting out components	U	U	
	Turbulent Inertial Deposition	NA	M	NA	Turbulent impaction	H	H	
	Gravitational Settling	NA	H	NA		H	H	

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Interception/Sweep Out	NA	L	NA				
	Electro-Static Deposition	NA	L	NA	(to a surface)			
	Aerosol Agglomeration	NA	H	NA		H	H	
	Hydrolysis of Peroxides (going from peroxide to hydroxide)	NA	H	NA	Produces hydrogen and oxygen, increases in size, happens fast, significant consequences (will come back to this)	M	L	Hydrolysis may not be lacking data, but aerosol behavior is the key concern. There is "some" data.
	Chemical transformation of aerosol (hydroxide to bicarbonates)	NA	L	NA	On the order a days,			

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Effective Emissivity of Deposit Layer	NA	L	NA				
	Resuspension	NA	L	NA				
	Condensation/Evaporization to Aerosols	NA	M	NA	Depends on relative humidity after chemical equilibrium is reached for sodium aerosols	M	M	

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
Cavity Liner Failure	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Liner Failure Pressure	NA	H	NA	Perforation	M	M	Likely no composite model exists for liner failure (due to the complexity involved in the modeling process and the constraints necessary). Also there is little data for combined effects.

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Liner Failure Thermal	NA	H	NA	Combined thermal pressure effects leading to melt failure	M	M	
	Reaction product swelling	NA	H	NA	If we solved the Na-concentration interactions this could be solved.	L	L	

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Corrosion of Liner (hydroxide)	NA	H	NA	Initial consensus was that this takes a long time, but the rate experienced at Monju was very fast (molten salt corrosion)[a,b] . The catwalk corroded, but the liner didn't. Humidity can accelerate corrosion. Mild corrosion is possibly severe.	M	M	Data does exist. The Japanese did some steel immersed into sodium tests[c,d].
Sodium-Concrete Interaction	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Aerosol Source term	NA	H	NA		L	L	No known models

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General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Inert concrete-sodium interaction	NA	H	NA	Similar behavior as the steel liner to protect "normal" concrete.	L	M	Some testing has been done at Cadarache.
	Basaltic concrete-sodium interaction	NA	H	NA	Na_2SiO_3 disrupts concrete layer and swells causing continual degradation until sodium is consumed (linked with liner failure with swelling)	L	M	Low because no inclusion of swelling is in these models; Experiments have been performed but little confidence in whether the data is understood.
	Carbonate concrete-sodium interaction	NA	H	NA	Sodium Carbonate generates heat but the reaction products are denser than sodium liquid and can stop chemical interaction by layering on substrate	M	M	Did not understand the data, model adequacy was accurate for small scale tests.
	Sodium-Concrete Reaction with Sodium Fires	NA	H	NA	Sodium pool reaches transition temperature leading to severe exothermic sodium concrete interactions	L	M	Open chamber test and covered tests exist, additional testing may be needed

General Phenomenon		**IMPORTANCE**				**KNOWLEDGE ADEQUACY**		
	Hydrogen production	NA	H	NA		H	M	Ablation rate prediction is complicated
Sodium-Melt-Concrete Interactions	**Specific Phenomenon**	**Hi-P Leak**	**Lo-P Leak**	**HtEx Leak**	**Comments**	**Modeling**	**Expt'l Data**	**Comments**
	Enhanced heating	NA	H	NA	Decay heat, additional FP interactions	H	H	
	Fission product partitioning in the melt, oxides, and gases	NA	H	NA		M	M	Partitioning may not be accurately known
	Fission product dissolution into Na-Conc products	NA	H	NA	Forming eutectics with sodium silicate	M	M	
	Sodium Coolant Boiling	NA	H	NA		H	H	

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L. Walters, J. Lambert, K. Natesan, A. Wright, A. Yacout, S. Hayes, D. Porter, F. Garner, L. Ott, M. Denman

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

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Sodium Fast Reactor Fuels and Materials: Research Needs

Leon Walters, Panel Chair
Advanced Reactor Concepts
Idaho Falls, ID 8341 USA

John Lambert, Ken Natesan, Art Wright, Abdellatif Yacout
Argonne National Laboratory
Argonne, IL 60439 USA

Steven Hayes, Douglas Porter
Idaho National Laboratory
Idaho Falls, ID 83415 USA

Larry Ott
Oak Ridge National Laboratory
Oak Ridge, TN 37831 USA

Frank Garner
Radiation Effects Consulting
Richland, WA 99354 USA

Matthew Denman
Sandia National Laboratories
Albuquerque, NM 87185 USA

ABSTRACT

An expert panel was assembled to identify gaps in fuels and materials research prior to licensing sodium cooled fast reactor (SFR) design. The expert panel considered both metal and oxide fuels, various cladding and duct materials, structural materials, fuel performance codes, fabrication capability and records, and transient behavior of fuel types. A methodology was developed to rate the relative importance of phenomena and properties both as to importance to a regulatory body and the maturity of the technology base. The technology base for fuels and cladding was divided into three regimes: information of high maturity under conservative operating conditions, information of low maturity under more aggressive operating conditions, and future design expectations where meager data exist.

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This work was overseen and managed by Jeffrey L. LaChance (Sandia National Labs), who provided guidance on the approach taken, attended the expert elicitation panel meeting, and provided useful input during the report preparation. Recognition is also given to Tyrell Arment (MIT) for his assistance during the expert elicitation process.

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ACRONYMS

4S	Super-Safe, Small, Simple
ACRR	Annular Core Research Reactor
AFC	Advanced Fuel Campaign
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARC	Advanced Reactor Concepts
ASME	American Society of Mechanical Engineers
AT	Applied Technology
ATR	Advanced Test Reactor
CO_2	Carbon Dioxide
CRBR	Clinch River Breeder Reactor
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor-II
FCCI	Fuel Cladding Chemical Interactions
FCMI	Fuel Cladding Mechanical Interactions
FFTF	Fast Flux Testing Facility
FM	Ferritic/Martensitic
HFIR	High Flux Isotope Reactor
IAEA	International Atomic Energy Agency
IFR	Integral Fast Reactor
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
LMFBR	Liquid-Metal Fast-Breeder Reactor
LWR	Light Water Reactor
ORNL	Oak Ridge National Laboratory
OSTI	Office of Scientific and Technical Information
PIE	Post Irradiation Experiments
PNNL	Pacific Northwest National Laboratory
PRISM	Power Reactor Innovative Small Modular
R&D	Research and Development
SA	Severe Accident
S-CO_2	Supercritical-CO_2
SFR	Sodium Fast Reactor

SNL	Sandia National Laboratories
TREAT	Transient Reactor Test
U.S. NRC	U.S. Nuclear Regulatory Commission

I. OBJECTIVE

This document describes results of an expert opinion elicitation on research needed to ensure reliability and regulatory confidence of the potential fuels and structural materials to be used in the Sodium Fast Reactor (SFR). The expert opinion elicitation focused on a generic SFR design.

The purpose for the meeting of fast reactor fuels and materials specialists is to determine what R&D is required to license the two most mature fuels designs, metal and oxide fuel. Both fuel types have more than 50 years of experimentation and analyses the results of which are captured in voluminous publications and reports.

Expert opinions were elicited on the current state of knowledge for of the underlying phenomena affecting SFR fuels and structural materials performance. Experts were asked to rank these phenomena according to the:

- Importance of the phenomena with respect to regulatory and reliability concerns,
- State of experimental database, and
- State of current, quantitative understanding of the phenomena.

For this work, only nonproprietary, publically available data were used.

II. BACKGROUND

The two relatively mature fuel forms that have been considered for sodium reactor deployment are metal (U-Fs*, U-Zr, U-Pu-Zr, U-TRU-Zr) and oxide ($U\text{-}O_2$, MOX, $U\text{-}O_2/TRU\text{-}O_2$). Metal fuels were the standard fuel for Experimental Breeder Reactor-II (EBR-II) for 30 years of operation, were used as a partial core loading in the Fast Flux Test Facility (FFTF) , and were the choice for the PRISM reactor where a licensing case was prepared. Several of the recent reactor designs have chosen metal fuel for their reference cores such as ARC-100, 4S, and TERRAPOWER. Oxide fuels were the standard fuel for FFTF and were the choice for the Clinch River Breeder Reactor (CRBR). In Japan, France, UK, Russia, and China, oxide fuels were the primary choice for their sodium-cooled fast reactors. Thus, a large knowledge base exists internationally for fast reactor oxide fuels.

The original vision for each fuel type was to start the first fast reactors with uranium-based fuel and then reprocess the fuel and blanket with a transition to a mixed oxide core (uranium/plutonium oxide) or a uranium/plutonium metal alloy core. Reprocessed fuel gave another dimension to the work needed to understand fuel performance due to the carryover from reprocessing of fission products and minor actinides from reprocessing.

All of the fuel research programs tended to follow similar paths. Small scale experiments, both in and out of the reactor, were conducted to arrive at the best design in terms of cladding and fuel combinations. During the course of these studies the most important phenomena were discovered. Fuel restructuring, fission gas release, the extent of fuel-cladding-chemical-interaction (FCCI) and fuel-cladding-mechanical-interaction (FCMI), and the extent of cladding swelling, creep, and embrittlement are examples of some of the important phenomena. In an effort to analyze, understand and predict these phenomena, property data were required for input into models and calculations.

With the availability of EBR-II and FFTF in the USA, and test reactors in other countries, a wide range of design variables and operation conditions were studied for full-size fuel assemblies. The individual fuel pin should be viewed as part of a fuel system where fuel pin bundle interaction, bundle duct interaction, and duct-duct interaction are as equally important to the performance as the performance individual pins.

With the availability of full-sized irradiated pins, the use of transient test reactors, such as TREAT in the USA, allowed the study of the behavior of irradiated fuel when subjected to relatively-severe over-power and loss-of-coolant-flow conditions. Transient testing of irradiated pins in hot-cell furnaces provided important complementary information.

Thus, the past fuels programs all contained the elements of in-core and ex-core experiments for both steady-state and transient conditions as well as the analyses and modeling of the important controlling phenomena.

---

* Fissium (Fs) is nominally 2.4 wt% Mo, 1.9 wt% Ru, 0.3 wt% Rh, 0.2 wt% Pd, 0.1 wt% Zr and 0.01 wt% Nb, designed to mimic the noble metal fission products remaining after a simple reprocessing technique based on melt refinement

Over the past 50 years the progress in the development of fast reactor fuels has been continuous yet sporadic. During the 1960s and 1970s fast reactor fuel and reprocessing Research and Development (R&D) was intense worldwide with PUREX reprocessing and CRBR being the focus of U.S. programs that centered on oxide fuel. With the cessation of reprocessing, during the Carter administration, funding for these activities was greatly reduced. Fast reactor fuel development was revitalized in the 1980s with the introduction of the Integral Fast Reactor (IFR) program that included further metal fuel development and the pyro-reprocessing of the metal fuel. The IFR program was well-funded until 1994 when the Clinton administration curtailed fast reactor development with the closure of EBR-II and later FFTF.

During the latter part of the 1990s until today, interest reemerged in fast reactor fuel development not for breeding but with the realization that fast reactors are the best route to fissioning the minor actinides in commercial used fuel and thus to reduce the heat load and radio-toxicity of used fuel repositories. This new interest required understanding the performance of fast reactor metal and oxide fuels that contain a substantial concentration of minor actinides (Am, Np and Cm in particular). Much of the fuel development that is currently ongoing in the U.S. is involved with these issues.

When addressing issues associated with the gap analysis for the fuel, both oxide and metal, the knowledge gap becomes larger as the fuels are subjected to more demanding requirements. For example, UO_2 fuel or U-Zr fuel at up to 10 atom % (at%) burnup without plutonium, not reprocessed, no initial minor actinide addition, and at modest heat rating and temperature may require no additional R&D for licensing. However, more unresolved issues exist with reprocessed fuel that contains plutonium, minor actinides, and carryover fission products. Recent design requirements push the fuel systems to higher exposures than data exists for in the current suite of cladding and duct materials. These designs may reach regimes where excessive irradiation induced swelling, creep, and embrittlement become the controlling phenomena.

The resolutions of gaps which require the availability of a fast reactor test facility are difficult with no facilities in the U.S. and few in the world. The same is essentially true for safety related issues due to the unavailability of TREAT. The thermal spectrum Advanced Tests Reactor (ATR) reactor is somewhat suitable for special effects testing but falls short of irradiating full size fast reactor qualification assemblies.

Thus the subject gap analysis will be a graduated assessment that moves from the possible licensing of the basic oxide and metal fuel systems to the more complex systems where resolution of outstanding issues fall outside of the existing data base.

III. Expert Opinion Elicitation

In order to most efficiently direct future research efforts to create a licensing case for the SFR, an expert panel was assembled to identify gaps in the fuels and materials research areas which need to be filled before the Nuclear Regulatory Commission (NRC) can confidently license a sodium reactor. This panel's expertise covers both operational and experimental experience with fuel, fuel cladding, and structural materials. The panel briefly interacted via email prior to a 2.5-day meeting at Argonne National Laboratory to ensure that all relevant issues would be discussed in an orderly process. Figure 1 shows a high-level description of how the gap analysis was conducted. This approach was also used in the previous gap analysis reports (Corradini et al., 2010) (Sackett et al., 2010) (Powers et al., 2010) (Schmidt et al., 2011).

The degree of regulatory acceptability of the various fuel and materials issues was ranked qualitatively by the use of High (H), Medium (M) and Low (L) variables. High indicates that the regulatory body will require a high degree of confidence in the experimental database, materials knowledge or modeling techniques because the phenomena of interest can directly lead to a material failure. Medium indicates that the regulatory body will desire information about the phenomenon, but that the phenomenon is of secondary importance to understanding overall material performance and failure. Low indicates that understanding the phenomenon of interest is not instrumental to predicting material performance and thus only a basic understanding of the phenomenon is required for licensing.

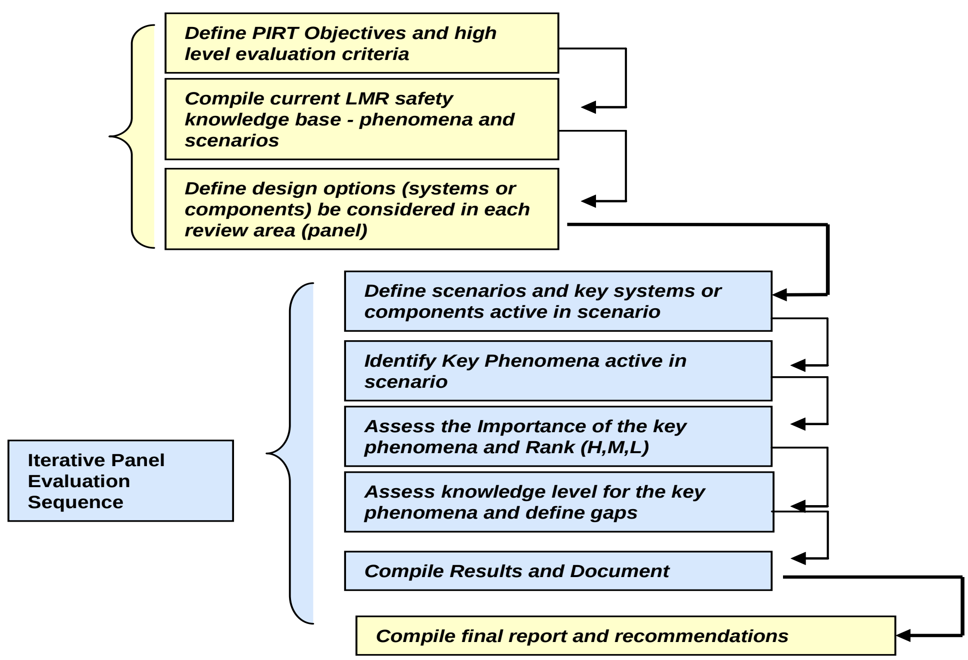

Figure 1. Sequence of gap analysis activities and panel process (Corradini et al., 2010).

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 panel discussed each phenomenon extensively during the evaluation with the general criteria for state-of-knowledge for each level of the assessment 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.

Uncertain (U)

- Information available to the panel was inadequate to assess the state of knowledge.

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.

IV. Evaluation of Gaps

The panel of experts opened the meeting with a general discussion of the state of knowledge for oxide and metal fast reactor fuels. Four fuel types were included in the discussion: fresh (which means not reprocessed) UO2, MOX, U-Zr, and U-Pu-Zr. The discussion was extended to include the additional issues associated with reprocessed fuel and fuel with additions of minor actinides.

The fuel performance issues that were addressed during the analysis included both steady-state and off-normal operation. Off-normal performance dealt with events that occur during steady-state operation such as load following and the behavior of fuel after cladding breach; i.e., run beyond cladding breach (RBCB) operation. In addition, the panel of experts was expanded to include experts knowledgeable in the area of accident-initiated events.

It was recognized that the performance characteristics of individual fuel pins which lead to cladding breach were not the only factor that determines the life of the reactor core. Fuel pin bundle interaction, which could restrict flow; bundle-duct interaction, which could affect fuel handling forces; and duct bow and dilation, which could affect both fuel handing forces and reactivity feedback were all concerns for core lifetime. However, the experts acknowledged that most of these concerns are design specific.

In the process of defining the important phenomena that impact fuel lifetime, the performance of cladding and duct materials may be controlling at high neutron irradiation exposures; i.e., high displacements per atom (dpa). Thus, in-depth discussion dealt with the state of knowledge for the current austenitic and Ferritic-martensitic cladding and duct materials as well as the use of advanced cladding and duct materials.

All the fast reactor test facilities in the US, which includes EBR-II, FFTF, and TREAT, have been shut down (TREAT, more specifically, being in “non-operational standby” status) for almost two decades. In the interim personnel have dispersed, and some records have been lost or stored in sub-standard conditions. This fact raised the question regarding how accessible and interpretable is the data base now and what condition has it been preserved for use in the future. This concern extended beyond irradiation data to the knowledge base for the procurement of materials and the fabrication of cladding, fuel, and duct components.

The expert panel first defined the Life-limiting phenomena for the fuel types at three burnup levels: 10 at%, 20 at%, and greater than 20at%. The rational for the three burnup levels was that for EBR-II (metal fuel) and FFTF(MOX fuel), the bulk of the irradiation data extended to about 10% because the exposure limits for the respective cores depended upon avoiding excessive fuel handling forces. For both oxide and metal fuel a limited number of assemblies were irradiated up to 20 at% burnup, but none beyond 20at%, even though some current reactor designs call for burnups greater than 20 at%. The state of knowledge was ranked according to the anticipated regulatory concern and then to the existing state of technical maturity and understanding. For metal fuel, virtually all the technical knowledge base belongs to the U.S.; whereas for oxide fuel, much of the knowledge base exists outside the US. It is uncertain how much of the foreign data would be available for U.S. reactor licensing.

Few if any knowledge gaps exist for fresh fuel of either oxide or metal fuel up to 10% burnup. However, the knowledge base is weak between 10 at% and 20 at% burnup and essentially non-existent beyond 20at% for fresh fuel. When reprocessed fuel was considered, the role of fission product carry-over, principally of lanthanide elements, resulted in identifying knowledge gaps. Further, if either oxide or metal fuel were to be utilized as hosts for minor actinides, then additional knowledge gaps were identified both with performance and fabrication.

Limiting phenomena were then identified for cladding and duct materials. The phenomena were ranked over four regimes: low dpa- low temperature, low dpa-high temperature, high dpa-low temperature, and high dpa-high temperature. High temperature was defined in this report to be up to than 630^{o}C for cladding and 580^{o}C for ducting. Low dpa was defined as less than 100 dpa and high dpa was defined as greater than 100dpa. Three materials were considered: 316 stainless steel, HT-9, and 9 Cr-1 Mo. By far the strongest data base for less than 100 dpa was 316, which is an adequate cladding and duct material for applications less than 100 dpa and nominal cladding temperatures less than 560ºC. There were few knowledge gaps for 316 under these conditions. However, void swelling would limit its application to exposures less than 100 dpa. Modified forms of 316 such as alloy D9, designed to improve resistance to irradiation effects, can perform to ~100 dpa.

The Ferric-martensitic alloy HT-9 exhibits low swelling up to about 200 dpa, but there are no data beyond that neutron exposure and not a great deal up to 200 dpa. HT-9 may be a reasonable cladding and duct candidate for either fuel type up to reasonably high burnup (20 at%) but only at cladding temperatures below 600 ºC due to the lower strength of HT-9 compared to 316 stainless steel. One potential gap identified for HT-9 is the lack of a vendor for the material. There appears to be an issue identifying a vendor who would be willing to become qualified to fabricate HT-9. Should a vendor be developed, then the question arises whether new heats of material would exhibit the same irradiation properties as the body of historical HT-9 irradiation data. This issue does not apply to type 316 stainless steel because this steel is a common fabrication material.

The martensitic material 9Cr-1Mo may solve both the issues of strength and swelling. However, less irradiation data exists for this class of alloys, and further, the qualification problem for potential vendors exists as is the case for HT-9.

After the Life-limiting phenomena were identified and ranked for fuel, cladding, and ducts, the important thermal and physical properties were identified and evaluated for gaps in knowledge by the same ranking system used for the phenomena. In general, the rankings mirrored the phenomena to which the property information applied. That is, for nominal fuel burnup and reasonably low dpa for the cladding and ducts, the property information was relatively well known. For higher exposures, gaps in the property information were evident.

The status of fuel modeling codes was discussed in an effort to identify gaps. The LIFE-metal code has been recently utilized routinely by a limited number of users. It was argued that the code does a reasonable job of describing existing irradiation data up to nominal burnups. The code is largely empirical and thus not useful for extrapolation to new operation regimes or new fuel designs. The main gap is that few people are knowledgeable enough to run the code and that substantial effort is required to document the code such that it can be transferred to new users.

The LIFE-oxide code (LIFE-4) is also being used routinely by a limited number of users and the documentation appears to be more complete than for LIFE-metal but some effort is required it to bring to current standards. For both codes documentation is required to describe the data base used to validate the codes.

A theme that ran through the entire meeting was whether or not the knowledge base for fast reactor fuels and materials was preserved intact. Fuel performance information is relatively available and retrievable through recent efforts to create computer searchable data bases. However, these data bases have stored publications and reports and not the original data. Whether or not the original data would be required in a licensing case was questionable. An attempt should be made to assess the availability and storage condition of original post irradiation data.

Operating information from EBR-II and FFTF is valuable to assess the performance of full assemblies. Duct bow and dilation measurements, assembly pull forces, and reactivity feedback information as a function of operating conditions are thought to exist at Pacific Northwest National Laboratories (PNNL) and Idaho National Laboratories (INL), but its location and condition need to be assessed.

Fabrication information for cladding, duct material, and fabrication information for both metal and oxide fuel exist in several locations. This information should be retrieved and assessed. Past practices would have to be duplicated to the extent possible for the existing data base to be valid for new fuels and materials.

Personnel capable of retrieving, assessing, and documenting outstanding information are ageing and leaving the workforce. Soon it will be nearly impossible to recognize and evaluate the value of existing data. Further, without the availability of testing facilities it will be impossible to duplicate subsets of the information that was generated over several decades and at great cost.

IV.A Presentation and Discussion of Rating Tables

The following tables were used as a means to first assess the importance of the various fuels performance characteristics to the regulatory licensing process and then assign a measure of the state of technologic maturity. For regulatory importance “H” indicated a characteristic of critical importance where the technologic maturity should also be “H” for successful licensing presentation. Where the regulatory importance was “H” and the technologic maturity was “L”, a definite knowledge gap exists. In the tables, the columns following the regulatory-concern column indicate the technology maturity levels.

The following categories were chosen for the tables as a means to envelop all possible licensing concerns.

1. Fresh metal and oxide fuel at 10 at%, 20 at%, and greater than 20 at% burnup.

2. Metal and oxide fuel with minor actinide additions at 10 at%, 20 at%, and greater than 20 at% burnup.
3. Metal and oxide fuel with carry-over of fission products from reprocessing at 10 at%, 20 at% and greater than 20 at% burnup.
4. Life-limiting phenomena and properties for 316 cladding.
5. Life-limiting phenomena and properties for HT-9 cladding.
6. Life-limiting phenomena and properties for advanced materials (e.g., 9Cr-1Mo or ferritic-martensitic steels).
7. Life-limiting phenomena and properties for 316 ducts.
8. Life-limiting phenomena and properties for HT-9 ducts.
9. Macroscopic thermal physical properties—metal UZr/UPuZr.
10. Macroscopic thermal physical properties—UO2/MOX

Table 1. Potential Life-Limiting Phenomena for Fresh Fuel

Fuel Phenomena	**Regulatory Concern, Metal/Oxide**	Metal, L.T. 10at%	Metal, L.T. 20at%	Metal, G.T. 20at%	Oxide, L.T. 10at%	Oxide, L.T. 20at%	Oxide, G.T. 20at%
Axial Growth	**L / (N/A)**	H	M	L	N/A	N/A	N/A
Fuel Swelling and FCMI	**H /M**	H	M	L	H	M	L
Gas Release	**H / H**	H	H	L	H	H	H
Fuel Constituent Redistribution	**M/ M**	H	M	L	H	M	L
FCCI	**H / M**	H	M	L	H	M	L
Fuel/Coolant Compatibility	**L / H**	H	H	H	H	L	L

Note: Experiment 496, a low smear density metal fuel test currently being irradiated, will increase our understanding of low smear density metal fuel

Table 2. Potential Life Limiting Phenomena for Fuel with Fission Product Carryover

Fuel Phenomena	**Regulatory Concern, Metal/Oxide**	Metal, L.T. 10at%	Metal, L.T. 20at%	Metal, G.T. 20at%	Oxide, L.T. 10at%	Oxide, L.T. 20at%	Oxide, G.T. 20at%
Axial Growth	**L / (N/A*)**	H	M	L	N/A	N/A	N/A
Fuel Swelling and FCMI	**H /M**	H	M	L	H	M	L
Gas Release	**H / H**	H	H	L	H	H	H
Fuel Constituent Redistribution	**M/ M**	H	M	L	H	M	L
FCCI	**H / M**	L	L	L	L	L	L
Fuel/Coolant Compatibility	**L / H**	H	H	H	H	L	L

*N/A – Not Applicable

Table 3. Potential Life Limiting Phenomena for Minor Actinide Bearing Fuel

Fuel Phenomena	**Regulatory Concern, Metal/Oxide**	Metal, L.T. 10at%	Metal, L.T. 20at%	Metal, G.T. 20at%	Oxide, L.T. 10at%	Oxide, L.T. 20at%	Oxide, G.T. 20at%
Axial Growth	**L / (N/A*)**	L	L	L	L	L	L
Fuel Swelling and FCMI	**H /M**	L	L	L	L	L	L
Gas Release	**H / H**	L	L	L	L	L	L
Fuel Constituent Redistribution	**M/ M**	L	L	L	L	L	L
FCCI	**H / M**	L	L	L	L	L	L
Fuel/Coolant Compatibility	**L / H**	L	L	L	L	L	L

*N/A – Not Applicable

Table 4. Phenomena and Properties for Stainless Steel 316 Cladding

Cladding Phenomena / Properties	**Regulatory Concern**	Low dpa (<100) / Low P.C.T.* (550-560^{o}C)	Low dpa / High P.C.T. (~630^{o}C)	High dpa (~200) / Low P.C.T.	High dpa / High P.C.T.
Creep Rate	**H**	H	H	IC***	IC
Swelling Rate	**H**	H	H	IC	IC
Fracture Toughness Properties	**L**	H	H	IC	IC
Yield Strength	**M**	H	H	IC	IC
Carbon Mass Transport	**M**	H	H	IC	IC
FCCI**	**M**	M	L	IC	IC

*P.C.T. – Peak Cladding Temperature, ** Only applicable to metal fuel, ,*** IC – Incompatible due to the poor high burnup performance of SS316 cladding.

Table 5. Phenomena and Properties for HT9 Cladding

Cladding Phenomena / Properties	**Regulatory Concern**	Low dpa (<100) / Low P.C.T.* (550-560^{o}C)	Low dpa / High P.C.T. (~630^{o}C)	High dpa (~200) / Low P.C.T.	High dpa / High P.C.T.
Creep Rate	**H**	H	M	H	L
Swelling Rate	**M**	H	M	H	L
Fracture Toughness Properties	**M**	H	M	H	L
Yield Strength	**M**	H	M	H	L
Carbon Mass Transport	**L**	N/A	N/A	N/A	N/A
FCCI**	**M**	H	M	H	M

*P.C.T. – Peak Cladding Temperature, ** Only applicable to metal fuel, ***N/A- Not Applicable
Note: Fabrication is not readily available, must be demonstrated to be consistent with historical HT9 database through mechanical and radiation testing.

Table 6. Phenomena and Properties for Advanced Cladding (e.g., 9Cr 1Mo, FMS)

Cladding Phenomena / Properties	**Regulatory Concern**	Low dpa (<100) / Low P.C.T.* (550-560°C)	Low dpa / High P.C.T. (~630°C)	High dpa (~200) / Low P.C.T.	High dpa / High P.C.T.
Creep Rate	**H**	M	M	M	L
Swelling Rate	**M**	M	M	M	L
Fracture Toughness Properties	**M**	M	M	M	L
Yield Strength	**M**	M	M	M	L
Carbon Mass Transport	**L**	N/A***	N/A	N/A	N/A
FCCI**	**M**	L	L	L	L

*P.C.T. – Peak Cladding Temperature, ** Only applicable to metal fuel, ***N/A- Not Applicable
Note: Fabrication is difficult but organizations claim that they can fabricate on an industrial scale.
Note: Japan and France have data from the phenomena/properties listed above, but it is unclear how available this data would be to a U.S. designer.

Table 7. Phenomena and Properties for Stainless Steel 316 Duct

Duct Phenomena / Properties f(dpa,T)	**Regulatory Significance**	Low dpa (<100) / Duct Inlet Temperature (400^{o}C)	Low dpa (<100) / Duct Outlet Temperature (550^{o}C)	Low dpa (<100) / Peak Duct Temperature (~580^{o}C)	High dpa (~200) / Duct Inlet Temperature (400^{o}C)	High dpa (~200) / Duct Outlet Temperature (550^{o}C)	High dpa (~200) / Peak Duct Temperature (580^{o}C)
Creep Rate	**M**	H	H	H	IC*	IC	IC
Swelling Rate	**M**	H	H	H	IC	IC	IC
Fracture Toughness Properties	**L**	H	H	H	IC	IC	IC
Yield Strength	**L**	H	H	H	IC	IC	IC
Carbon Mass Transport	**L**	H	H	H	IC	IC	IC
Dimensional Distortion	**H**	H	H	H	IC	IC	IC
Bundle Interaction	**H**	M**	M**	M**	IC	IC	IC
Bundle-Duct Interaction	**H**	M**	M**	M**	IC	IC	IC
Duct-Duct Interaction	**M**	H**	H**	H**	IC	IC	IC

* IC – Incompatible, ** If information has been preserved

Table 8. Phenomena and Properties for HT9 Duct

Duct Phenomena / Properties f(dpa,T)	**Regulatory Significance**	Low dpa (<100) / Duct Inlet Temperature (400^oC)	Low dpa (<100) / Duct Outlet Temperature (550^oC)	Low dpa (<100) / Peak Duct Temperature (~580^oC)	High dpa (~200) / Duct Inlet Temperature (400^oC)	High dpa (~200) / Duct Outlet Temperature (550^oC)	High dpa (~200) / Peak Duct Temperature (580^oC)
Creep Rate	**M**	H	H	H	M	M	M
Swelling Rate	**M**	H	H	H	M	M	M
Fracture Toughness Properties	**H**	H	H	H	M	M	M
Yield Strength	**L**	H	H	H	M	M	M
Carbon Mass Transport	**L**	H	H	H	H	H	H
Dimensional Distortion	**H**	H	H	H	M	M	M
Bundle Interaction	**H**	M*	M*	M*	M*	M*	M*
Bundle-Duct Interaction	**H**	M*	M*	M*	M*	M*	M*
Duct-Duct Interaction	**M**	H*	H*	H*	M*	M*	M*

* If information has been preserved

Table 9. Macroscopic Metal Fuel Thermal Physical Properties (UZr / UPuZr)

Physical Properties	**Regulatory Significance**	Low BU (<10%)	High BU (>10%)
Thermal Conductivity	**H**	H / H	L / L
Heat Capacity	**H**	H / H	L / L
Cladding Comp. Diffusivity	**M**	H / H	L / L
Free Energy of Formation	**L**	H / H	H / H
Phase Relationships	**H**	H / M	H / M
Primary Comp Diffusivity	**M**	M / M	L / L
Minor Actinide Diffusivity	**M**	L / L	L / L
Yield Strength	**L**	L / L	L / L
Thermal Creep Rate	**L**	L / L	L / L
Radiation Creep Rate	**L**	L / L	L / L
Young's Modulus	**L**	M / L	L / L
Thermal Expansion	**H**	H / H	L / L
Poisson's Ratio	**L**	M / M	L / L
Hardness	**L**	M / M	L / L

Table 10. Macroscopic Oxide Fuel Thermal Physical Properties (UO_2 / MOX)

Physical Properties	**Regulatory Significance**	Low BU (<10%)	High BU (>10%)
Thermal Conductivity	**H**	H / H	M / M
Heat Capacity	**H**	H / H	L / L
Cladding Comp. Diffusivity	**M**	IC / IC	L / L
Free Energy of Formation	**H**	H / H	L / L
Phase Relationships	**M**	H / H	L / L
Primary Component Diffusivity	**M**	L / L	L / L
Minor Actinide Diffusivity*	**L**	L / L	L / L
Yield Strength	**L**	M / L	L / L
Thermal Creep Rate	**L**	H / M	L / L
Radiation Creep Rate	**L**	H / M	L / L
Young's Modulus	**H**	H / M	L / L
Thermal Expansion	**L**	H / H	L / L
Poisson's Ratio	**L**	H / M	L / L
Hardness	**H**	H / M	L / L

*Note: Much of the high burnup data is Japanese or French in origin

Due to the unavailability of the proper expertise during the panel meeting, regulatory gaps in structural materials were not directly considered by the panel. Instead, this report leveraged a number of previous studies which considered the current state of SFR structural materials. Technology status evaluations for materials in various components and environments can be found in Tables 11- 15 (Chopra and Natesan, 2007).

Table 11. Reactor System Structural Components: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
A-1	Vessel	Stainless steel (Type 316)	Primary sodium and argon gas or air	Thermal aging Irradiation Weld integrity	Neutron fluence Service temperature Service life	Good	Sufficient	1
A-2	Vessel Enclosure	Stainless steel (Type 316)	Primary sodium & argon?	Thermal aging Irradiation Weld integrity	Neutron fluence Service temperature Service life	Good	Sufficient	1
A-3	Rotatable Plug for Reactor Head	?	?	?	?	?	?	?
A-4	Guard Vessel	Stainless steel (Type 316), Fe-9Cr-Mo Steel	Argon gas and leaking sodium	Corrosion	Temperature Exposure time Oxygen content in sodium	Good	Sufficient	1
A-5	Core Support Structure	Stainless steel	Primary sodium	Irradiation Thermal aging Crevice corrosion	Temperature Exposure time Crevice chemistry Oxygen content in sodium	Good	Sufficient	1

Table 12. Primary Heat Transport System: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
B-1	Internal Piping	Stainless steel (Type 316)	Primary sodium	Corrosion Carburization Radioactive mass transport	Temperature Delta T Oxygen and carbon in sodium Sodium flow velocity Sodium purification capability	Good	Fairly good Needs system assessment for ABR regarding dynamic carbon level	2
B-2	Mechanical Pump (Impeller, Diffuser)	Stainless steel (Type 316)	Primary (388°C) or Secondary (344°C) Sodium	Corrosion Fatigue resistance	Flow velocity Vibration Applied load Sodium purity Temperature	Good, need to identify vendors	Fairly good	2
B-3	Electromagnetic Pump	TBD	Primary (388°C) or Secondary (344°C) Sodium	Corrosion Fatigue resistance Electrical compatibility	Flow velocity Vibration Applied load Sodium purity Temperature Electrical interference	Unknown	Poor	3
B-4	Intermediate Heat Exchanger Shell	SS (Type 304 or 316)	Primary Sodium inside (510/355°C)	Sodium Corrosion Swelling Thermal Creep Irradiation Creep Fatigue and creep-fatigue Interstitial Element transfer	Oxygen and carbon in sodium Service life Temperature Mechanical load	Good	Adequate	2
B-5	Intermediate Heat Exchanger Tubes	SS (Type 304H), Fe-9Cr-Mo Steel	Secondary Sodium inside (333/488°C)& Primary Sodium outside (510/355°C)	Sodium Corrosion Swelling Thermal Creep Irradiation Creep Fatigue and creep-fatigue Interstitial Element transfer	Oxygen and carbon in sodium Service life Temperature Mechanical load	Good	Adequate	2

Table 13. Secondary Heat Transport System: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
C-1	Secondary System Pump	TBD	Secondary sodium	Corrosion Fatigue resistance	Flow velocity Vibration Applied load Sodium purity Temperature	Good, need to identify vendors	Fairly good	2
C-2	Sodium Piping	SS (Type 316)	Secondary sodium	Corrosion Carburization Radioactive mass transport	Temperature Delta T Oxygen and carbon in sodium Sodium flow velocity Sodium purification capability	Good	Adequate	2

Table 14. Power Conversion System, Supercritical CO2 Brayton Cycle: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
D-1.1	Compressor	TBD	CO_2, moisture, impurities?, high pressure	Oxidation Carburization Creep Fatigue Creep fatigue	Temperature Gas purity Applied load	Probably adequate	Limited	3
D-1.2	Turbine Generator	TBD	CO_2, moisture, impurities?, high pressure	Oxidation Carburization Creep Fatigue Creep fatigue	Temperature Gas purity Applied load Gas velocity	Unknown	Seriously lacking	3
D-1.3	Sodium to CO_2 Heat Exchanger	TBD	CO_2, moisture, high pressure	Oxidation Carburization Creep Fatigue Creep fatigue Sodium-CO_2 reaction	Temperature Delta T Gas purity Applied load Tube/channel failure Plugging Thin section material	Unknown	Seriously lacking	3
D-1.4	Recuperator	TBD	CO_2, moisture	Oxidation Carburization Creep Fatigue Creep fatigue	Temperature Gas purity Applied load	Adequate?	Probably good	2

Table 15. Power Conversion System, Steam Rankine Cycle: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
D-2.1	Steam Generator Shell	Ferritic Steel (Fe-2 1/4Cr-1Mo)	Secondary Sodium (502-344°C)	Sodium corrosion Interstitial element transfer Thermal aging Sodium-water reaction Caustic effect Thermal creep Fatigue & creep-fatigue	Sodium purity Temperature Delta T Steam leak Applied load Transients Long term aging	Good	Fairly good	2
D-2.2	Steam Generator Tubing	Ferritic Steel (Fe-2 1/4Cr-1Mo)	Water or Steam inside (287-482°C at ≈11 MPa) & Secondary Sodium outside (502-344°C)	Sodium Corrosion Interstitial Element transfer Thermal aging Sodium-water reaction Caustic effect Thermal creep Fatigue & creep-fatigue	Sodium purity Temperature Delta T Steam leak Applied load Transients Long term aging	Good	Fairly good	2
D-2.3	Hot Leg Steam Piping	Ferritic Steel (Fe-2 1/4Cr-1Mo)	Steam (482°C)	Flow-Assisted Corrosion Fatigue & creep-fatigue Thermal aging	Flow velocity Steam pressure Temperature Service time Thermal aging Water chemistry	Good	Fairly good	2
D-2.4	Cold Leg Steam Piping	Carbon Steel (SA 106 Gr. B)	Treated Water (287°C)	Flow-Assisted Corrosion General corrosion Fatigue	Flow velocity Steam pressure Temperature Service time Thermal aging Water chemistry	Good	Fairly good	1
D-2.5	Steam Turbine	Ferritic Steel (Intermediate chromium)	Steam	Steam oxidation Scale exfoliation Creep and fatigue Flow induced corrosion	Temperature Service time Applied load Steam flow velocity	Good	Fairly good	1

IV.B Discussion of Metal Fuels

Uranium-based metallic fuels have been used as the driver fuel for multiple SFRs, including the Experimental Breeder Reactor-I, the Dounreay Fast Reactor, the Enrico Fermi Fast Breeder Reactor, and most recently over 30 years of operation in the Experimental Breeder Reactor-II (EBR-II). Furthermore, the FFTF performed an extensive set of irradiations of metallic fuel qualification subassemblies and was poised to convert its core to a metallic driver fuel just prior to its shutdown. In EBR-II, U-10Zr metallic driver fuels operated reliably to 10 at% burnup, with extensive experimental testing of U-Zr and U-Pu-Zr metallic fuels to burnups of 20 at% conducted in both EBR-II and FFTF. It is not surprising, therefore, that the conviction of the fuels experts participating in this Gap Analysis was that the data base for a licensing case for metallic fuels (especially U-10Zr, but extending in large measure to U-20Pu-10Zr) is, in general, strong for burnups up to 10 at%, decreasing above that burnup level due to the reduced amount of experimental data.

IV.B.1 Life-limiting Phenomena

The major irradiation performance phenomena having the potential to limit the life, or the reliable performance, of metallic fuels are:

- Axial Growth
- Fuel Swelling & FCMI
- Gas Release
- Fuel Constituent Redistribution
- FCCI
- Fuel-coolant Compatibility

These irradiation performance phenomena were shown in Table 1 along with the consensus of the fuels experts as to their technological maturity level assessed from the perspective of regulatory importance. The rationale behind each score is briefly described in the subsequent paragraphs.

Axial Growth. Axial growth is of low concern from a regulatory perspective. U-Zr fuels can grow considerably during the first few at% burnup (i.e., as much as 10 at%), although U-Pu-Zr fuels exhibit considerably less growth. This phenomenon is more of an operational concern rather than a safety concern. Axial growth of metallic fuel is a source of negative reactivity for the core, for which reactor operations must be able to compensate. Extensive data have been collected on axial growth as a function of burnup in metallic fuels and reported in the literature (Hofman and Walters, 1994).

Fuel Swelling & FCMI. Metallic fuel is well known to be a high swelling fuel form under SFR conditions, with essentially no difference between U-Zr and U-Pu-Zr alloys. In the early days of metallic fuels, which were fabricated with either no fuel-cladding gap or a very small gap, fuel swelling quickly led to extensive FCMI resulting in fuel failure at low burnups (< 2 at%). However, it was eventually learned that this high swelling behavior is driven by rapid fission gas bubble nucleation and growth, which if allowed to swell unconstrained will result in an interconnection of bubbles and release of a large fraction of the fission gases produced in the fuel

(~80%). This interconnection phenomenon is strictly a geometrical effect that occurs at 33 vol.-% swelling, which for metallic fuels is reached at 2-3 at% burnup, after which continued accumulation of solid fission products drives further fuel swelling at a greatly reduced rate. Modern metallic fuel designs make use of a large fuel-cladding gap (i.e., 75% smear density) which allows this point of dramatic reduction in fuel swelling to occur prior to FCMI; with this design accommodation, FCMI typically does not threaten to limit metallic fuel lifetime until well over 10 at% burnup. Extensive data has been collected on metallic fuel swelling as a function of burnup in metallic fuels and has been reported in the literature (Hofman and Walters, 1994).

Gas Release. As discussed under the Fuel Swelling & FCMI heading, modern metallic fuel designs allow for early interconnection of fission gas bubbles, resulting in fission gas release values of approximately 80% above a few percent burnup. This leads to a need for a large fission gas plenum to accommodate such high fission gas release. Too small of a plenum can result in creep rupture of the cladding being the most significant Life-limiting irradiation performance phenomenon in metallic fuels, while too large of a plenum can be a significant economic penalty; this is a design trade off issue that can raise regulatory concerns. Nevertheless, extensive data has been collected on metallic fuel gas release as a function of burnup in metallic fuels and has been reported in the literature (Hofman and Walters, 1994).

Fuel Constituent Redistribution. Both U-10Zr and U-20Pu-10Zr metallic fuel systems undergo fuel constituent redistribution in the radial fuel dimension due to the fact that the traditional fuel temperature operating regimes span a miscibility gap in both alloy systems. The behavior is not identical in the binary and ternary systems, although both can result in radial zones having depleted Zr content under irradiation. Since the Zr content of the metallic fuel alloys is largely responsible for keeping the fuel solidus temperature high, this raises the regulatory concern that the solidus temperature is reduced in any Zr-depleted radial zone. The safety case for a metallic fuel must take this into account in demonstrating thermal margin under all operational scenarios. Data on constituent redistribution of irradiated metallic fuels is difficult to obtain, so less has been collected and reported in the literature than for most of the other phenomena discussed. Nevertheless, the theoretical understanding of the phenomenon seems to be well known, and several models have been developed that appear to adequately explain the reported experimental data (Hofman et al., 1996) (Kim et al., 2006).

Fuel-cladding Chemical Interaction. FCCI in metallic fuels results primarily from lanthanide fission products (i.e., La, Ce, Pr, Nd, Sm) that transport through the fuel and react with stainless steel cladding alloys. There is an incubation period associated with birth and transport of the fission products, after which the cladding reaction seems to follow a typical Arrhenius dependence on temperature. The reaction that occurs on the cladding inner surface produces a brittle interaction layer that grows with burnup and is generally considered as wastage. Thus, FCCI acts to thin the cladding wall, thus increasing the cladding stress, which must be accounted for in cladding creep rupture assessments. Nevertheless, it has not served to limit metallic fuel lifetimes for burnups to 10 at% and peak cladding temperatures less than 600°C. Extensive data has been collected on metallic fuel-stainless steel cladding chemical interaction as a function of burnup and has been reported in the literature (Keiser, 2009).

Fuel-coolant Compatibility. Fuel-coolant compatibility is a non-issue for metallic fuels in sodium-cooled reactors. Metallic fuels generally include liquid sodium as a thermal bonding agent in the fuel-cladding gap, and extensive run-beyond-cladding-breach testing for metallic fuels was performed in EBR-II. Metallic fuel is totally compatible with sodium (Crawford et al. 2007).

IV.B.2 Thermo-physical Properties

Since the licensing case for a nuclear fuel is made with considerable reliance on analysis and modeling of fuel behavior under reactor conditions, those thermo-physical properties needed for such analyses are very important. Typically, either experimental measurements or conservative assessments are required to support a safety or licensing case for any nuclear fuel. Table 9 shows the thermo-physical properties of importance in metallic fuel analyses, although they are not all of equal importance. The most critical are those properties that support the thermal analysis of fuel under irradiation and assessments of limiting conditions. For metallic fuels, these are: thermal conductivity, heat capacity, thermal expansion, and phase relationships. Thermo-physical properties beyond these are either of minimal regulatory significance or are easily estimated with adequate conservatism. In general, adequate knowledge of all the thermo-physical properties important to developing a licensing case for metallic fuels seems to be in hand for low (< 10 at%) burnups.

Thermal Conductivity. Knowledge of the thermal conductivity is vital for any nuclear fuel since calculated fuel temperatures are directly proportional to it. For metallic fuels, thermal conductivity is a function of alloy content, temperature, and burnup. Thermal conductivity for both U-Zr and U-Pu-Zr metallic fuels have been widely determined experimentally as a function of temperature and reported in the literature. Experimental determination of thermal conductivity for irradiated metallic fuels, though, has apparently been estimated in only one set of measurements (Bauer and Holland, 1995). Methods for determination of the thermal conductivity with burnup, which would appear to be conservative, have been reported. In any event, the relatively low temperature at which metallic fuels typically operate, with considerable thermal margin to the solidus temperature, means considerable uncertainty on the effect of burnup should be able to be accommodated.

Heat Capacity. Knowledge of the heat capacity of a nuclear fuel is needed for transient thermal analyses. Heat capacity for both U-Zr and U-Pu-Zr metallic fuels has been widely determined experimentally as a function of temperature and reported in the literature. Analyses have generally assumed that heat capacity does not change with burnup.

Thermal Expansion. Knowledge of the thermal expansion of a nuclear fuel is needed for both steady-state and transient thermal and thermo-mechanical analyses. Thermal expansion for both U-Zr and U-Pu-Zr metallic fuels have been widely determined experimentally as a function of temperature and reported in the literature. Analyses have generally assumed that thermal expansion does not change with burnup.

Phase Relationships. Knowledge of the phase relationships of a metallic nuclear fuel is needed for both steady-state and transient thermal analyses. Specifically, the solidus temperature as a function of alloy composition is taken as the effective melting temperature of a metallic fuel, and therefore represents a limiting condition from a regulatory perspective. Phase diagrams for both

U-Zr and U-Pu-Zr metallic fuels have been constructed with experimental validation and reported in the literature. Analyses have generally, but not always, assumed that the solidus temperature does not change with burnup.

IV.B.3 Notes on Metallic Fuels with Minor Actinide Additions

Interest in metallic fuels for actinide burning applications, for which Np and Am are incorporated up to a few percent into the fuel at fabrication, began in the early 1990's right at the time EBR-II and FFTF operations were terminated. Thus, there has not been extensive testing of metallic fuels with minor actinide additions. One experiment was performed in EBR-II prior to its shutdown that incorporated Np and Am into U-Pu-Zr metallic fuel (i.e., X501). It was irradiated to 8 at% burnup without failure. Post-irradiation examination revealed that considerable He gas was generated by the transmutation of Am and resulting decay chains, which was released at 90%. Radial redistribution of the Am, similar though not identical to Zr, was also observed. Fuel-cladding chemical interaction was apparently not affected in a measureable way by the presence of the minor actinides (Meyer et al., 2009). Nevertheless, gas release (i.e., He) will be increased and constituent redistribution will be affected by the addition of minor actinides to metallic fuels. While additional testing of metallic fuels with minor actinide additions is on-going in the Advanced Test Reactor, these tests are not entirely prototypic of a fast reactor environment and have not yet been fully assessed (MacLean and Hayes, 2007). Thus, it is acknowledged that the data is likely not currently in hand to license a metallic fuel with minor actinide additions (see Table 3).

IV.B.4 Notes on Metallic Fuels with Fission Product Carry-over

The primary concern with the licensing case for metallic fuels resulting from recycle using an electro-chemical process is the anticipation of carry-over of some lanthanide fission products, perhaps as much as 1 wt % in re-fabricated metallic fuels. As these elements are those primarily responsible for FCCI, the obvious concern is the FCCI could be accelerated for metallic fuels fabricated from recycle feed streams. This is an area of current research activities (Mariani et al., *in press*). Thus, it is acknowledged that the data is likely not currently in hand to license a metallic fuel fabricated using recycle feed streams (see Table 2).

IV.C Discussion of Oxide Fuels

As well as fueling all light-water reactors (LWRs) worldwide, uranium-based oxides have been used extensively as the driver fuel for several sodium fast reactors (SFRs) in Russia and Kazakhstan, including the BOR-10, BOR-60, BN-350, and BN-600 reactors; the latter 600-MWe SFR, for example, is currently in its thirty-second year of full power operation with stainless steel-clad UO_2 pellet fuel and with an enviable plant factor.

Plutonium-bearing mixed-oxide (MOX) fuel has been used as driver fuel in a wider range of SFRs: in the Southwest Experimental Fast Oxide Reactor (SEFOR) and the Fast Flux Test Facility (FFTF) in the U.S.; in the U.K. Prototype Fast Reactor (PFR); in the Rapsodie, Phenix and Superphenix reactors in France; in the German KNK-II reactor; and in the JOYO and MONJU reactors in Japan.

Additionally, extensive domestic fuels irradiation programs were performed in the Experimental Breeder Reactor-II (EBR-II) and the FFTF to determine Life-limiting phenomena in MOX fuel, including (in EBR-II) mild transient behavior of the fuel, and its potential for operating with breached cladding. Reviews of this U.S. work were given by Lambert and Strain (1994) and by Crawford et al (2007). It is no wonder, given this significant domestic and foreign experience, that fuels experts participating in this Gap Analysis were convinced that the case for licensing SFR oxide fuel is strong. Also, the domestic experience regarding the non-Pu-bearing material UO_2 is substantially less than with its Pu-bearing counterpart $(U,Pu)O_2$; however, the broad database and understanding developed for LWRs with UO_2 fuel can be extrapolated to SFR conditions without difficulty.

IV.C.1 Life-limiting Phenomena

Major phenomena that may have potential to limit reliable performance of oxide fuels are:

- Axial growth
- Fuel swelling and fuel-cladding mechanical interaction
- Fission gas release
- Fuel constituent redistribution
- Fuel-cladding chemical interaction
- Fuel-coolant compatibility

These irradiation performance phenomena are shown in Table 1 along with the consensus of the fuels experts as to their technological maturity level assessed from the regulatory viewpoint. The rationale for each score is briefly described.

Axial Growth: Axial growth nowadays is of low concern from a regulatory perspective. This was not the case in the early days of SFR development in the U.S. In fact, the SEFOR reactor was built specifically to check reactivity feedback in an oxide core (particularly the Doppler effect) and to determine the extent of axial growth in this comparatively new fuel type (Noble et al., 1972). In contrast to metallic fuels with substantial axial growth (≥10 at%), oxide fuels exhibited less than 1% change in overall length to a significant burnup. For this reason, axial growth is considered to be a non-applicable (N/A) phenomenon for oxide fuels.

Fuel Swelling and Fuel-Cladding Mechanical Interaction (FCMI): Oxide fuels are well known as medium swelling fuel forms, UO_2 and MOX exhibiting similar behavior. At temperature below about 1000^oC, the swelling of both fuels is in the region of 1.7% $\Delta V/V$ per at% burnup. At higher temperatures, where fission gas release occurs due to thermally-induced equiaxed and columnar grain growth, a value nearer 1.0% $\Delta V/V$ per at% burnup applies. Such fuel swelling can be partly accommodated by porosity in the sintered fuel and by additional porosity associated with unhealed thermal cracks in the fuel pellets and the residual fuel-cladding gap.

With judicious choice of fuel density and fuel-cladding gap size, the incidence of FCMI can be minimized in SFR oxide fuel elements to high burnup. For this reason fuel swelling and FCMI are deemed Life-limiting phenomena of medium regulatory concern. The experience level with these phenomena is considered to be high for burnups below 10 at%, medium for burnups between 10 and 20 at%, and low for burnups above 20 at% [being limited to data on a small number of examined fuel elements from the Core Demonstration Experiment (CDE) in the FFTF in which peak burnups of 24 at% were achieved (Bridges et al., 1993)].

Gas Release: As ceramics with low thermal conductivity, SFR oxide fuels operate at high temperatures and release significant quantities of the fission gas generated in them. At linear powers of 25-30 kW/m (and higher), fission gas release exceeds 50% at low burnup (≤5 at%), and increases steadily with increasing burnup as linkage of grain-boundary bubbles occurs in the fuel interior. Fission gas release is then typically 75% and higher.

This phenomenon is considered of high priority in licensing for the simple reason that if insufficient plenum volume is available stresses in the cladding can be high enough to cause significant creep and possibly lead to pin failure. The confidence level in the data on this phenomenon is considered high at all burnup levels. For example, at burnups above 20 at% where data are limited, a conservative value of 100% gas release can be assumed for licensing purposes.

<u>Fuel Constituent Redistribution</u>: UO_2 and $(U,Pu)O_2$ are single phase to their melting point, so that phase changes *per se* do not lead to constituent redistribution, which they do in metal fuels. However, redistribution <u>does</u> take place by vapor transport and is strongly affected by fuel stoichiometry. In hypostoichiometric MOX, i.e., $(U,Pu)O_{2-x}$ normally used in SFRs, Pu species tend to move up the temperature gradient and U species down the temperature gradient.

Such redistribution will result in mild radial changes in the heat production rate in-reactor. In turn, this can increase fuel centerline temperatures. Overall it is not a large effect but is one that needs to be addressed during licensing; it is considered of medium regulatory concern. The maturity level in knowledge of this phenomenon mirrors the distribution in the data: high below 10 at% burnup, medium above 10at % burnup, and low above 20 at%. Early work on this area was performed by Meyer (1974).

<u>Fuel Cladding Chemical Interaction (FCCI)</u>: FCCI is an in-reactor phenomenon once feared as Life-limiting in SFR stainless-clad MOX fuel elements. It is caused by the radial migration to the fuel-cladding interface of fission products cesium, iodine and tellurium, and oxygen freed by fission. Oxidative corrosion of the steel inner surface in the presence of fission products can then occur, either as a uniform reaction, or—under the right conditions—as attack along grain boundaries of the steel which have become denuded of $Cr_{23}C_6$ precipitates in-reactor, i.e., thermally sensitized.

Uniform FCCI can be considered as simple wastage or thinning of the cladding; grain-boundary FCCI is less predictable and potentially could lead to cladding failure, although none has ever been observed. It was discovered, however, that FCCI can be largely suppressed by lowering the initial stoichiometry of the fuel; oxygen-to-metal ratios of 1.94-1.95 will inhibit occurrence of FCCI to well beyond 10at % burnup. The phenomenon was much studied in the 1970s and 1980s, and is considered to be well understood; it has also been extensively reviewed(Lawrence et al., 1990). For these reasons FCCI is designated of medium regulatory concern. The maturity level in knowledge is high up to 10at % burnup, medium to 20 at% burnup, and low above 20 at% (post-irradiation examination of the CDE MOX elements from FFTF should soon alter the latter designation).

<u>Fuel-Coolant Compatibility</u>: Fresh UO_2 and $(U,Pu)O_2$ react directly with sodium to form sodium uranate (Na_3UO_4) or sodium urano-plutonate (Na_3MO_4, where M = U,Pu). The reaction takes place in the fuel-cladding gap and in open porosity and cracks in the fuel up to about 1000^oC (the

dissociation temperature of the reaction products). The reaction products have about half the density of the fuel they replace, so that local swelling occurs and a cladding breach may be extended. For medium burnup fuel, in which cesium uranate or cesium urano-plutonate has already formed at the fuel surface, the reaction is different: the chemically more active sodium replaces the cesium in the reaction phase and cesium is lost to the primary coolant; for these conditions local swelling is much less, although with brittle cladding a breach may still propagate.

Fuel-coolant incompatibility is thus considered a Life-limiting phenomenon of high regulatory concern; there is no analogous phenomenon for metallic fuels, which are entirely compatible with sodium. However, a 15-year program of run-beyond-cladding-breach (RBCB) testing of MOX fuel elements in EBR-II (Lambert et al., 1990) cast an interesting light on the phenomenon. Provided the swelling from fuel-sodium reaction does not cause unstable splitting of the cladding, the fuel-sodium reaction product, once formed, becomes a barrier (a "scab") to further reaction so that continued operation with the failed fuel is possible for several months, including shutdowns and startups. Because of experience from this RBCB program, knowledge of the phenomenon is considered high to 10 at% burnup, decreasing at burnups above 10 at%.

IV.C.2 Thermophysical Properties

Table 10 lists the thermo-physical properties of importance in analyzing oxide fuels for licensing purposes. Of highest regulatory importance are those properties supporting thermal analysis of the fuels under irradiation—thermal conductivity, heat capacity, thermal expansion, and melting temperature. Other thermo-physical properties are of lesser (medium or low) importance.

Being a universal fuel for LWRs, UO_2 has been thoroughly studied, particularly for its thermal conductivity. Recent compilations of thermo-physical properties of UO_2 were published by the IAEA (1997), and stored in the THERSYST system at Stuttgart University; and by the NRC (2011), and stored in the MATPRO database maintained for the U.S. industry. The IAEA report also included SFR MOX fuel. Among many other reviews, the most succinct comparison of the thermophysical properties of UO_2 and MOX was given by (Carbajo et al., 2001).

It was concluded by the Gap Analysis experts that the thermo-physical properties of both UO_2 and MOX fuels have already been sufficiently well determined to be used immediately in support of the licensing of an oxide fueled SFR.

IV.C.3 Minor Actinide Additions

Table 3 indicates the minimal U.S. experience with SFR oxide fuel containing minor additions of actinides. Some irradiation tests have been performed in the ATR reactor in Idaho and in the Phenix reactor in France. Although PIE of these tests is still in progress, initial results are encouraging (Hayes, 2011). Work has also been performed in the JOYO reactor by JAEA to study the effect of minor actinide additions on the thermal conductivity of MOX fuel. Again, results were encouraging—Am additions of up to 3 wt% only slightly reduce the thermal conductivity of MOX fuel (Morimoto et al., 2008). Nevertheless, it is clear that significantly more work is required before any licensing case could be made for MOX fuel containing actinides.

IV.C.4 Fission Product Carry-over

Fission product carry-over is really only a concern for metal fuel that has been recycled by an electrochemical process, wherein fission-product lanthanides can be carried over into the re-fabricated fuel. If the Purex process is employed for oxide recycle this is not a real concern for any of the fission products.

It should be noted that both the PFR and Phenix SFRs have been operated with reprocessed MOX driver fuel obtained via the Purex route. No deleterious effects attributable to fission-product carry-over have been reported.

IV.D Cladding and Duct Materials

At the current time only two alloy classes have enough radiation response data to seriously consider them as structural materials for a licensable reactor. The first alloy class is austenitic stainless steel (Type 316 or D9). This steel is limited to doses of ~100 dpa (and therefore the fuel burn-up to 10-11at%) due primarily to embrittlement concerns arising from void swelling and secondarily from both linear and volumetric distortion induced by swelling and irradiation creep. For such exposure limits there are essentially no significant gaps in required data and knowledge.

The second alloy class contains ferritic and ferritic-martensitic steels, especially those in the 12Cr category. These steels are much more resistant to the onset of void swelling but suffer from loss of creep strength at the higher temperatures required for efficient power generation. HT9 is the U.S. candidate and has been used to successfully build and operate without failure a set of fuel assemblies in FFTF. If this alloy is used within the boundaries of the FFTF experience there are no significant gaps in required knowledge. Such boundaries are probably insufficient for efficient power generation, however.

There is currently a lot of attention being paid to oxide dispersion strengthened ferritic-martensitic alloy in order to retain both swelling resistance and high temperature creep strength at the same time. Given the very limited data on radiation response of these alloys it is premature to consider these steel seriously for a licensable fast reactor and therefore a gap analysis is also premature.

IV.D.1 Introduction

The structural alloys used in any fast reactor can be grouped into three major categories describing their function, each with a different set of limitations and each with a different set of potential gaps in knowledge of the needed properties. In order of increasing severity of nuclear, thermal and chemical environments that the alloy will experience, these categories are the out-of-core structural components, the ducts that contain the fuel assemblies, and the fuel pin cladding, with the latter including the wire wrap used to separate the pins.

Any steel used as structural components in liquid metal cooled fast reactors must withstand an exceptionally strenuous and challenging environment, even in the absence of neutron irradiation. Depending on the particular fast reactor concept, the inlet temperature during reactor operation can range from ~250°C to ~400°C, although the start-up and shut-down sequences of these reactors sometimes utilize lower standby temperatures. The maximum temperature can range as high as 650–700°C for some components, although most non-fueled components reach maximum temperatures in the range of 400–560°C. During operation the steel must also withstand the corrosive action of fission products on some surfaces and flowing liquid metal coolant on other surfaces. Dependent on the nature of the component and the length of its exposure, there may also be significant and time-dependent levels of stress acting on the component.

Most importantly, however, the steel must survive the macroscopic consequences of a continuous microstructural alteration that arises from atomic displacements arising from collisions of energetic neutrons with atomic nuclei. The driving forces for this alteration are primarily displacements of atoms from their lattice sites, measured in units of displacements per atom (dpa), and secondarily from transmutation to both gaseous (He, H) and solid transmutants. Reviews of these processes and their consequences in austenitic stainless steels are presented in refs. (Garner, 1994) (Garner, 2010) (Garner, in press).

Neutrons in fast reactors have a spectrum of energies that is dependent on the reactor type, core loading, coolant and fuel type. The most energetic neutron spectrum will be found in fast reactors with heavy metal coolant such as Pb-Bi. The mean neutron energies in the center of a sodium-cooled core are on the order of 0.8 MeV in metal-fueled reactors and 0.45–0.55 MeV in mixed oxide-fueled reactors. The lower mean energy of the latter fuel type reflects the better neutron moderating ability of oxygen in the oxide fuel. These spectra produced approximately 5 dpa per 10^{22} n cm^{-2} (E > 0.1 MeV) in the center of EBR-II and 4.2–4.6 dpa per 10^{22} n cm^{-2} in various core loadings of FFTF. This in turn corresponds to peak atomic displacement rates on the order of 10^{-6} dpa/s. When using dpa as an exposure parameter for the structural alloy we need not be concerned by the type of coolant or the type of fuel (metal or ceramic) inside of the cladding.

The radiation-induced microstructural evolution of the steel leads to changes in physical properties such as elastic modulii and thermal conductivity, but also causes very pronounced changes in mechanical properties. Even more importantly, however, new forms of dimensional instability arise. In increasing order of significance are strains arising from radiation-induced segregation and precipitation or dissolution of precipitates, larger strains arising from radiation-enhanced creep (by orders of magnitude larger than thermal creep at lower temperatures), and finally, void swelling in which large increases in volume can occur.

The latter arises from the formation of vacuum-filled, crystallographically-faceted cavities that eventually come to dominate the microstructure. Examples of both microscopic and macroscopic consequences are shown in Figures 2 and 3. The onset of void swelling is very sensitive to metallurgical starting state, composition, irradiation temperature, dpa rate and helium generated by transmutation. The onset of swelling is somewhat less sensitive to applied stress.

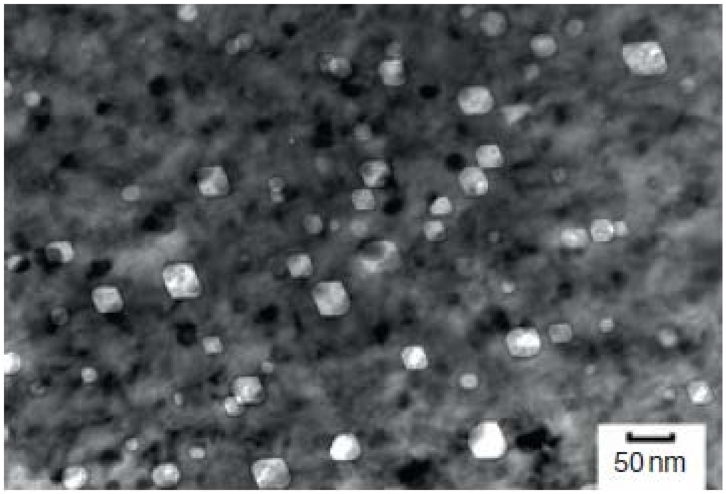

Figure 2. Void swelling and M23C6 carbide precipitation produced in annealed 304 stainless steel after irradiation in the EBR-II fast reactor at 380°C to ~22 dpa (Garner, 2002).

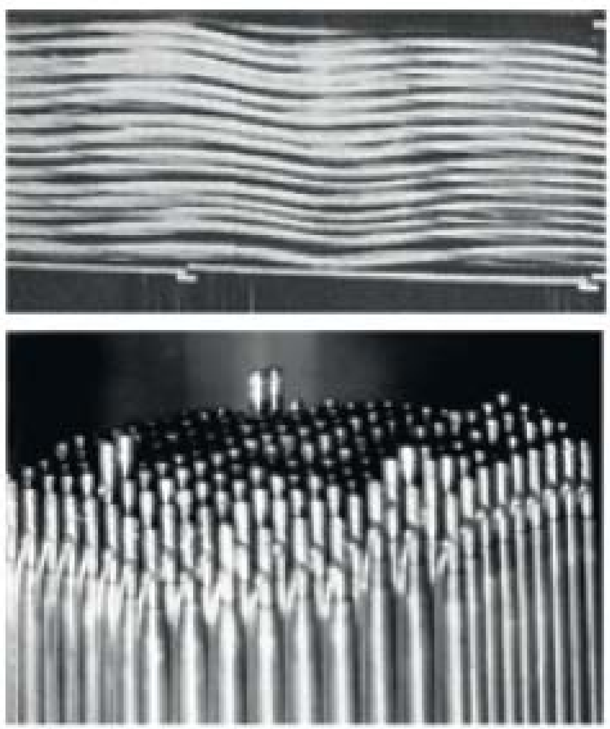

Figure 3. (top) Spiral distortion of AISI 316-clad fuel pins induced by swelling and irradiation creep in an FFTF fuel assembly; (bottom) Swelling-induced changes in length of fuel pins in this assembly in response to gradients in temperature, dpa rate and production lot variations (Makenas, Chastain and Gneiting, 1990).

IV.D.2 Problems associated with irradiation creep and void swelling

Problems arising primarily from irradiation creep can be mitigated somewhat by design considerations such as gas plenums, thicker cladding, larger flow spaces, intermediate supports restraining long components, etc., but swelling and its consequences are more difficult to mitigate.

Void swelling was found to be the Life-limiting phenomenon in austenitic stainless steels in fast reactor application. Working in conjunction with irradiation creep, tremendous distortions and volume changes can be produced by swelling, although to some extent these distortions can be accommodated in the design process if they can be accurately predicted in advance. For every design, however, there is some Life-limiting swelling limit that can be tolerated, sometimes based on closure of flow channels but often on other factors such as interference with movement of safety rods or development of unacceptable withdrawal forces.

The latter is particularly important in that void swelling when passing beyond ~10 at% increase in volume leads to development of a severe form of embrittlement in austenitic steels whereby there is total loss of elasticity and the tearing modulus of austenitic steel goes to zero. Voids at >10% swelling so modify the microstructure and compositional distribution that austenitic stainless are driven toward a martensite instability from which there is no return. This new form of embrittlement in effect becomes the Life-limiting criterion and poses a safety issue of relevance to responsibilities of USNRC. An extreme example of such embrittlement is shown in Figure 4 where ducts of three assemblies in BOR-60 failed due to high withdrawal loads arising from swelling-induced bowing and fattening of the ducts. It is of particular significance that the duct failed since ducts operate at lower temperature than fuel pins and as a consequence generally swell less.

It is difficult to preclude large levels of void swelling at higher dpa levels unless the reactor is operated at very low temperatures (<300°C) so the emphasis has been on developing an understanding of swelling and then optimizing the compositional, fabricational and environmental conditions of the steel. In general, the path chosen in most national programs was to develop a "D9-type" steel, with increases in Ni, Si. P, Ti and other elements (relative to AISI 316) to delay the onset of swelling, especially when combined with cold-working in the range of 20-25%. Eventually, however, all steels will start to swell at ~1%/dpa at all operating conditions of relevance to fast reactor operation.

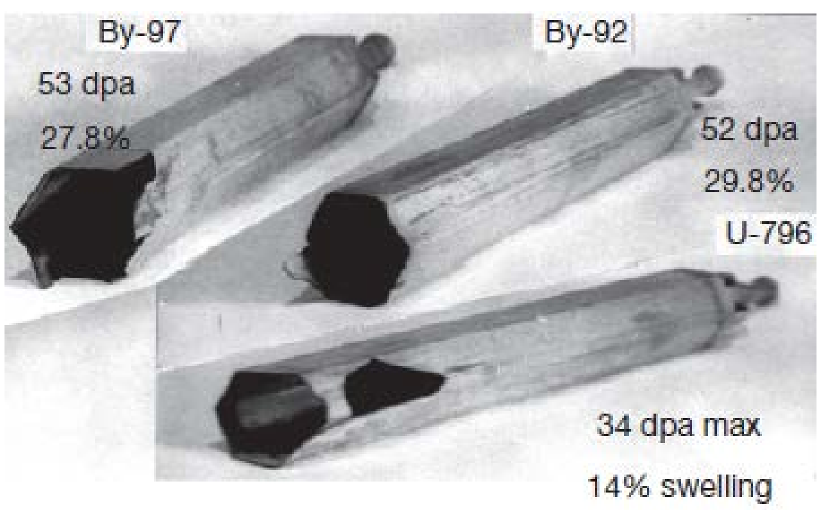

Figure 4. Severe embrittlement and failure of three BOR-60 reflector assembly ducts made of annealed X18H10T, the Russian equivalent of AISI 321 stainless steel (Neustroev et al., 2000).

Failure arose from high withdrawal loads arising from swelling and bending, the latter in response to flux gradients across the assemblies.

Another very significant problem with void swelling is that limits imposed on maximum exposure limit the burn-up of the fuel, strongly impacting the economics of the reactor. With 316 Stainless Steel and mixed oxide fuel having typical enrichments, maximum burn-ups of 10-11 at% were reached in the U.S. LMR program. Similar upper limit values were reached in other national programs. Although under some conditions steady-state swelling can be postponed until ~150 dpa, a more practical and dependable limit is ~100 dpa, especially since various of-normal histories can abruptly end the incubation regime of swelling. Such a burn-up limit is not a show-stopper for fast reactors if used for power generation. BN-600 for instance has successfully and economically generated power for decades using austenitic steels as cladding. However, if the fast reactor is envisioned to serve another purpose such as transmutation of actinides, then austenitic steels are not adequate.

If a burn-up limit of 10-11at% is accepted for power generation in the U.S. then the 316 stainless steel-clad, mixed oxide or metal fuel technology is a mature technology within the U.S. and there are very few gaps in required knowledge to be filled. Other non-swelling-related issues such as sodium compatibility and fuel-clad mechanical interaction are suitably understood and under control. However, if higher burn-up is required, then lower-swelling alloys compared to stainless steel are required, and therefore exposures greater than ~100 dpa must be sought before steady-state swelling begins.

IV.D.3 Other possible alloy classes to reduce swelling

Extensive national programs in U.S., U.K., Japan, France, Germany and the Soviet Union have all explored various categories of other alloys. With only relatively minor differences of composition chosen, all programs have reached roughly the same general conclusions.

High nickel alloys, both solute-strengthened and precipitate-strengthened offer longer delays before swelling goes into steady-state but are accompanied by new forms of embrittlement, especially involving the formation of brittle phases that coat the grain boundaries. Additionally higher levels of helium and hydrogen form as a result of the higher nickel content and appear to contribute additionally to grain boundary embrittlement. Several national programs, including in the U.S. but especially in the U. K., invested considerable research and development on this class of alloy but eventually abandoned this approach.

The use of more exotic materials, especially refractory alloys, were found to be inherently brittle after irradiation and were therefore found to be completely unsuitable. Only ferritic and ferritic-martensitic alloys offer the required superior swelling resistance, primarily because the bcc crystal structure is more resistant to void swelling, with an apparent steady-state swelling rate of ~0.2%/dpa, approximately one-fifth that of the austenite fcc structure (Garner, Toloczko and Sencer, 2000). Additionally, the incubation dose for swelling in bcc steels is much larger, partially due to the crystal structure, but also because the absence of nickel in these alloys significantly reduces the transmutation-production of helium to enhance stability of void nuclei. The most comprehensive reference on these alloys and their radiation response is contained in (Klueh and Harries, 2001).

Most national programs have focused on ferritic and ferritic-martensitic alloys, and national favorites have emerged in all countries. For the purpose of this analysis we will focus on only two alloys, both produced in the fully-tempered condition.

The first alloy is EP-450, a ~12Cr duplex alloy (~50% ferrite and 50% tempered martensite) extensively used in countries of the former Soviet Union (Bibilashvili et al., 2005). Ducts surrounding austenitic-clad fuel pins in BN-350 and BOR-60 are routinely made from this alloy.

EP-450 has the largest, best-documented and most varied data base of any alloy. It is cited here as an example of the promise of ferritic-martensitic alloys in general, best demonstrated by reaching ~160 dpa in fuel assemblies in BOR-60 (Povstyanko et al., 2010). Void swelling was still in the incubation regime without hint of steady-state swelling when this experiment was terminated. Recent ion irradiation studies confirm that the eventual steady-state swelling rate in EP-450 is the previously predicted ~0.2%/dpa, however, but will not be obtained before ~200 dpa has been reached (Voyevodin et al, 2011).

The second alloy is HT9, the U.S. candidate alloy for fast reactor application, produced in the fully tempered condition. Its primary attractiveness is that it was used to construct the Core Demonstration Experiment (CDE), a mixed-oxide sub-core of fuel assemblies irradiated in the center of FFTF, which yielded outstanding results for clad, wire and duct (Laidler and Jackson, 1990) (Leggett and Walters, 1993). Maximum swelling of only ~0.3% was reached in one examined duct at ~155 dpa at ~440°C (Sencer et al., 2000). The cladding appears to have swelled less by virtue of its higher temperature but examination is still in process.

Pressurized tubes of HT9 irradiated in FFTF at ~208 dpa and ~400°C swelling at 0.9-2.1%, however, increasing with stress level (Garner, Toloczko and Sencer, 2000). Usually, stress-enhancement of swelling signals the end of the transient regime of swelling so the transient regime at 400°C should be assumed to be on the order of approximately 200 dpa.

However, it should be noted that it was necessary to drop the power level of FFTF from 400 MW to 280 MW to accommodate the CDE activity. Ferritic and ferritic-martensitic steels do not maintain their strength at increasing temperatures in a manner comparable to those of austenitic steels. Therefore it was necessary to reduce the outlet temperature of the fuel assembly. Otherwise the fuel pins will fail as fission gas pressure builds up inside the pins.

All other relevant issues such as corrosion, fuel-clad interaction, etc. appeared to be well in control as evidenced by the successful operation without any failure. However, the CDE sub-core, though probably capable of reaching higher exposure, was terminated largely due to programmatic changes. Therefore its upper limit of HT9 subassemblies was not reached.

If future reactor designers are content to stay within the dose-temperature parameter space explored by the CDE sub-core then there are no significant knowledge gaps. Such a limitation on assembly outlet temperature would severely limit the economic viability of a power-generating plant, however, making it unlikely that licensing would be sought for a plant of that design.

IV.D.4 Oxide dispersion strengthened alloys

There is currently a significant effort in various national programs in Europe, Asia and the U.S. to extend the operating temperature range of ferritic-martensitic steels to higher temperatures while still retaining the lower swelling characteristics of ferritic-martensitic steels, not only for light water cooled reactors but also for fusion and spallation-driven devices. A number of studies have shown that in the absence of neutron exposure, finely dispersed particles of various metal oxides delay the onset of accelerated thermal creep to temperatures relevant to efficient power generation (Odette et al., 2008) (Odette and Hoelzer, in press) (Alinger, Odette and Hoelzer, 2009) (Ohnuma, Suzuki, and Ohtsuki, 2009) (Miller, Russell, and Hoelzer, 2006) (Sasasegawa, et al., 2009). One ODS alloy, MA957, was shown to retain its creep resistance in the presence of neutron exposure in FFTF (Toloczko et al., 2004).

The oxide phases can be introduced by mechanical alloying but usually result in highly textured microstructures with anisotropic properties. As a consequence it has been found that it is very difficult to manufacture tubing and to weld it into fuel pins. Newer approaches focus on growing nano-oxides in place as one way to reduce anisotropy and its consequences.

For the purposes of this gap analysis, however, it is considered to be premature to consider this class of alloys for serious near-term application. Most promising alloys have little or no irradiation data. Therefore a gap analysis is not relevant to this alloy class.

On Table 5 for cladding and Table 8 for ducts it is seen that for HT-9 the technological base is mature up to about 100DPA but additional data is required beyond 100DPA.

On Table 7 and Table 8 for 316 and HT9 for duct phenomena, respectively, data may exist for bundle, bundle-duct, and duct-duct interactions but it was uncertain if the data has been preserved in a useable form.

Table 6 shows that the technological base for 9Cr1Mo (T91) is not mature even though it remains a good alloy possibility. Additional data likely exists in France and Japan but its availability is uncertain.

IV.E Fuel Performance Codes

The following sections provide a summary of steady state and transient fuel performance codes.

IV.E.1 LIFE-METAL

The LIFE-METAL fuel performance code (Billone et al., 1986) (ANL-IFR-169, 1992) has been developed to predict the behavior of metallic fuel pins in fast reactors environment as a function of reactor operating history. The code has evolved from the LIFE series of codes (Jankus and Weeks, 1972) which perform steady-state and design-basis-transient analyses for the thermal, mechanical, and irradiation behavior of nuclear fuel pins. The original code was developed for UO2 and mixed oxide fuels for use in fast reactor systems where LIFE-4 Rev .1 is the latest oxide fuel version of the code (Boltax et al., 1990). Another version of the code is LIFE-4CN (Liu, Zawadzki and Billone, 1979), which was the basis for LIFE-METAL, and included two fuel options ((U, Pu)C and (U, Pu)N). All code versions include detailed thermo-mechanical analysis that is performed in the radial direction with provisions to specify up to 20 radial rings for the fuel/cladding system, where different rings are used for thermal and mechanical analysis.

Axial variations in operating conditions are accounted for by using powers and fast fluxes for up to nine fuel axial nodes and one plenum node. Thermally, the axial nodes are coupled through the calculated coolant temperatures. Axial heat conduction is ignored and there are no provisions for mechanical coupling between axial nodes. A detailed mechanical analysis is performed for both fuel and cladding utilizing the generalized-plane-strain assumption for each axial segment and incorporating a large strain capability. The solution procedure involves iteration on local total strain within each time step, and the solution procedure is explicit in time.

LIFE-METAL code development has been associated with the Integral Fast Reactor (IFR) program (Chang, 1989) where the code was the focus of the program activities related to prediction of fuel-pin behavior under normal operating conditions. Predictions of interest to the nuclear design are changes in fuel length and fissile content due to burnup and breeding. Thermal predictions of fuel temperature, design margins to fuel melting, and design margins to low-melting-temperature alloy (e.g., U-Fe) formation are also of interest. Mechanical predictions useful to designers are fuel-cladding mechanical interaction (FCMI) and fuel-cladding chemical interaction (FCCI), cladding deformation and design margin to significant coolant flow area reduction, and cladding damage and design margin to cladding failure due to fission gas pressure loading.

LIFE-METAL Validation

The following validation discussion is based on the last code validation activities performed by Billone (Billone, 1994). This validation effort has been extensive as it used post irradiation examination (PIE) data that are available from a large number of metallic fuel-pin irradiations at EBR-II and FFTF (Crawford, Porter and Hayes, 2007). Post irradiation examination (PIE) data include fission gas release, fuel volumetric and fuel length change, cladding diametral change, and cladding wastage. Axial profiles are available for fuel radial growth at low burnup (prior to and including initial fuel-cladding contact) and for cladding radial growth for a wide range of burnups and fast fluences. Some data that are available on a more limited basis are radial and axial variations in U, Pu and Zr content, fission gas porosity, axial variations in fraction of porosity filled (logged) with Na; and depth of C-depleted and Ni-depleted zones in HT9 and D9, respectively. Fairly complete sets of data are available for 80 fuel-pin irradiations (111 pins in total were used in the validation). Limited data (e.g., fuel length change, cladding diameter change) are available for hundreds of irradiated fuel pins.

The validation database includes three cladding types (cold-worked, austenitic D9 and 316 stainless steels and HT9 ferritic/martensitic steel) and eight fuel compositions (U-10Zr, U-3Pu-10Zr, U-8Pu-10Zr, U-19Pu-6Zr, U-19Pu-10Zr, U-19Pu-14Zr, U-22Pu-10Zr and U-26Pu-10Zr, where the numbers represent weight percents). The data from the 111 pin irradiations fall into one or more of the following categories: fission gas release, fuel axial strain, fuel diametral strain, cladding diametral strain and penetration depth (wastage) at the cladding inner diameter due to ingress of fission products and egress of cladding constituents. For the last three categories, axial profiles are often available. This implies a large number of data points per fuel pin irradiation. Also, in the case of fuel axial expansion and peak cladding strains, which are routinely measured for all elements within a subassembly, the number of data points is much larger than the number of validation cases.

LIFE-METAL Status

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 done to the code 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 documentations and re-validation effort to release it. Further, calibration and validation effort of the code can be done once further data from other EBR-II experiments are generated as part of efforts to create a database for metallic fuel irradiated at EBR-II.

LIFE-METAL Limitations

The code lacks the implementation of mechanistic models in a good part of its development. Thus, the code is limited in extrapolating fuel performance outside of the validation range of parameters, since a lot of the code models are based on correlations rather than mechanistic models. However, the code still can be useful for scoping calculations outside of its range of validation. As stated previously, the thermo-mechanical modeling part of the code has limitations due to the axial nodes being thermally coupled only through the calculated coolant temperatures, axial heat conduction being ignored, and there being no provisions for mechanical coupling between axial nodes. Finally, there are no models in the code that are relevant to evaluations of transuranics bearing fuel.

IV.E.2 LIFE-4 (oxide)

The LIFE-4 (Rev. 1) code (Boltax et al., 1990) was developed to calculate the thermal and mechanical behavior of mixed oxide fuel elements in a fast-reactor environment. The code is the reference national code for modeling the thermal, mechanical and materials performance of fast reactor oxide fuel and blanket pins during normal operation and during transients up to cladding breach. It integrates a broad material and fuel-pin irradiation database into a consistent framework for use and extrapolation of the database to reactor design applications. It can also be used in the design and analysis of transient fuel-pin experiments, and in the identification of critical experimental areas. The code has a one-dimensional generalized-plane-strain mechanical analysis procedure for fuel ([U,Pu] O_2 and UO_2) and cladding (several types including different types of 316 stainless steel, HT9, and D9). It has a steady state and transient thermal analysis system for fuel, cladding and various flowing or static coolants. Thermal and mechanical behavior including thermal expansion, fuel restructuring, cladding deformation, and cladding breach is calculated from a number of phenomenological models. The thermal analysis includes the following:

- Establishing fuel and cladding temperatures

- Calculating stress-strain independent, but temperature dependent, diffusion phenomena such as:
    - Pore migration
    - Grain growth
    - Oxygen migration
    - Pu migration
    - Fission gas release.

The mechanical analysis is a stress-strain calculation in the fuel and cladding and includes the following:

- Elastic creep (irradiation and thermal creep)
- Plastic flow
- Swelling
- Thermal-expansion strains.

The code also includes wastage model, and it takes into account the fuel cladding mechanical interaction in calculating the cladding diametral strains.

Similar to the LIFE-METAL code, the fuel pin is divided axially into a maximum of nine fueled sections and one plenum section. Each axial section is divided radially into a maximum of 20 cylindrical rings for mechanical analysis. Each axial node is mechanically independent of all other axial nodes. Thermally, the axial nodes are coupled through the calculated coolant temperature. Axial “lock-up” effects or similar axial coupling effects are not modeled. Thus, if a given pin section was run as part of a one-node problem or as part of a five-node problem, the computed temperatures, stresses, etc. would be the same except for two calculated parameters which are:

- The plenum pressure, which will be different because of differences in the total amounts of fission gas released.
- The axial vapor-phase transport, where the differences are very small for long (~ 1 meter) pins.

Because of this axial independence, each axial node can be solved separately. Finally, the code has a special option for blanket fuel pins that are characterized by larger diameters.

LIFE-4 Validation

LIFE-4 (Rev. 1) has been validated against pin data from 64 pins that were irradiated under steady-state conditions in EBR-II, 12 pins that were transient-tested in the TREAT reactor, and 13 pins that were irradiated in FFTF. Data include both (U,Pu)O_2 mixed-oxide fuel pins and UO_2 blanket pins which were irradiation tested under steady-state and transient conditions. For fuel pins under steady-state conditions, this calibration/validation covered the following ranges of operating conditions: 16-50 kW/m (peak power), 1.5-5x10^{15} n cm^{-2} s^{-1} (peak flux), 360-700 ^{o}C (peak cladding temperature), 0-13% burnup and 0-1.5x10^{23} n.cm^2 (peak fluence). The code calibration was done by adjusting the less well-defined parameters in the fuel models and properties within physically realistic limits or measurement uncertainties to minimize the deviations between code results and fuel pin post-irradiation examination (PIE) data. The beginning-of-life thermal calibration was performed by adjusting the high temperature end of the fuel thermal conductivity equation, the accommodation coefficient in the gap conductance model and the constants in the pore velocity expression. For the later-in-life thermal calibration, changes were made to the dependence of fuel thermal conductivity on burnup (due to solution of fission products). The primary parameters employed for the mechanical calibration were constants in the fission gas release expressions and the fuel swelling model. The key fuel pin data used were fission gas release and cladding mechanical strains. Also, several other fuel constants were changed during the course of calibration to achieve improvements in code predictions not obtainable by varying the parameters described above.

The code validation was performed by investigating code predictions for fuel pins, which were not used in the calibration of the code. No adjustments were made to code calibration constants for the analysis of the validation pins. This is a summary of the validation data:

- The cladding for all pins except the high burnup F20 pins was 20%cold-worked AISI Type 316 stainless steel of the N-lot type.
- The high burnup F20 pins used solution-annealed AISI Type 316 stainless steel cladding.
- All pins are from mixed fuel pin tests, except the W20 pins, which are from blanket pin tests.
- All pins except DEA-2 pins were irradiated in EBR-II. DEA-2 pins were irradiated in FFTF.

LIFE-4 Status

The LIFE-4 (Rev.1) version of the code is well documented and maintained through the national code center. This version dates to the early 1990’s and is the latest code version. There have been further code validation efforts and modifications using FFTF experimental data that were not available during Rev.1 version. We do not have access to documentations of this but it might become available through the recent effort to preserve FFTF documents and codes. Again, the code does not have any models specific to transuranics fuels.

LIFE-4 Limitations

There are a number of limitations on the code ability to predict various features of the actual behavior of fuel pins, which are caused by errors in, or lack of, models of fuel and cladding behavior. The following are examples of important code short-falls, which are mainly related to the limitation, that were imposed on the code developers in the past, mainly in order to reduce the computation time, in addition to inherent use of correlations in some parts of the code compared to mechanistic models. For example, the axial mechanical analysis in the code is a simple one, which assumes that no axial section of fuel is affected by the condition or behavior of any other axial section. That is, axial lock-up effects are not taken into account. This leads to mechanical diametral strains near the bottom of the fuel column that tend to be under-predicted. The same strains near the top of the fuel column tend to be over-predicted.

Another consequence of the simplicity of the code's mechanical analysis is that the thermal and mechanical analyses in the code are primarily coupled through the gap-conductance model. Both analyses are not solved simultaneously; instead, the thermal analysis uses the fuel-cladding gap or contact pressure from the mechanical analysis of the previous time step. Consequently, the code has a tendency to become unstable and produce oscillating solutions under certain conditions. The code is limited in use to oxide or mixed oxide fuels without transuranics.

IV.E.3 FPIN2
FPIN2 (Hughes and Kramer, 1986) (Kramer et al, 1992) code has been developed to model the thermal and mechanical behavior of IFR metallic fuel pins. It was developed primarily to analyze the behavior under accident transient conditions. The emphasis in the development and validation of the code has been the incorporation of models relevant to the time scale and temperature range of accident transients. It requires the user to provide data describing the pre-transient condition of the fuel pin. FPIN2 has relied on TREAT tests as the primary source of data for overall code validation. The regime of extended transients requires an extrapolation of FPIN2 to longer times. It has been also applied to the Whole Pin Furnace tests conducted at Argonne as part of the IFR program. There are many similarities between LIFE-METAL and FPIN2 in the fundamental assumptions that determine the governing equations that are solved. For instance, both codes use a finite difference formulation of the heat transfer which assumes that heat is only conducted radially in the fuel and the cladding and convected axially by the coolant. For the purpose of analyses of experiments such as the Whole Pin Furnace tests, options are available to bypass the coolant calculation and input directly the known cladding surface temperatures. The mechanical analyses in both codes assume axial symmetry and generalized plane strain which results in an essentially one-dimensional (radial) calculation for each axial node of the fuel pin. LIFE-METAL uses a finite element formulation of the governing equations whereas FPIN2 uses a finite pin formulation. In addition, both codes use similar fundamental properties of metallic fuel pins. As mentioned above, differences in the regimes that the LIFE-METAL and FPIN2 address have resulted in major differences in the models contained within the codes. To some extent these differences have also influenced the formulation of the equations and solution procedures (e.g., finite difference vs. finite element). One of the primary differences in methodology that affects the calculations is how the code determines the deformation and failure of HT9 cladding. Here, LIFE-METAL has based the models on long-term creep tests, while FPIN2 has based the models on tensile test data and FCTT (Fuel Cladding Transient Tester experiments (Cannon, Huang and Hamilton, 1991) data.

IV.E.4 SEIX3
The SIEX3 code (Baker and Wilson, 1986) was developed at Hanford Development Engineering Laboratory (HEDL) during the project to design, construct, and operate the Fast Flux Test Facility (FFTF). The code is strongly based on experimental data obtained from irradiations performed in EBR-II, and in FFTF. The code contains correlated models based on experimental data that describe thermal performance and phenomena causing dimensional changes. The code predicts cladding damage and failure due to stress rupture in MOX fuel pins irradiated at steady-state conditions in an LMR.

In SIEX3 modeling, the pin of interest is analyzed using the driver fuel pin correlations if the fuel Pu/(Pu+U) ratio is greater than 0.12 and fuel pellet diameter is less than 0.32 inch; otherwise it is assumed to be a radial blanket fuel pin and is analyzed using the blanket fuel pin correlations. In either case, the fuel column is divided into a user-specified number of axial segments (up to 21) of equal length. One-dimensional radial heat transfer is calculated at the axial center of each segment. The heat generated is integrated over the axial segments to calculate the coolant temperatures along the pin, from input values of axially varying linear power, coolant mass flow rate and inlet temperature. The temperature-drop across the cladding wall includes the power in the cladding due to gamma heating (0.7% of total power). The following are calculated for each segment:

1. Coolant temperature
2. Cladding outer and inner surface temperatures
3. Fuel-cladding gap size
4. Movement of void volume from the fuel-cladding gap to the central void in the fuel
5. Solid fission product (gray phase) buildup in the fuel-cladding gap
6. Fuel-cladding heat transfer coefficient
7. Fuel radial temperature distribution
8. Fuel restructuring radii for equiaxed and columnar zones
9. Displacement of the fuel and cladding caused by swelling and thermal expansion
10. Fission gas generation and release
11. Cladding wastage, stress, creep and damage.

The cladding performance calculations in SIEX3 are based on the observed performance of reference design liquid metal reactor fuel pins with a fuel smear density of 85% TD. The calculations assume that pin diameter changes depend primarily on neutron-induced swelling and cladding creep strain caused by fission gas pressure loading only. SIEX3 calculates the cladding thermal expansion, cladding swelling, wastage, thermal creep, irradiation creep, and stress rupture damage [cumulative damage fraction (CDF)].

The code uses two time steps, one defined by the code user for writing output results, and the other of two effective full power days (EFPD) for carrying out mechanical calculations. This short time step is used to ensure accuracy of calculations that are sensitive to a small change in fuel pin fluence, stress, or burnup.

SIEX3 has the following important capabilities:

1. The code allows annular fuel. The fuel pellet central void model in the code accounts for two processes:
    a. Densification of the columnar fuel grain region, and
    b. Porosity movement from the fuel-cladding gap to the central void, due to fuel cracking and subsequent healing.
2. The fabricated fuel oxygen-to-metal ratio is an input to the code. This ratio is used in correlations for columnar grain growth temperature, and equiaxed grain growth temperature.
3. For rapid startup with fresh fuel, the code uses a special columnar grain growth temperature, in addition to that for normal irradiations.
4. The code accounts for void volume caused by fabricated end dishes in the fuel pellets.

5. The code provides a single convenient input parameter to specify the recommended cladding performance models for twelve cladding materials.
6. For modeling radial blanket fuel pins, the code uses correlations specifically developed from the irradiation test database for blanket pins.

SIEX3 Status and Limitations

The version 3 of the code, SIEX3, is the latest version of the code available through the national code center. No further developments are done with the code.

SIEX3 has the following important limitations:

1. The code uses a single input value of the external pressure acting on the cladding outer surface. It does not account for the axial variation of the coolant pressure acting on the cladding outer surface.
2. There is no provision to model a fuel pin fission gas plenum located below the fueled section of the pin.
3. The code does not compute and print the change in fuel column length.

IV.E.5 FEAST

FEAST-METAL and FEAST-OXIDE are fuel performance codes developed for predicting steady state and transient behavior of U-Pu-Zr metallic fuel alloys and mixed oxide fuels with stainless steel clad in sodium fast reactor environments. The codes are developed at MIT with support from NRC. The code properties and validation databases are derived from information available in the open literature for both types of fuel. Other validation and calibration data used with the LIFE series of codes were not used in the validation of FEAST, so it has a limited validation database. The thermo-mechanical models in the code are similar to those in LIFE codes models, while the fission gas release models used in FEAST are mechanistic rate equations based models compared to the correlations used in the LIFE codes.

IV.E.6 DEFORM-4

The DEFORM-4 model was developed at ANL as part of the SAS4A safety analysis computer code system. DEFORM-4 contains detailed phenomenological models of MOX fuel behavior, including fission gas generation and release, porosity migration, fuel and cladding swelling, and fuel-cladding mechanical interactions. The models in DEFORM-4 are coupled and integrated with the fuel pin heat transfer, coolant dynamics, and material melting and relocation models in SAS4A. The modeling in DEFORM-4 has been upgraded by a German-Japanese-French consortium (Imke, Struwe and Pfrang, 1995) to reflect fuel materials and experimental data generated in their national LMR development programs. The upgraded model is designated DEFORM-4.

IV.E.7 Knowledge preservation and database development (Metal fuel performance)

Most of the existing knowledge base for metallic fuel was generated during the IFR program, in addition to knowledge base generated earlier at EBR-II with U-Fs fuels and limited experience with reprocessed fuels. This knowledge base includes ANL reports (green packs), IFR reports, EBR-II run reports, memos, PIE reports, drawings, experiments qualification reports, and publications in journals and conferences, information regarding measured properties (e.g., IFR metallic fuels handbook), and out of pile experiments. Also, measurement documents like micrographs, profilometry data, fission gas release data, and other measurements data are well documented in most cases. A lot of those documents are available in digital form as part of the IMIS database that was developed closer to the end of the IFR program. However, this database is not complete as it does not include detailed pin-by-pin data associated with the different metallic fuels experiments conducted at EBR-II. This detailed information was generated for only four experiments (X425, X430, X441, and X447). Detailed data for those experiments included pin by pin axial profiles of operating parameters for each run that the experiment was present in EBR-II, pin location in each experiment, in addition to the detailed PIE data associated with each pin. Analysis of remaining experiments depended on operating parameters for EBR-II that were generated with an older methodology that did not include pin-by-pin depletion calculations and other related details for calculating the temperatures within a subassembly and pin temperatures. Although, information from this older methodology was adequate at the time to qualify the experiments and analysis, there is a need to go back and look in detail into those experiments and associate its detailed information with the experimental observations to have thorough consistent analysis of those experiments. The newer methodology was developed close to the end of the IFR program and it was used to generate the operating parameters for each of the four experiments mentioned above this methodology depends on ANL suite of codes, REBUS/EBRFLOW/RCT/RCTP/ SUPERENERGY-II. Further effort will be needed to generate such detailed set of data for the remaining experiments and make it available to current and future analysts interested in metallic fuels. In addition, effort is needed to relate those detailed data sets to the available documentations and PIE information in an advanced database that will facilitate access to this information and connect between the experimental data and the detailed calculation for those experiments. This needs to be done by staff remaining from the IFR program that are familiar with the data and are capable of generating this comprehensive database.

A detailed description of knowledge preservation and database development for oxide fuel could not be completed in time for inclusion in this report.

IV.F Fuel Fabrication

The U.S. experience with fuel fabrication for a sodium-cooled fast reactor (SFR) comes largely from the fueling of the EBR-II and FFTF. EBR-II operated with various designs of metallic fuel in stainless steel cladding and FFTF used mixed oxide (MOX, $UPuO_2$) in stainless steel cladding. The other U.S. SFR was the Enrico FERMI reactor. The fuel design used in the FERMI reactor (U-10Mo in zirconium alloy clad) would not be used in a reactor today as the cladding material has limited use at proposed operating temperatures. The assembly design, where a Type 347 stainless steel square grid supported the fuel elements, was also unique and of little interest today.

Two recent publications (Burkes, Fielding, Porter, Crawford and Myers, 2009) (Burkes, Fielding, Porter, Meyer, and Makenas, 2009) reviewed the fabrication process development for both metallic and MOX fuel, as well as mixed carbide and mixed nitride fuel. The two journal articles provide a good bibliography of published accounts of domestic SFR fuel fabrication development.

IV.F.1Fabrication Records

Detailed specifications, procedures, and batch fabrication records would be key to facilitate a new production of these fuels without the burden of development and repeating earlier mistakes. The evolution of the specifications often reveals how lessons learned were applied to the next generation of specification.

Most of these documents resided internal to the organizations, Argonne National Laboratory for EBR-II and Hanford Engineering Development Laboratory, HEDL (aka Westinghouse Hanford Company, WHC) for FFTF and perhaps had limited distribution.

IV.F.1.a EBR-II

Fuel and assembly fabrication procedures, fuel specifications, and at least most fabrication records can be located using the Idaho National Laboratory's Engineering Document Management System (EDMS). This system documents where 'hard' copies of these materials are stored. An inventory has not been done to assess whether all relevant documents can be found there but a cursory check found that all types of this documentation were catalogued and stored. A 'word' search can be used in this system to find a listing of the location of a document being sought. Some of the documents, such as select specifications, have been scanned into the system as individual documents. If records such as these are to be made useful to future fuel development efforts they should be scanned into electronic media where they can be searched and data can be extracted as needed. These efforts have become relatively inexpensive.

Fabrication statistics were compiled for some fuel fabrication campaigns. Likewise data concerning rates of various types of rejects and other losses, returns, and pin properties (size, composition, etc.) for the most recent fuel campaigns exist that can be examined statistically to review for future process improvements. They have already been used to predict process loss for a reprocessing design. These are not currently published in the open literature.

IV.F.1.b FFTF

An inquiry was sent to individuals still working with DOE Richland area contractors who may have knowledge of fuel-related records related to FFTF fuels and experiments [Ron Omberg (PNNL) and Ron Baker (RL)]. According to Omberg, there is an active program to recover the records on FFTF fuel testing and driver fuel fabrication information, limited in rate and scope by annual funding. A summary report on FFTF codes and standards is due to be produced at the end of fiscal year 2011. The report will address both the reactor plant and the fuel and so will not focus on the fuel alone. It appears that records do exist for FFTF fuel but these individuals will not know what the extent is until funding is advanced to examine the stored records.

IV.F.2 Source of Cladding/Duct Materials and Tube/Duct Fabrication

A known 'gap' in the knowledge/experience related to fuel and assembly fabrication is the existence of an experienced and qualified supplier of materials and especially the fabrication of hardware components like tubing (cladding) and hexagonal ducts. There are no current suppliers at least if ferritic/martensitic stainless steels are the material choices.

When the Clinch River Breeder Reactor (CRBR) was going to built, FFTF was under construction, and EBR-II was a working reactor and required a steady source of hardware, there were domestic suppliers of nuclear grade stainless steel (Types 304 and 316) tubing and duct. A few of these suppliers, especially Carpenter Technologies (CarTech) and Superior Tube were funded to develop the techniques to manufacture components from what were called at the time 'advanced alloys' such as D9 (Ti-modified austenitic) and HT9 (12Cr-1Mo ferritic/martensitic, F-M). CarTech supplied HT9 tubing and duct for both FFTF and EBR-II. CarTech also drew some modified 9Cr-1Mo F-M (T91) ducts for use in EBR-II.

Recently a small scale study was undertaken to ascertain the ability of the U.S. industry to produce F-M cladding to a specification that was similar to previous FFTF and EBR-II cladding specifications. The resulting study showed that given enough scheduling time a heat of F-M material could be produced using the required double melt process of vacuum induction melting followed by vacuum arc remelting. The only real issue is the available furnaces are scheduled 18 months or more out. Although an industrial sized heat is feasible, the forming of cladding tubes is problematic. Currently there are few if any industrial applications of precision drawn F-M tubing. Tubing manufacturers can produce high quality austenitic stainless steel tubing or other alloys which are currently used in various industries, but F-M steels need specific processing parameters to produce a consistent product that can meet the precise specification of nuclear cladding. Recent efforts resulted in a low yield percentage of tubing that met both dimensional standards and surface finish standards that were similar to past cladding standards. It was assumed this was because the F-M steels must be processed differently than more common alloys. It is likely that F-M cladding could be produced; however, a significant development effort would be required to recapture the processing techniques.

There have been very recent inquiries by commercial nuclear interests into providing core loadings of HT9 or T91 duct and cladding. Kobe Steel, Japan, has shown interest and competency with providing similar materials of nuclear grade. The lack of domestic, qualified suppliers is a 'gap' in domestic technology that should be closed at first onset of a new mission to build a fast reactor.

IV.F.3 Material and Fabrication Development – The National Cladding and Duct (NCD) Development Program

The Fast Breeder Reactor (FBR) development program contained a large effort to develop materials and design for use as fuel cladding and assembly ducts. This was the National Cladding and Duct (NCD) Program. The Hanford Engineering Development Laboratory (HEDL, aka Westinghouse Hanford Company [WHC], GE Nuclear, Westinghouse – Advanced Reactors Division (WARD), Argonne National Laboratory, Oak Ridge National Laboratory and the Naval Research Laboratory were the primary participants in this work and held regular inter-laboratory meetings to review data and develop design codes to represent various material properties and effects. A large amount of what is known about the performance of these stainless steel materials in an irradiation environment was produced in this program. Unfortunately much of the work was never published because at the time the work was considered Applied Technology, a categorization designed to keep the information within a very limited distribution. At the end of the program the materials were studied for application to fusion reactors. The new funding organization allowed this later work to be published openly and does not require an information recovery process.

The older, and larger, body of this work was largely documented in a Quarterly Report (HEDL TC-160-XX). It needs to be recovered to allow access and in a way so it may be word-searched. The quarterlies consisted of a series of short papers on specific subjects such as, “ANALYSIS OF RESULTS FROM THE SECOND INTERIM EXAMINATION OF THE ADVANCED ALLOY CREEP IN BENDING EXPERIMENT” There is often no index or listing of the articles contained within a report but they are arranged by general subject areas, such as creep, swelling, microstructure, simulation (ion irradiation, etc.), mechanical properties, coolant compatibility, fabrication, etc., usually five or six at a time that comprised the NCD program. However, these subject areas changed several times during the life of the NCD Program. It is therefore very difficult to locate the information for which you are searching without a word-searchable database, and currently that does not exist. Since much of the information in these ‘Applied Technology’ reports has never been published and were not otherwise available, we are destined to repeat the work if such a database is not created and advertised to the new generation of researchers.

The National Cladding and Duct (NCD) Quarterly Reports (HEDL TC-160-xx) contain some of the only documented accounts of developing the fabrication methods used to make fuel cladding and assembly hardware from alloys such as HT9. For a time the NCD quarterlies did have a chapter, “Group D, ‘Fabrication and Development’” and this is where some of this information can be found. A scanned searchable database is needed to make this important information useful to current-day researchers and development engineers, and to help to qualify hardware vendors.

IV.H Transient Behavior

Behavior of advanced fuels must be acceptable under a wide range of normal and off-normal environments and conditions that can potentially arise in sodium cooled fast reactors (SFRs) (Wright, Dutt and Harrision, 1990). Specifically, qualification of fuels and the approval of reactor designs include understanding and reliable prediction of the transient behavior of fuels and cores under the full range of anticipated and postulated conditions through cladding breach and beyond. Thus, in addition to having proven, excellent performance, fuels must be shown to have acceptable behavior under off-normal conditions as well as design-basis accident conditions and beyond -- as needed for approval of lead test assemblies and in plant licensing.

Off-normal conditions will arise from local defects and/or plant transients. These may potentially result in local fuel damage, cladding failures, and/or extensive fuel damage. Outcomes will depend in large part upon the action of on-line diagnostics, operator actions, and automatic plant protection systems. Outcomes will also depend on key properties of fuel, cladding, and core structural materials at elevated temperature -- specifically, the interactions and compatibility of those materials with each other and with the sodium coolant. Increasing severity of off-normal conditions to which fuel might be subjected may result in an expansion of the types and ranges of phenomena that characterize the response and the damage that may result.

Historically, fuel behavior studies for oxide and metallic fuels have extended far beyond characterization of fuel and cladding behavior under normal conditions. In addition, models and codes were developed to specifically address transients for conditions short of cladding breach as well as those associated with severe accidents. For SFRs, the models and codes needed for such analyses typically involve a combination of thermal-hydraulics, mechanics, neutronics, and materials behavior. Furthermore, fundamental differences in the transient behavior between oxide fuels and metallic fuels resulting from local faults or whole-core accident initiators have required separate modeling and validation bases.

Because early (1960s) metallic fuel designs were limited to much lower burnup than were oxide fuel designs, transient behavior of oxide fuels was studied far more extensively than that of metallic fuels for the next two decades. However, during the 1970s and 1980s, interest in metallic fuels increased due to advances in understanding of fuel behavior, improved irradiation performance, attractive fabrication and reprocessing features, and favorable transient behavior characteristics of that fuel form. To date, both up-to-date oxide and metal fuel designs have performed well under normal conditions at least for burnups up to about 10-12 at%, and both fuel types have reached burnups as high as 19-20 at%. Cladding failure thresholds (expressed in terms of power-over-flow ratios) are similar for both oxide and metal fuel types. Nevertheless, the demonstrated response of both fuel forms to severe accidents has prompted work to further improve existing designs.

With oxide fuels, there is motivation to reduce the probability of molten fuel being released from cladding into coolant channels causing energetic coolant expulsion from the core and complete coolant channel blockage by freezing fuel and/or cladding, resulting in a severely-disrupted, heat-generating, uncooled, metastable core configuration which might become re-critical. With metallic fuels, iron-based cladding and structural materials are vulnerable to the formation of low-melting-point compositions with uranium- (and plutonium-) fuel materials. While formation of such molten phases is a principal cause of cladding failure in metallic fuels, there appears to be little tendency to form coolant flow blockages. The gap between current knowledge and what may be needed for licensing, either for metal or oxide fueled cores, is associated with future designs capable of reaching high burnups (20 at% and above) along with improved safety performance.

The transient behavior of SFR oxide fuels of current designs through at least the mid-1980s has been extensively investigated and is generally well predictable with fuel behavior and whole-core-accident codes (on a fuel macro-scale, up to medium burnups, and with significant reliance on empirical correlations). Extending those codes to describe improvements in fuel pin and subassembly designs capable of reaching fuel burnups of 20 at% and above along with improved safety performance will require additional validation and model development. Examples of oxide design improvements under study include annular fuel with annular axial blanket/reflector to facilitate pre-failure molten fuel axial dispersal, longer-life cladding materials such as oxide-dispersion-strengthened steel, and sub-assembly designs to facilitate axial flow of molten core materials out of the core during postulated subassembly or whole-core melt accidents.

While experience with metallic fuel has been considerable, the empirical and analytical knowledge base for metallic fuels is considerably smaller than for oxide fuels. In particular, transient testing of modern metallic fuels has not been nearly as extensive as was performed for oxide. Correspondingly, performance and accident codes for metallic-fuel are less well developed. Proposed fuel design improvements in metallic fuel to prevent fuel-cladding chemical interaction (such as fuel additives or cladding liners or coatings) will need to be investigated regarding their efficacy during normal operation and influence upon the fuel transient response.

Overall, little is known about the steady-state performance or transient response characteristics of metal or oxide fuels and cores either:

1. At high burnups (> 12 at%),
2. With recycle fuels, or
3. With fuels having high minor actinide content.

Even less is known about the steady-state and transient response of future fuel design improvements to reach burnups well above 20 at%, such as fuel pins having lower fuel smear densities and advanced claddings.

Evaluating advanced fuels and core designs regarding such key issues as margin to fuel melting, margin to cladding failure, and acceptable response beyond cladding failure will need to be a continuing activity. A high importance level of such issues coupled with a minimal existing knowledge base to resolve those issues leads to a conclusion that the closing the knowledge gaps described above is of high priority [see “Experimental Facilities for Sodium Fast Reactor Safety Studies,” 2011 OECD report NEA/CSNI/R(2010)12] and requires the availability of suitable transient testing capabilities.

IV.I Structural Materials

Due to the unavailability of the proper expertise during the panel meeting, regulatory gaps in structural materials were not directly considered by the panel. Instead, this report leveraged a number of previous studies which considered the current state of SFR structural materials. This section summarizes the findings of these studies (Chopra and Natesan, 2007) (Natesan et al, 2008).

An objective of these reports was to evaluate the licensing and design implications of the ASME code qualification on an SFR (Natesan et al., 2008). It was noted that Clinch River Breeder Reactor (CRBR) and the Power Reactor Innovative Small Module (PRISM) faced regulatory questions concerning compliance with the elevated temperature structural integrity criteria (ASME Code Section III Subsection NH). It should be noted that Subsection NH has not been approved by the NRC and a version of Subsection NH will need to be adopted by the NRC before an SFR can be licensed in the US. Currently, only 5 alloys are included in Subsection NH including: 304SS, 316SS, 2.25Cr-1Mo, Alloy800H, and Mod. 9Cr-1Mo (grade 91).

Thirteen major gaps were identified:

- Lack of materials property allowable data/curves for 60 year design life
- Lack of validated weldment design methodology
- Lack of reliable creep-fatigue design rules
- Lack of hold time creep-fatigue data
- Improved mechanistically based creep-fatigue life predictive tools are needed for reliable extrapolation of short term data to 60 year life

- Lack of understanding/validation of notch weakening effects
- Methodology for analyzing Type IV cracking in 9Cr-1Mo weldment
- Lack of inelastic design procedures for piping
- Lack of validated thermal striping materials and design methodology
- Material degradation under irradiation
- Materials degradation under thermal aging
- Materials degradation in sodium environment
- Degradation under sodium-water reaction

Tables 16 and 17 list the materials historically used in SFRs and their associated Subsection NH limits.

Table 16. Materials Used in Past Sodium-Cooled Reactors

Country	Reactor	Vessel	Intermediate Heat Exchanger (IHX)	Steam Generator	
				Evaporator	Superheater
USA	Fermi	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	EBR-II	304L	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	FFTF	304	316	a	a
	CRBR	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
UK	DFR	316	316	321	321
	PFR	321	321	Fe-2¼Cr-1Mo	316H
Russia	BOR-60	304	304	Fe-2¼Cr-1Mo (Alloy 800)	Fe-2¼Cr-1Mo
	BN-350	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	BN-600	304	304	Fe-2¼Cr-1Mo	304
Germany	SNR-300	304	Fe-2¼Cr-1Mo-1Nb	Fe-2¼Cr-1Mo-1Nb	Fe-2¼Cr-1Mo-1Nb
France	Rapsodie	316L	316	a	a
	Phenix	316L	316	Fe-2¼Cr-1Mo	321
	SuperPhenix	316	316	Alloy 800 tubes and 304, 316L shell	b
Japan	Joyo	304	304	a	a
	Monju	304	304	Fe-2¼Cr-1Mo	304

[a]Sodium to air heat exchanger.
[b]Evaporator and superheater are combined in a single unit.

Table 17. Materials included in Subsection NH allowable

Material	Temperature (°C)	
	Primary stress limits[a]	Fatigue
304	816	704
316	816	704
2.25Cr-1Mo	593[b]	593
Mod.9Cr-1Mo	649	538
800H	760[b]	760

[a]Allowable stresses extend to 300,000 h (34 years).
[b]Temperatures up to 649°C allowed for no more than 1000 h.

Tables 11-15 summarizes potential systems and components of an SFR and identifies gaps in fabrication and performance.

Table 11 indicates that little additional research efforts are needed to develop Reactor System Structural Components. All ranked structures and components are satisfactory in fabrication, degree of knowledge and sufficiency of the structure or component to complete the desired mission. The only gap identified was the design of the rotatable plug for the reactor vessel head.

Tables 12 and 13 indicate that additional work is needed for all primary and secondary structures and components in the heat transport system. The Electromagnetic pump was identified as a major gap.

Table 14 indicates that a large research effort will be needed if the Supercritical CO_2 (S-CO_2) Brayton cycle is to be incorporated into an SFR design. Most components, excluding potentially the Recuperator, need to select potential materials, develop fabrication capacity and improve the experimental database. Much of this work is now being conducted at the S-CO_2 test loop at Sandia National Laboratories.

Table 15 indicates that a Rankine power conversion cycle could be incorporated into an SFR design with minimal additional work needed to improve the technology status of the steam generator shell, steam generator tubing, and hot leg steam piping.

V. Identification and Discussion of Significant Gaps

The current state of knowledge of SFR fuel and structural material performance is sufficient for designing and licensing an SFR today within the envelope of a conservative data base. The boundaries of a conservative data base 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 data base 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 QA 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 data base is applicable. It must be appreciated that few, if any, vendors of these materials exist. Thus for fuel from zero to moderate burnup, two gaps exist:

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

A reactor designed fuel burnup up to 20 at% will have a database weakens substantially for both metallic 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. Thus, there is no whole assembly experience or whole core experience at high burnup. Without the availability of a test reactor such a design to high burnup could not be licensed. Thus, the major gaps for fuel irradiated beyond 10 at% are the following:

1. A need for irradiation of a significant number of prototypic assemblies to high burnup in the steady state conditions.
2. Subject a number of high burn pins to off-normal (TREAT) tests.

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. As shown on Table 3 the technological data base is weak for either oxide or metal fuel that contains substantial quantities of minor actinides. Experiments are underway in the ATR reactor 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 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 with PUREX, which has minimal carry-over of fission products to the reprocessed fuel. Pyro-processing or UREX have the potential for substantial carry-over of fission products. As shown in Table 2 the data base 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, while for metal fuel the migration of lanthanide fission products to the fuel cladding interface may result in low melting compounds. Experiments are underway in ATR to aid in the resolution of these issues. Thus the gap identified in this area is identical to that identified above for fuel that contains additions of minor actinides.

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 carry-over, particularly the issue of lanthanide migration to the fuel-cladding interface in metallic fuel.

The ferritic/martensitic alloys have the potential to solve the irradiation enhanced swelling issue for both cladding and ducts up to at least 150dpa and perhaps 200dpa[5,6]. 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):

1. 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.
2. Recent attempts to obtain a small heat of HT-9 revealed that there are no vendors readily available to produce reactor grade material.

Several gaps were identified in the discussion of fuel performance codes.

1. 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 when adequately validated with existing data.
2. In addition, relatively 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 analyses. It was generally concluded that should an SFR be designed in the near future, based on 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 the design of the rotating plug and the lack of performance data should a large primary electro-mechanical pump be part of the design. The overall materials technological base for the Brayton cycle would require a significant research effort, though this cycle offers many advantages to more traditional power cycles.

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

1. For most of the identified gaps a test SFR such as EBR-II or FFTF is required to enhance the existing knowledge base.

2. The state of the existing knowledge base is uncertain. Operating information, fuel performance data, and fabrication experience exists in a number of locations. Some exists 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.

It is extremely important to preserve the existing data base because without EBR-II, FFTF, and TREAT the 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.

VI. Conclusions

The main conclusion reached by the panel was that an SFR could be designed and licensed based upon the technology base developed from the successful operation of EBR-II and FFTF. However, the design would be constrained within the limitations of the technology base.

From a fuels and materials perspective these limitations are the following:

1. For either oxide or metal fuel a maximum burnup of 10 at%.
2. A peak cladding temperature of 600^{o}C.
3. The use of D9 stainless steel cladding and duct material.
4. A peak irradaition exposure of 100 dpa on the cladding and duct.
5. The use of fresh fuel (fuel with neither additions of minor actinides or fission product carry-over from reprocessing)
6. Limited-load following operation for oxide fuel

For burnups greater than 10 at% for either fuel type, the data are limited up to 20 at% and nonexistent beyond 20 at%. For cladding temperatures above 600^{o}C the strength of D9 diminishes rapidly, and irradiation data for higher strength alloys is sparse. The capability to manufacture components from austenitic steels such as D9 exists, but for advanced alloys the industrial base needs to be developed.

Although neutron exposures for advanced alloys such as HT-9 exist up to 150 dpa, the data are meager. Research is on-going in the national labs to study the effects of minor actinide additions and fission product carry-over for both oxide and metal fuels. However, the work is only partially completed.

Metal fuel has been shown to be robust when subjected to load-following conditions, whereas these data are lacking for SFR oxide fuel.

A primary concern of the panel was the status of the existing technology base for SFR fuels and materials. If a serious attempt were made to license a SFR, could the information be retrieved in a credible form to be used for licensing? Many of the scientists and engineers who were involved in the data generation are no longer in the workforce, and much of the information resides in a number of locations in a variety of formats such as electronic media, internal reports, and publications. It would be prudent to form a task group to assess the state of the technology base and provide recommendations for long-term preservation of the information.

Although SFR fuels and materials have the potential to operate well beyond the limitations expressed above, it is difficult to extend the technology base without the availability of test facilities such as EBR-II, FFTF, and TREAT. Many of the gaps that were identified such as higher burnup, higher dpa, etc, depend on the availability of test facilities. Without domestic facilities and with limited access to diminishing foreign capability, SFR designs will be limited to the existing base that must be preserved for future SFR designs.

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Advanced Sodium Fast Reactor Accident Source Terms: Research Needs

D.A. Powers, B. Clément, R. Denning, S. Ohno, R. Zeyen

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

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Advanced Sodium Fast Reactor Accident Source Terms: Research Needs

D.A. Powers
Advanced Nuclear Energy Programs
P.O. Box 5800, Albuquerque, NM, 87185-0736, USA

Bernard Clément
IRSN/DPAM.SEMIC Bt 702
BP3 13115 Saint-Paul-lez-Durance, FRANCE

Richard Denning
Department of Mechanical Engineering, Ohio State University
201 West 19th Avenue, Columbus, OH 43210-1142, USA

Shuji Ohno
Computational Fast Reactor Engineering Group
Advanced Nuclear System Directorate, Japan Atomic Energy Agency
4002 Narita-cho, O-arai-machi, Ibaraki, 311-1393, JAPAN

Roland Zeyen
Institute for Energy Petten
P.O. Box 3, Building 250, F-13115 Saint-Paul-lez-Durance, FRANCE

ABSTRACT

An expert opinion elicitation has been used to evaluate phenomena that could affect releases of radionuclides during accidents at sodium-cooled fast reactors. The intent was to identify research needed to develop a mechanistic model of radionuclide release for licensing and risk assessment purposes. Experts from the USA, France, the European Union, and Japan identified phenomena that could affect the release of radionuclides under hypothesized accident conditions. They qualitatively evaluated the importance of these phenomena and the need for additional experimental research. The experts identified seven phenomena that are of high importance and have a high need for additional experimental research:

- High temperature release of radionuclides from fuel during an energetic event
- Energetic interactions between molten reactor fuel and sodium coolant and associated transfer of radionuclides from the fuel to the coolant
- Entrainment of fuel and sodium bond material during the depressurization of a fuel rod with breached cladding
- Rates of radionuclide leaching from fuel by liquid sodium
- Surface enrichment of sodium pools by dissolved and suspended radionuclides
- Thermal decomposition of sodium iodide in the containment atmosphere
- Reactions of iodine species in the containment to form volatile organic iodides.

Other issues of high importance were identified that might merit further research as development of the mechanistic model of radionuclide release progressed.

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I. OBJECTIVE

This document describes results of an expert opinion elicitation on research needed to develop a predictive, mechanistic model of the source term for use in the licensing and risk analysis of a sodium-cooled fast reactor (SFR). In this context, the source term is a description of the time-dependent release of radionuclides to the containment and the behavior of these radionuclides in the containment under conditions of an accident. Radionuclides of interest include fission products, capture products and activation products produced during normal operation and during the accident. The expert opinion elicitation focused on the Advanced Burner Reactor being considered by the US Department of Energy [DOE, 2010]. The Advanced Burner Reactor is to be a sodium-cooled fast reactor.

Expert opinions were elicited on the phenomena responsible for the release of radionuclides from reactor fuel and elsewhere to the reactor containment for accidents involving substantial damage to the reactor core. Experts were asked to rank these phenomena according to:

- Importance of the phenomena with respect to the magnitude and timing of radionuclide release, and
- State of current, quantitative understanding of the phenomena.

For this work, only nonproprietary, publically available data were used.

II. BACKGROUND

A. Use of the Accident Source Term in Reactor Regulation

The potential for accidents to cause the release of radionuclides into the public environment is, of course, the source of safety concern with the use of nuclear reactors for power generation, research and actinide transformation. Safety concerns with nuclear power plants are sufficient that a conservative safety strategy termed “defense in depth” has been adopted essentially universally. This strategy requires nuclear plants to have features that prevent radionuclide release and multiple barriers to the escape from the plants of any radionuclides that are released despite preventive measures. Consequently, considerations of the ability to prevent and mitigate release of radionuclides arise at numerous places in the safety regulations of nuclear plants. The effectiveness of mitigative capabilities in nuclear plants is subject to quantitative analysis. The radionuclide input to these quantitative analyses of effectiveness is the source term. All features of the composition, magnitude, timing, chemical form and physical form of accidental radionuclide release constitute the source term.

A substantial body of experimental and analytical information on the accident source term has been developed to support the licensing and regulation of power reactors cooled and moderated by light water. Nearly all the currently operating light water reactors in the USA have been designed and licensed using the so-called “TID-14844 Source Term” [DiNunno, *et al*., 1962; US NRC, 1974a,b]. This description of the accident source term was developed about 50 years ago based on experiments involving furnace heating of irradiated reactor fuel chips. The source term specifies radionuclide release in a severe accident to include 100% of the noble gas fission products (Xe and Kr), 50% of the core inventory of halogens (mostly radioactive iodine) and 1% of all other radionuclides. Halogens are assumed to be predominantly in the form of gaseous molecular iodine though a small fraction is considered to be in the form of airborne particulate. About half of the iodine is assumed to deposit along the release pathway to the containment. All radionuclides other than the noble gases and the halogens are assumed to be released to the containment in the form of aerosol particles. An especially critical feature of the TID-14844 source term is that the radionuclide releases to the containment are taken to be instantaneous. Safety systems have to cope with the full release from the moment of accident initiation. Consequences of the TID-14844 source term include demands for very fast closure of main steam isolation valves, rapid startup of emergency diesels, and safety systems designed to mitigate gaseous iodine.

The first typical use of the TID-14844 source term was for assessing the suitability of proposed sites for nuclear power plants. The regulations (10 CFR Part 100.11) require that applicants for a reactor license hypothesize a substantial radionuclide release to the containment and analyze how much of this radioactivity would escape into the environment. It is noted that the hypothesized release should be equivalent to that expected in an accident involving substantial damage to the core. Maximum doses to an individual at the site boundary over the first two hours of an accident (subsequently changed to the worst two hours of the accident), doses to control room operators of the plant and doses in the low population zone around the plant for 30 days following the accident are evaluated. Evaluations, typically, are done by the applicant and confirmed independently by the regulatory staff.

The siting source term is, then, an element of the regulatory defense-in-depth strategy. Within the context of the current risk-informed, performance-based regulatory strategy, the siting source term is a “structuralist” element of defense in depth. It provides a performance-based mechanism to evaluate the mitigative capabilities of the plant that is independent of reactor fuel, coolant, and nuclear design. The evaluations of doses do take credit for engineered safety features within the containment that are not expected to be compromised by the accident. The leak rate from the plant, despite an accident that is assumed to cause substantial damage to the core, is taken to be the design basis leak rate.

The source term used for siting nuclear power plants has also been used for evaluation of control room habitability issues and qualification of safety related equipment in reactor containments among other features of the regulations.

Radionuclide release to the reactor containment during the accident at Three Mile Island [US NRC, 1981] did not follow closely the prescription of the TID-14844 source term. The discrepancy prompted the US Nuclear Regulatory Commission to undertake a massive investigation of the nature of radionuclide release from power plants during accidents that went beyond the design basis. A significant finding of this research was, of course, that the nature of radionuclide release from a plant depends very much on the accident progression. Phenomena taking place during an accident can affect not only the timing and magnitude of radionuclide release, but also the chemical and physical forms adopted by the radionuclides that reach the containment. The physical and chemical forms of the radionuclides strongly affect the efficiency with which radionuclides released from the reactor fuel reach containment. These properties also affect strongly the likelihood that radionuclides will remain suspended in the containment and available to leak to the environment. A simple prescription of the source term like that provided by the TID-14844 source term was found to be insufficient for the realistic assessment of plant safety and the effectiveness of mitigative measures.

Instead of a source term prescription, mechanistic models of the radionuclide release to containment under accident conditions were devised. A first generation of mechanistic model was the Source Term Code Package [Gieseke, *et al*., 1986]. Subsequently, this model was replaced with the MELCOR computer code [Gauntt *et al*., 2000]. Licensees often use the MAAP computer code for similar analyses [Henry, *et al.*, 1994]. These models track the damage to the core, the release of radionuclides, the transport of radionuclides through the reactor coolant system and the behavior of radionuclides in the containment as functions of the conditions prevailing as a result of processes taking place during the reactor accident. There have been extensive efforts to validate the predictions of the source term models by comparisons to small-scale, separate-effects, tests and large-scale integral tests [Allelein *et al*., 2009].

Many calculations with the source term and accident phenomena models showed some clear trends for both pressurized water reactors and boiling water reactors. These trends have been used to formulate an alternative to the TID-14844 source term for regulatory purposes. The alternative is often called either the "alternative source term model (AST)" or the "NUREG-1465 source term" [Soffer *et al*., 1995]. This alternative source term divides radionuclides into eight chemical classes based on similarity of chemistry under accident conditions:

- Noble Gases (Xe, Kr)
- Halogens (I, Br)
- Alkali Metals (Cs, Rb)
- Tellurium Group
- Ba & Sr Group (Ba, Sr)
- Refractory Metals (Ru, Mo, Pd, Rh)
- Lanthanides (La, Y, Pm, Nd, etc.)
- Cerium Group (Ce, Pu, U, Zr, etc.)

The behaviors of radionuclide elements within a given class are assumed to be sufficiently similar that these behaviors can be approximated adequately by modeling the behavior of a representative element. Some effort has been made to select the representative element of each class in terms of its radiological importance. There are, of course, many metrics for the radiological importance of a radionuclide element. Results of one effort [Alpert *et al.*, 1986] to distinguish the radiological importance of elements are shown in Table 1. This table shows the short-term and long-term consequences of equal release fractions of various radionuclide elements normalized to the radiological consequences of iodine for short-term effects and cesium for long-term effects.

The alternative source term also divides a "representative" reactor accident into several time intervals that are typical of the progression of severe accidents in light water reactors but are not applicable necessarily to sodium-cooled fast reactors:

- **Coolant Release Phase:** The time period between the initiation of an accident and the onset of radionuclide release from the fuel. During this period, the coolant boils from the core. There can be some modest amount of radionuclide release to the containment during this period as a result of contaminated liquid carryover during the boiling. This modest radionuclide release is not quantified or considered in subsequent analyses of the radionuclide behavior when the alternative source term is used in regulatory activities.

- **Gap Release**: Once the coolant level drops sufficiently below the top of the core, metal alloy cladding on the fuel will expand ("balloon") and rupture. Radionuclide gases and vapors that have accumulated in the fuel rod plenum and the gap between fuel and the cladding will vent promptly once the cladding ruptures.

- **In-vessel Release**: This is radionuclide release from the fuel that takes place following Gap Release as the fuel heats up and begins to interact chemically with residual cladding, liquefy and flow out of the core region – eventually reaching the lower plenum of the reactor vessel.

- **Ex-vessel Release**: Once core debris penetrates the reactor pressure vessel and cascades into the reactor cavity, it can interact with concrete and accumulated water in the reactor cavity. Both of these processes have been shown to cause some release of radionuclides as well as release of large amounts of nonradioactive aerosol.

- **Late In-vessel Release**: In parallel with the Ex-Vessel Release phase, radionuclides deposited in the reactor coolant system can heat and revaporize or be mechanically resuspended. The vapors or resuspended particles can be carried into the reactor containment by natural convection.

The alternative accident source term specifies fractions for each of the eight chemical classes of radionuclides that reach the containment during each of the accident phases. An example specification for pressurized water reactors is shown in Table 2. Release rates are taken to be constants peculiar to each radionuclide class and each accident phase. For regulatory analyses involving design basis reactor accidents, only releases associated with the Gap Release phase and the In-vessel Release phase of the representative accident need to be considered [US NRC, 2000].

It is assumed that the noble gases will remain gaseous throughout the accident. With the exception of the Noble Gas and the Halogens classes, other radionuclides are expected to enter the containment as aerosol particles. Most of the halogens are assumed to also be released to the containment as aerosol particles, but 5% of the released material is assumed to be in gaseous form.

Some fraction of this gaseous iodine is taken to be a volatile organic iodide such as methyl iodide (CH_3I). The rest is usually assumed to be molecular iodine (I_2).

The alternative source term does not provide all the source term information needed for the defense-in-depth analysis of the containment and safety system performance. It does not provide, for instance, the particle size distribution for aerosols, nor does it provide the releases of nonradioactive materials that will contribute to the aerosol mass loading of the containment atmosphere. Analysts are required to define and justify these aspects of the source term that are considered particular to the plant in question. Models of varying levels of mechanistic detail are used to do this.

Use of the alternative source term is optional for current licensees. The alternative source term has proven to be quite attractive to licensees of the US Nuclear Regulatory Commission [J.Y. Lee, 2006]. The more realistic timing and physical forms of the radionuclide releases allow better design and analyses of safety systems to mitigate radionuclide release outside of containment. The alternative source term is not used typically for plant-wide risk analyses. Source terms derived from mechanistic models such as MELCOR or MAAP are used for risk analyses.

Future light water reactors are required to use the alternative source term for design basis accident analyses or to provide a justified alternative. Indeed, the applicability of the alternative source term is highly constrained. It is not considered applicable to light water reactors using mixed oxide (MOX) fuel or even light water reactors using conventional, low enrichment, urania fuel to burnups much beyond about 45 GWd/t. It is certainly not applicable to fast reactors or reactors using coolants and moderators different than light water.

The evolution of the source term for light water reactor licensing and risk analysis does provide some lessons useful in the development of a regulatory source term for sodium-cooled fast reactors:

- Licensing source terms are better based on mechanistic analyses of radionuclide behavior under accident conditions than being based on bounding or very conservative assumptions. The problem encountered with such bounding source terms is that mitigative systems are designed to cope with radionuclides of types and amounts that are unlikely to be present during an accident. Systems may or may not cope with more realistic amounts and chemical forms of the radionuclides.
- It is possible to develop acceptable source terms based on chemical classes rather than based on specific radionuclides.

- Timing is an essential feature of the source term specification. To develop this timing, the accidents can be broken down into distinct phases according to the dominant accident processes taking place. The timing of accident phases adopted in the alternative source term is based on the statistical distributions of the timing calculated for a wide range of hypothetical accidents.

Based on the experience gained over the decades since the accident at the Three Mile Island nuclear power plant and the development of modern accident source terms for light water reactors, it is evident that there are advantages associated with the development of a mechanistic source term model for sodium-cooled, fast reactors. It is not possible to adopt or to adapt the NUREG-1465 source term. This source term does provide a useful framework for considering the development of a mechanistic source term model for sodium-cooled reactors to be used in both licensing activities and risk analyses.

Table 1. Relative Importances of Various Radioactive Elements in Reactor Fuel [Alpert, *et al*., 1986]. Consequence analyses to develop these relative importances were based on assuming equal release fractions (10%) of each radionuclide individually.

	Early Exposure* (normalized to iodine)			**Long-term Exposure** (Normalized to cesium)
Element	**4 hr. bone marrow dose**	**24 hr. bone marrow dose**	**Lung dose**	**Latent cancers**
Co	0.007	0.008	0.01	0.07
Kr	0.2	0.1	0.04	0.001
Rb	0.0002	0.0002	0.0002	0.00001
Sr	**1.0**	**0.7**	**0.9**	**0.7**
Y	0.07	0.07	**3.5**	0.4
Zr	**1.0**	**1.0**	**2.0**	**0.7**
Nb	0.3	0.3	0.4	0.2
Mo	0.1	0.1	**0.7**	0.06
Tc	0.02	0.03	0.03	0.06
Ru	0.3	0.3	**3.0**	**1.0**
Rh	0.01	0.01	0.08	0.004
Sb	0.06	0.06	0.1	0.004
Te	**0.8**	**0.8**	**0.6**	0.1
I	**1.0**	**1.0**	**1.0**	0.1
Xe	0.02	0.01	0.005	0.0001
Cs	0.15	0.14	0.09	**1.0**
Ba	**0.6**	**0.5**	**0.6**	0.2
La	**1.1**	**1.2**	**1.6**	0.08
Ce	0.1	0.2	**8.0**	**2.0**
Pr	0.004	0.003	0.8	0.08
Nd	0.03	0.03	0.3	0.03
Np	**1.6**	**1.4**	**5.0**	0.04
Pu	0.004	0.003	**1.4**	**3.0**
Am	0.002	0.001	0.01	0.03
Cm	**0.6**	**0.4**	**5.0**	**1.1**

* early exposure via cloud, inhalation and either 4 or 24 hours of groundshine.

Table 2. Alternative Source Term for Pressurized Water Reactors [Soffer *et al.*, 1995].

	Gap Release	In-vessel Release	Ex-vessel Release	Late In-vessel Release
Duration (hours)	**0.5**	**1.3**	**2.0**	**10.0**
Release Fractions of Radionuclide Groups				
Noble Gases (Kr,Xe)	**0.05**	**0.95**	**0**	**0**
Halogens (Br,I)	**0.05**	**0.35**	**0.25**	**0.1**
Alkali Metals (Rb, Cs)	**0.05**	**0.25**	**0.35**	**0.1**
Alkaline Earths (Sr, Ba)	**0**	**0.02**	**0.10**	**0**
Tellurium Group (Te)	**0**	**0.05**	**0.25**	**0.005**
Refractory Metals (Ru, Pd, Re, etc.)	**0**	**0.0025**	**0.0025**	**0**
Lanthanides (Y, La, Sm, Pr, etc.)	**0**	**0.0002**	**0.005**	**0**
Cerium Group (Ce, Pu, Zr, etc.)	**0**	**0.0005**	**0.005**	**0**

B. Background on Sodium-cooled Fast Reactor Source Terms

Issues of radionuclide release during accidents at sodium-cooled fast reactors have received research attention in the past. Much work took place well before the accident at Three Mile Island and the recognition of the value of mechanistic source term models. Past work readily available in the literature addressed two major issues:

- Behavior of aerosols in the reactor containment
- Retention in the sodium coolant of radionuclides released from reactor fuel.

Experimental investigations of the behavior of aerosols in reactor containments were instrumental in the development of highly sophisticated aerosol behavior models such as MAEROS [Gelbard, 1982] and the CONTAIN-LMR computer code [Murata, 1993]. It is noteworthy that the MAEROS model developed initially for sodium-cooled reactors is used today for analysis of aerosol behavior in light water reactor accidents [Gauntt, *et al.,* 2000]. Consequently, the model has been maintained as new data pertinent to light water reactors and superior aerosol physics models have been obtained.

Investigations of radionuclide retention in sodium coolant were fairly integral in nature and exploratory in intent [Jordan and Ozawa, 1976; Schütz, 1980; Jordan, 1976; Koch *et al*., 1990; Sauter and Schütz,1983; Berlin *et al.*, 1982]. By in large, these investigations did show some retention of radionuclides in sodium, but they also showed that there would be some release from sodium pools of especially the more volatile radionuclides such as isotopes of cesium, iodine, tellurium and strontium. The earlier studies also showed that there could be some mechanical release of the less volatile radionuclides.

In addition to these studies directed specifically at source terms, there have been a variety of studies of sodium spray fires, sodium pool fires, sodium interactions with concrete and molten fuel interactions with liquid sodium (See bibliography in Appendix B). These studies certainly examined the accident processes of fire and materials interactions. Most did not specifically address the radionuclide releases associated with processes.

Recently, there has been a resurgence of technical interest in source terms associated with accidents at sodium-cooled fast reactors. Japanese investigators have reported on some very sophisticated investigations pertinent to the development of mechanistic source term models (see Appendix B). There is an indication that research into source term issues for sodium-cooled fast reactors may be expanding in Europe [Fiorini, *et al*., 2009]

There has also been a resurgence of public interest in accident source terms for sodium-cooled fast reactors in the USA. Some of this interest stems from questions about the magnitude of the radionuclide release to the containment during an accident that damaged fuel at the Sodium Reactor Experiment in the Los Angeles Basin [Lochbaum, 2006].

III. APPROACH

A. Expert Opinion Elicitation

From the discussion above in the section entitled Background, it is apparent that mechanistic descriptions of accident source terms are useful. Furthermore, source term models for accidents in sodium-cooled fast reactors commensurate with those available for light water reactors do not now exist. The body of data and analyses that exists to support light water reactor source term modeling does not exist in the public domain for sodium-cooled fast reactors. At issue, then, is what experimental and analytical research needs to be done to support development of a sodium-cooled, fast reactor, source term model adequate for the meeting needs for reactor licensing and risk assessment. Models adequate for these purposes require a level of accuracy that is consistent with the uncertainties associated with the prediction of the progression of severe accidents. Also, the models need to be consistent with available experimental information and the uncertainties in model predictions have to be quantified.

The approach adopted here to identify the information needed to construct an adequate source term model for a sodium-cooled reactor is based on an expert opinion elicitation. Expert opinion elicitations have been much used in the resolution of source term and accident phenomenology issues associated with light water reactors. Use of expert opinion elicitations became much more formalized in the uncertainty analysis of a Nuclear Regulatory Commission study of the risk associated with five representative nuclear power reactors [US NRC, 1990]. Formal guidelines have been developed for the expert opinion elicitation process [Budnitz, *et al*., 1997]. To the extent possible, these guidelines have been followed in the process reported here. But, as will be clear from the ensuing discussions, not all the guidelines could be followed and it was necessary to adapt the recommended process.

Elements of the expert opinion elicitation process that were adopted are:

- **Selection of Experts**: Experts having a wide range of backgrounds but still knowledgeable in issues of reactor accident source terms were identified and agreed to participate in the expert opinion elicitation. Brief biographies of these experts are provided in Appendix A of this document. The experts come from several countries – two of which have active sodium-cooled fast reactor programs. The experts have differing regulatory perspectives. All are currently active in experimental or analytical aspects of reactor accident source terms. One expert is employed by an academic institution. Others are employed by the equivalents of national laboratories.

- **Problem Definition**: The experts were acquainted with the overall issue of research needs for a mechanistic source term for sodium-cooled fast reactors and provided with background information. The experts also provided additional background information. It was emphasized that the expert opinion elicitation was to define research needs and not to actually define an accident source term for sodium-cooled fast reactors. It was also emphasized that only publically available data and analyses should be invoked explicitly in the formulation of opinions.

- **Background Information**: A bibliography of background information collected for the expert opinion elicitation is provided in Appendix B to this document.

- **Problem Specification:** Additional, detailed questions were provided to the experts for their consideration prior to the expert opinion elicitation. A draft structure for the elicitation was provided for expert consideration.

- **Expert Meeting and Reformulation of the Approach:** The initial expert opinion elicitation took place at a face-to-face meeting. One expert was unable to attend due to an airline traffic controller strike. He was able to provide input electronically. At the meeting, the experts revised the structure for the elicitation and revised the major areas of interest. They agreed to a generic set of possible accident events discussed later in this chapter. They provided their independent assessments of the important source term phenomena and information needs associated with these phenomena. Rankings for both importance and information needs were on a high-medium-low scale. These qualitative rankings were made to a figure of merit which was the amount of radioactive material that is suspended in the containment atmosphere. The definitions of the ranking levels for the importance of phenomena were taken to be:

  High Importance: The phenomenon is essential to consider in the development of a mechanistic source term model.

Medium Importance: The phenomenon should be considered in the development of a mechanistic source term model.
Low Importance: The phenomenon can be considered to improve the accuracy and defensibility of a mechanistic source term model, but will not greatly change predicted results.

The definitions of the ranking levels for the need for further research were taken to be:

High Need for Research: there are insufficient data or understanding to formulate confidently even an approximate model.
Medium Importance for Research: Further research on this phenomenon would greatly improve the quantitative accuracy of a mechanistic model.
Low Importance for Research: Further research and data on this phenomenon could be used to refine a mechanistic source term model.

The experts discussed the rationale for the rankings. Some revisions were made to rankings provided by individual experts based on these discussions to develop a consensus ranking.

- **Development of Results:** Results of the initial expert opinion elicitation were accumulated. They were used to derive a consensus ranking. For the development of the consensus, all experts were treated equally except where they had indicated unfamiliarity with the issue usually by not providing a response.

- **Review and Amendment of the Results:** Experts were provided for review the draft results of the expert opinion elicitation and allowed to change their individual evaluations and to argue for changes in the consensus rankings.

B. Constraints

Several constraints on the expert opinion elicitation prevented adoption of all the guidelines usually associated with such an undertaking. These constraints include:

- **Plant Design**: No decision has been made on whether the Department of Energy prefers a sodium-cooled reactor of the loop type or the pool type. In the expert opinion elicitation, both types of reactors had to be considered. Most of the source term work done to date for sodium-cooled reactors has focused on reactors of the loop design. In such designs, accidents that involve a loss of coolant flow can play an important role in the overall risk. Pool type reactors, on the other hand, are usually designed so that natural convection provides sufficient cooling for the core once the reactor is no longer critical. Although the radionuclide inventory available for release depends very much on the reactor power, the processes that must be modeled are largely the same regardless of the power level. It was assumed for the purposes of the expert opinion elicitation that the reactor power is 2000 MWth.

- **Reactor Fuel**: A number of fuel forms have been considered for sodium-cooled fast reactors including oxide, carbide, nitride and metallic fuels. At this time, the fuel forms of most interest for immediate application are a conventional, metal-clad, mixed oxide (urania-plutonia-actinide oxide) type and a metallic (Zr-U-Pu-actinide) type. Most previous source term work has been on conventional oxide fuels. Much less is known about radionuclide releases under accident conditions for metallic fuels. Both fuel types had to be considered in the expert opinion elicitation. The sodium-cooled fast reactor of interest here may be used for the destruction of minor actinides in irradiated light water reactor fuel. The effects of radionuclides on source terms for the oxide and metal fuels were not addressed explicitly in this study. Radionuclide inventories of the fuel were assumed to be the same for both fuel types and were taken to be those estimated by Kim and Yang [2006] which are shown in Table 3.

- **Accident Type**: Details of the progression of reactor accidents are known to have a very important influence on the release of radionuclides from the fuel and the subsequent behavior of the released radionuclides. Although it is possible to identify some generic characteristics of severe accident scenarios for sodium-cooled fast reactors without having a specific design, it is impossible to define “risk-dominant” accident types. Consequently, the experts constructed a generic accident scenario that would not be directly representative of the wide variety of possible scenarios, but does include all the major accident processes likely to influence the accident source term. This generic scenario is described in the next section of this chapter.

Table 3. Estimated Radionuclide Inventories of a 2000 MWth Sodium-cooled, Fast Reactor [Kim and Yang, 2006].

Element	Mass (kg)
Noble gas	0.621
Halogen	0.127
Alkali Metal	71.4
Alkaline Earths	15.5
Ruthenium	3.39
Molybdenum	0.0015
Technetium	0.0019
Zirconium (fission product)	3.47
Uranium	9249
Plutonium	7147
Neptunium	99.1
Americium	626
Curium	316

C. Generic Accident Scenario

As a framework for the identification of phenomena that can affect the source term, the experts developed a hypothetical scenario that is not intended to be representative of any particular accident in a sodium-cooled fast reactor. Indeed, it is unlikely that a real accident would include all the elements listed in the generic accident sequence. The sequence was devised by the experts to serve as a framework for the identification of phenomena that should be considered in the development of a mechanistic model of radionuclide releases under accident conditions.

It is important to understand how the expected progression of accidents in sodium-cooled fast reactors differs from the progression of accidents in light water reactors. For most light water reactor severe accidents, severe fuel damage is initiated once coolant is lost from the core region. The fuel then heats up due to both decay heat and the exothermic energy release from steam reacting with the zirconium alloy cladding. The fuel pins rupture and noble gases as well as some volatile radionuclides in the fuel-cladding gap and the plenum of the fuel pin escape into the steam and hydrogen atmosphere of the reactor coolant system. The magnitude of this early release is typically small in comparison to the releases of radionuclides in later stages of a light water reactor severe accident. Continued progression of the light water reactor accident leads to the formation of molten mixtures of fuel and cladding (Zr-U-O) that flow (in the jargon “candles”) down the rods. Eventually, an encrusted pool of molten core debris can accumulate within the core region. This degradation of the reactor fuel takes place over a protracted period and is accompanied by significant releases of radionuclides into the steam-hydrogen atmosphere of the reactor coolant system. Reactivity insertion accidents involving more rapid heating and dispersal of fuel are possible and are considered in the analysis of reactor design bases. Such reactivity insertion accidents, however, are not major contributors to risk and do not figure prominently in the definition of an accident source term used for either regulatory or risk analyses.

In the later stages of a light water reactor severe accident, the molten core debris can relocate to the lower plenum and penetrate the reactor vessel. Core debris spilling from the vessel will attack structural concrete in the reactor cavity. This attack releases gases that sparge through the core debris which leads to additional releases of radionuclides.

The conditions leading to severe fuel damage in a sodium-cooled fast reactor appear to be more varied, are design dependent, and are dependent on the type of reactor fuel. It is less likely that coolant will be lost from the reactor vessel and expose the fuel. The operating pressure of a sodium-cooled fast reactor typically is only slighter greater than atmospheric pressure. Even at atmospheric pressure, the coolant is sub-cooled under normal operating conditions, so the coolant would not “flash” should there be a break in the coolant system. Typically, sodium-cooled fast reactors provide a guard vessel or guard piping to prevent core uncovery in the event that the primary coolant system does rupture allowing sodium to leak from the primary system. Thus, a breach in the reactor coolant system is not very likely to occur and lead to core uncovery. This is particularly true in the case of a sodium-cooled fast reactor with a pool design. Extended loss of heat removal capabilities at a plant could lead to evaporation of the sodium coolant and core uncovery. Uncovery would occur after a very long time because of the large heat capacity of the primary system inventory of coolant. Release of radionuclides from fuel following uncovery would be into either a sodium vapor atmosphere or an oxidizing atmosphere depending on the mode of failure of the primary system boundary. A rupture to this boundary that allowed an oxidizing environment to develop around the exposed fuel would add the complications of exothermic reactions that would be particularly severe in the case of proposed metallic fuels.

As shown by the accident at Fermi-I [Page, 1979], blockage of coolant flow through the reactor fuel by a foreign object could result in fuel damage in a sodium-cooled fast reactor. To prevent this, most sodium-cooled fast reactor designs include features to prevent or reduce the likelihood of a flow blockage. Should a flow blockage occur, molten fuel could exist within a fuel bundle, perhaps for an extended time. However, radionuclides escaping the fuel would be released into an overlying pool of sodium which could trap most of the radionuclides save for the noble gases (Kr, Xe).

Accident analyses for sodium-cooled fast reactors have focused in the past on transient events in which there is a failure to “scram” the reactor. This focus has been driven by the potential severity of events and particularly the potential for such an event to lead to an energetic criticality excursion. In today’s risk-informed regulatory environment, it is less clear that accident scenarios of this type will dominate the perceived risk of sodium-cooled fast reactors. Transient tests performed in the EBR-II reactor indicated that sodium-cooled fast reactors can be designed with inherent characteristics that shutdown the reactor even without control rod insertion. Also, the reliability of reactor protection systems is thought to be much improved now. Nevertheless, in the more severe transients involving failure of the reactor protection system, melting of the fuel begins within a fuel pin prior to failure of the cladding. The location and the mode of subsequent cladding failure can affect the progression of the accident. Molten fuel may be ejected into the coolant by fuel pin depressurization following cladding failure. This ejected fuel could be swept from the core region by the flow of coolant and could then freeze somewhere in the reactor coolant system. Large quantities of noble gas radionuclides that accumulate in the fuel pin plenum during normal operations will vent from the failed fuel pin along with radionuclide vapors produced by the melting of the fuel. This mixture of radionuclides will vent into the liquid sodium coolant pool as bubbles. Some fraction of the radionuclides will be scrubbed from the bubbles by the liquid sodium which will attenuate the prompt release of radioactivity to the gas phase of the reactor coolant system.

Depending on the conditions in the fuel channels at the time of fuel failure, it is possible that fuel degradation and molten fuel pool formation could progress along lines akin to the “candling” process expected for accidents in light water reactors. Development of a very large pool of core debris may not occur. Formation and growth of a molten pool within the core of a sodium-cooled fast reactor could lead to a recriticality and an energetic disassembly of the accumulated mass of core debris. High temperatures produced in the core debris could cause volatilization of even the more refractory radionuclides. The vapors produced during the temperature excursion would enter the overlying sodium which, again, would limit the prompt release to the reactor coolant system atmosphere.

Radionuclide release to the sodium coolant would continue after the degraded core debris quenched. Sodium will leach or chemically extract radionuclides from the core debris. The extracted radionuclides can be released from the sodium pool to the atmosphere of the reactor coolant system.

In light of these potential events in sodium-cooled fast reactors, the elements of the generic accident sequence are:

- Accident initiating event, possibly compounded by system failures, leading to fuel failure.
- Energetic event that damages fuel in the reactor core
- A period of slow degradation of reactor fuel.
- Quiescent pool in which radionuclides are leached or dissolved from fuel debris dispersed in sodium coolant
- Transport in the reactor coolant system
- Radionuclide behavior in the containment
- Sodium fires (spray and pool) and sodium interactions with concrete.

Phenomena and processes within each of these elements of the generic accident sequence that affect radionuclide behavior are discussed in the subsections that follow. Schematic depictions of the more important processes are shown in Figures 1-4. Note that the generic scenario does not address scenarios involving the slow boiling of coolant to expose fuel that are commonly addressed in the safety analysis of light water reactors. Many of the phenomena that would arise in such scenarios, however, are addressed. The one exception is that exposure of fuel to a highly oxidizing atmosphere is not considered. The radionuclide release associated with the exposure of fuel to a highly oxidizing atmosphere after coolant boiloff was not considered by the experts. Metallic fuels especially would react vigorously with oxidants and release radionuclides extensively. It was generally thought that such exposures of the fuel to highly oxidizing conditions were of very low likelihood.

1. Initiating Event

The source term experts did not deliberate at length on the initiating events that could lead to an accident that could lead to radionuclide release to the reactor containment. The experts did proceed recognizing that accidents could be initiated by a variety of events that lead either to overheating of the fuel or undercooling of the cladding. Prompt progression of the accident would occur in accidents initiated by inadvertent reactivity insertions or transients without scram. Slower progression would be expected for accidents initiated by station blackout or loss of ultimate heat sink. The nature of the initiating event could affect substantially the timing of radionuclide release.

2. Fuel Thermal Excursion and Clad Failure Event

Some initiating events could lead to a temperature excursion in the reactor fuel. The amount of the fuel involved in the excursion could vary from as little as one fuel assembly to a major fraction of the entire core. Sudden heating of the core could cause the metal alloy cladding on the fuel to rupture and vent the radionuclides that accumulated during operations in the fuel-cladding gap and the plenums of the fuel rods. These would be primarily fission gases (Xe and Kr), but would also include vapors of the more volatile radionuclides such as cesium, iodine and tellurium. Depressurization of the fuel rod following clad breach could lead to gas flow velocities sufficiently high to entrain some condensed particles or droplets. In the case of oxide fuel, small, solid fuel particles could be entrained. For metallic alloy fuels, droplets of sodium used to thermally “bond” the fuel and the cladding could be entrained. Very likely this thermal bond sodium could be contaminated with dissolved radionuclides. A sufficiently energetic initiating event could lead to expulsion of fuel particles or droplets through the clad breach and into the coolant channel.

The combination of gases vented from the fuel rods, particles and droplets entrained in the flow, and the vaporization of sodium coolant would produce a large gas bubble that would rise toward the surface of the coolant pool. The bubble would encounter fuel rods and in-core structures that could cause it to breakup into smaller bubbles. Within the bubbles, mass transport of vapors and particles could lead to deposition of radionuclides on the bounding, sodium surfaces of the bubbles.

3. Extended Degradation Phase

Following fuel failure in a temperature excursion, fuel could remain sufficiently hot that there would be continued degradation of the fuel by melting and alloying. Accidents initiated by events that did not involve a prompt temperature excursion could reach this stage once cladding temperatures exceeded the sodium boiling point. The progression of fuel damage during this accident phase would be different for oxide fuel and metal fuel. For oxide fuel, the progression would bear some similarities to the degradation of light water reactor fuel during severe accidents. The heating, however, would not be driven by exothermic reaction of the coolant with the cladding alloy as it is in a light water reactor. Cladding on fast reactor fuel is expected to be a stainless steel alloy or a similar alloy that does not vigorously react with the sodium coolant even at accident temperatures. There would also not be the very strong chemical interactions of the molten stainless steel cladding alloy with the fuel the way there is between molten zirconium alloys and fuel in light water reactor accidents. There would be some chemical interactions especially between chromium constituents of the cladding alloy and the fuel. Melting of the oxide fuel would not be expected at temperatures as low as that observed in experiments with light water reactor fuel since a low temperature monotectic reaction between clad and fuel would not occur.

The extended degradation phase in the case of metallic fuel would be quite different. Even before clad melting there could be strong chemical interactions between iron and nickel in the cladding and zirconium, plutonium and uranium in the reactor fuel. These strong interactions would be driven by the exothermic formation of Laves phases and the heat of dilution of clad constituents in fuel and vice versa. Eutectic reactions in the combined system of elements could lead to melting and cladding failure at relatively low temperatures.

During the extended degradation phase of accidents with either metallic or oxidic fuel there would be a diffusive release of radionuclides from the fuel. That is, radionuclides would diffuse through the fuel and vaporize into the gas phase. While the fuel was solid, radionuclides would diffuse to grain surfaces. Saturation of the grain surfaces would lead to linkage of the surfaces with porosity in the fuel and radionuclide vapors could percolate into the coolant channel. Once the fuel melted, radionuclides dissolved in the fuel would diffuse to the melt surface and vaporize. Transport to free surfaces would be facilitated by growth of fission gas bubbles and foaming of the molten or plastic fuel.

Molten core debris could drain from the core region into the lower plenum of the reactor vessel. The quenched droplets of core debris would form a debris bed. Or, molten core materials could freeze on core structures and accumulate. The accumulated mass could eventually be of sufficient size that it would remelt to form a molten pool bounded by frozen crusts. Natural circulation of the molten pool within the frozen crusts would transport heat to the boundaries. If the molten pool included a second, insoluble phase such as molten metal in the case of oxide fuel, the so-called “focusing effect” could impose particularly high heat loads on the radial boundaries of the molten pool. The solidified material making up the boundaries could melt and there could be a catastrophic failure of the retaining boundaries of the pool. The molten mass of core debris would then pour into the residual coolant.

4. Energetic Events

For scenarios involving the accumulation of fuel material in the core region, there is the possibility of achieving a supercritical state and producing a reactivity excursion. If the events lead to further compression of fuel, a hypothetical core disruptive accident could result with attendant release of energy and conversion of thermal energy to mechanical work. Under these conditions, the temperature of the fuel could increase substantially. At elevated fuel temperatures, substantial releases of radionuclides and even fuel vaporization could occur. Historically, this scenario has been of high interest because of the associated risk of primary system failure and even containment failure. In a risk-informed regulatory environment, system designers develop safety cases to demonstrate that the likelihood of a hypothetical core disruptive accident is so low that it falls below a threshold of risk importance.

When molten, oxide fuel flows into sodium, energetic interactions can take place [Armstrong, *et al*., 1976; Chu, *et al*., 1979]. Such energetic interactions would vaporize sodium and quench core debris into high surface area solids that would settle to the lower plenum of the reactor vessel. The interactions might have other effects on the progression of the accident and the integrity of the reactor coolant system that were not discussed by the source term expert panel. Of more interest to the panel was the possibility that energetic interactions could also enhance the transfer of radionuclides from the fuel to the coolant.

5. Quiescent Pool

Regardless of the initial phases of a severe fuel damage accident, it is likely that eventually some fuel debris will accumulate on the bottom head of the reactor vessel. This debris bed could be coolable or not. A bed of core debris would be susceptible to radionuclide leaching or dissolving into the ambient sodium. If the debris bed is not coolable, radionuclide release to the sodium might be by way of radionuclide vaporization into boiling sodium vapor. Transport of radionuclides away from the debris would be limited by the debris bed porosity and flow of sodium into the debris. In the deliberations of the source term experts, it was assumed that details of thermal hydraulics of sodium in debris and the issues of debris coolability would be available to a model of the source term. These issues were not discussed by the source term experts.

Even milder accidents that lead to clad rupture but do not involve extensive fuel disruption would also involve radionuclide leaching or dissolution into molten sodium. Mass transport of sodium into and out of a breach in the cladding would limit the rate of attack. The expert panel assumed that information on the extent of cladding damage and the flow of sodium into and out of the damaged cladding would be available from other sources for a mechanistic model of radionuclide behavior. These issues were not discussed further by the source term experts.

Radionuclides will dissolve in the sodium to the point of saturation. There is a data base on the solubility of many of the radionuclides of interest in pure sodium. But, especially in the later stages of an accident, it is possible that air may leak into the sodium or that some water vapor will react with the sodium. Instead of being pure sodium, the coolant will be a Na(O) solution. Data on the effects of dissolved oxygen in sodium on radionuclide solubility are less abundant.

Sodium that washes the debris bed and leaches radionuclides from the fuel will be hotter than the bulk sodium pool. Consequently, as the saturated sodium flows away from the debris bed and cools, saturation solubilities could be greatly exceeded. Dissolved radionuclides could precipitate onto structural surfaces in the sodium pool or nucleate to form particles suspended in the pool. Fuel particles that are small enough could also be entrained in the sodium flow.

The pool of residual sodium, then, will contain both dissolved and suspended radionuclides. These radionuclides can be released to the atmosphere by two mechanisms:

- vaporization
- mechanical entrainment.

Vaporization will occur at the free surface of the sodium pool. The driving force for vaporization is the free energy differences between dissolved radionuclides and the vapor. Because of the high surface tension of sodium, there may be a tendency for some radionuclides to accumulate or avoid the surface, so that concentrations used in the analysis of the free energy of the dissolved radionuclides may not be the same as bulk concentrations.

The rate of radionuclide vaporization will be limited both by the driving force and by the mass transport limits on both the gas-phase and sodium-phase side of the boundary. The mass transport of radionuclide vapors will be affected by the simultaneous vaporization of sodium. The sodium will be, of course, radioactive itself and contains both ^{22}Na ($t_{1/2}$ = 2.6 yrs) and ^{24}Na ($t_{½}$ = 15 hrs.). Perhaps of more importance, the vaporization of sodium can enhance or retard the vaporization of radionuclides into a cover gas. In any event, the vaporization of sodium will make the surface of the sodium pool slightly cooler than the bulk sodium. This cooling further complicates the analysis of vaporization processes.

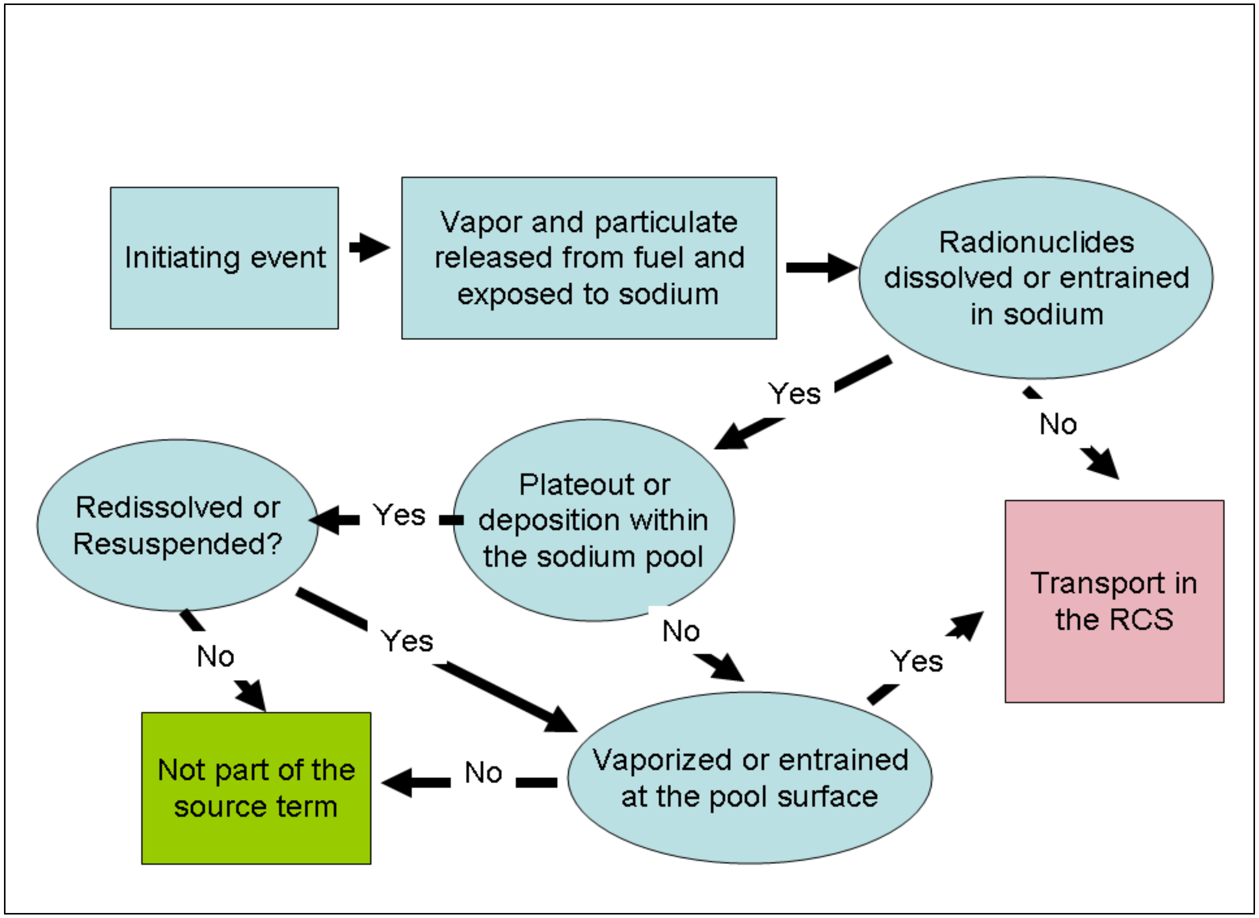

Figure 1. Schematic representation of the major events of radionuclide behavior following an initiating event that damages fuel.

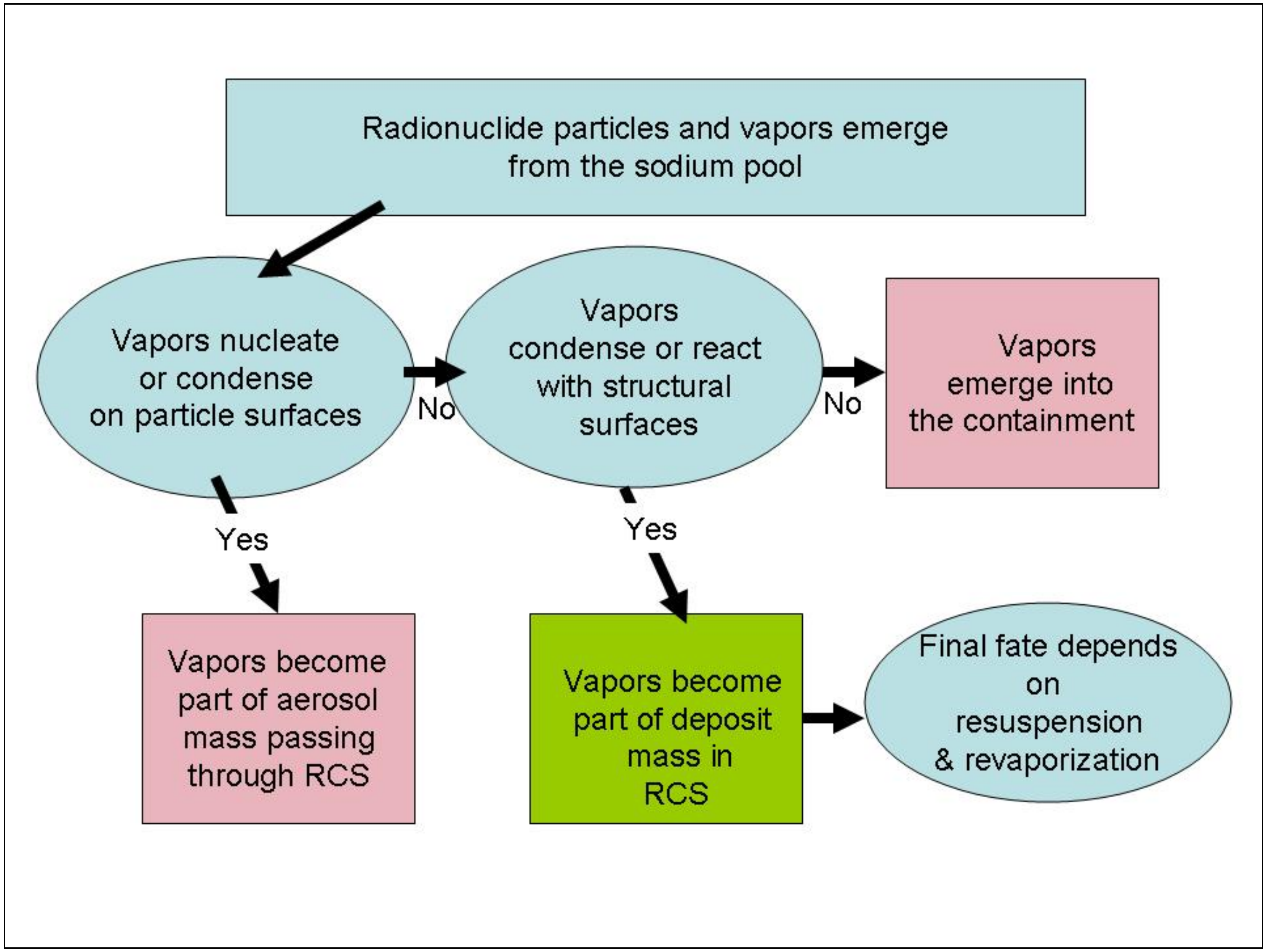

Figure 2. Behavior of vapors released from the surface of the sodium pool.

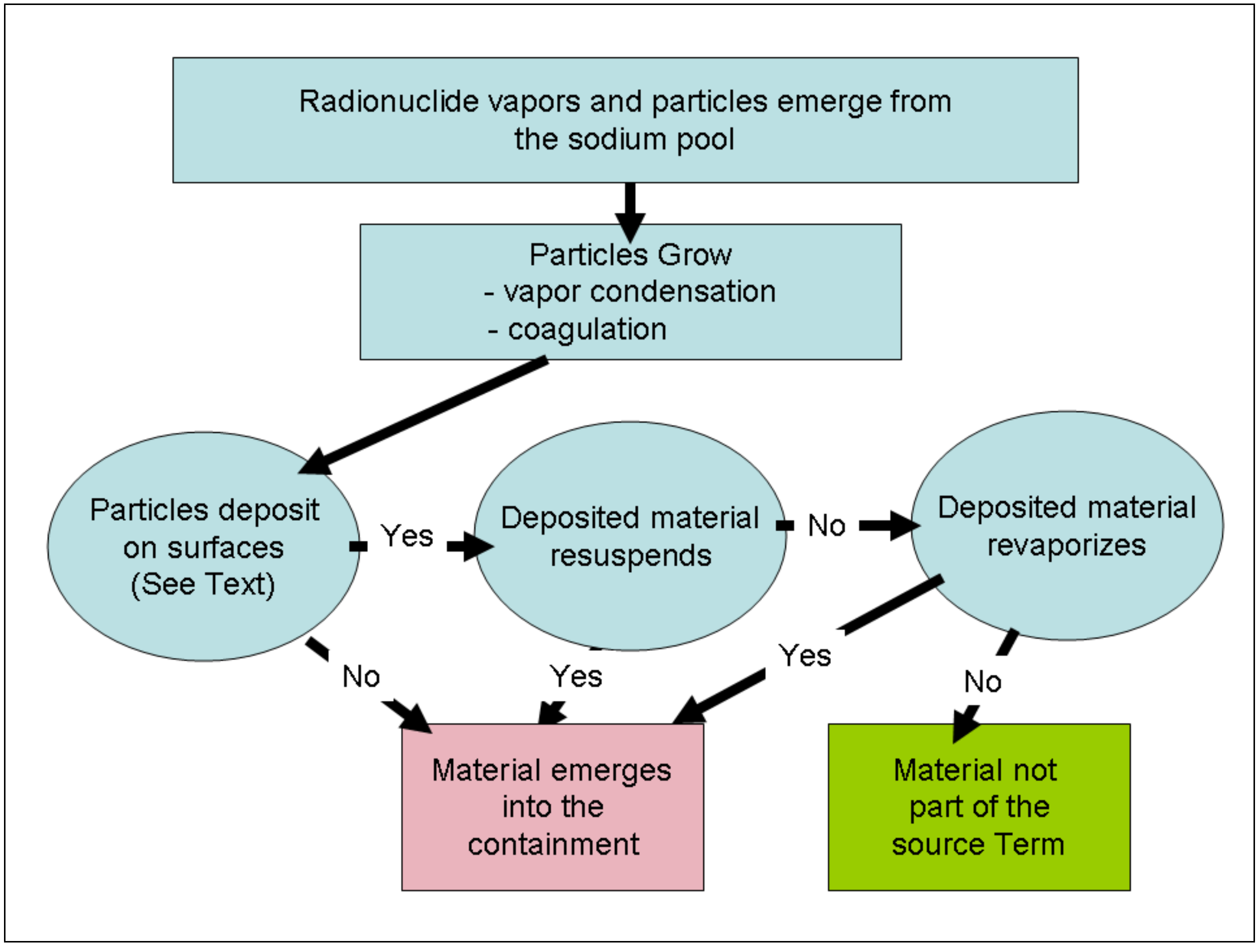

Figure 3. Behavior of particles released from the surface of the sodium pool.

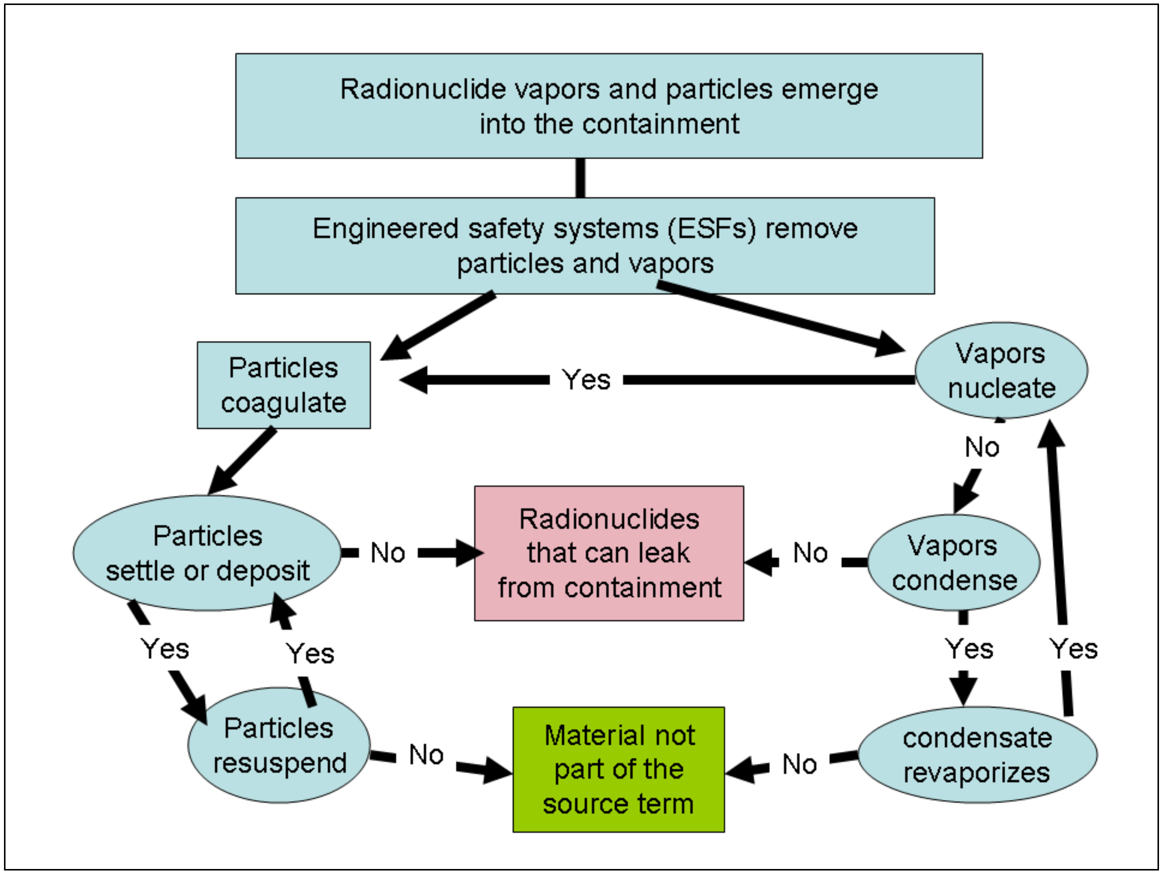

Figure 4. Behavior of radionuclides in the reactor containment.

Mechanical entrainment of sodium contaminated with dissolved and suspended radionuclides can occur when bubbles produced by boiling of the sodium or the release of fission gases burst at the sodium pool surface. Bursting bubbles will produce small droplets (~5 micrometers) of contaminated liquid that can be entrained in the gas flow or settle back onto the sodium pool surface.

Leakage of sufficient air or water vapor into the reactor vessel can saturate the sodium with oxygen and lead to a layer of sodium oxides, hydroxides and carbonates on the surface of the sodium. This layer will affect both the vaporization and mechanical entrainment of radionuclides from the sodium pool.

6. Transport in the Reactor Coolant System

Vapors and particles released from the surface of the sodium pool will enter an environment that is cooler than the sodium pool. To contribute to the radionuclide release to the environment, these particles and vapors must reach the containment if there is no release pathway to the environment that bypasses containment. Substantial physical transformations of both vapors and particles take place that can affect the ability of radionuclides to negotiate passage through the reactor coolant system to the containment (See Figures 2 and 3).

Vapors evolved from the sodium surface will be able to condense. They may condense onto structural surfaces or they may condense onto surfaces of particles. The driving force for vapor condensation onto surfaces of structures or particles can be enhanced if there is a chemical reaction between the vapor and the surfaces.

Vapors concentrations and cooling rates may be high enough that particles can nucleate from the vapors. Particles that initially form from the vapors can be quite small (< 0.1 micrometers). Nucleated particles will agglomerate with particles released from the sodium pool. The particles agglomerates will have a distribution of sizes with means perhaps in excess of one micrometer.

During transport through the reactor coolant system, the particles can deposit on structural surfaces by a variety of mechanisms including:

- Gravitational settling
- Inertial impaction
- Thermophoresis
- Diffusion

Deposition can be enhanced by diffusiophoresis if there is a great deal of sodium vapor condensing onto cool surfaces.

Deposition of particles and vapors on structural surfaces within the reactor coolant system will prevent these radioactive materials from reaching the containment. The extent of deposition in the reactor coolant system depends on the type of plant and the location of any rupture in the primary coolant system. A failure in the reactor head region would create a direct path to containment with little opportunity for radionuclide deposition on structures. Ruptures at more distant locations would provide opportunities for substantial deposition and consequent attenuation of the prompt release to the containment.

The retention of radionuclides deposited in the reactor coolant system may not be permanent. The surface deposits will be heated by a combination of convective heat transfer by gases emerging from the sodium pool region and by the decay heat of radionuclides in the deposits. Temperatures sufficient for revaporization of the deposited materials may be reached. Revaporized materials could flow from the reactor coolant system to the containment.

The deposited materials may also be mechanically resuspended. Resuspension can occur when there are sudden increases in the gas flow velocity or when there are shocks and vibrations of the substrates for the deposits. Resuspension by changes in flow velocity have been researched [Allelein *et al*., 2009]. Resuspension by shock or vibration and its synergism with changes in flow have not received research attention sufficient to make predictive models.

7. Radionuclide Behavior in the Containment

Some fraction of the fission gases, radionuclide vapors, sodium and radioactive particles released from the sodium pool will eventually emerge into the reactor containment (See Figure 4). This fraction will depend very much on the nature of the flow path from the reactor coolant system to the containment. If no flow path other than normal leakage paths develop, very little radioactivity will reach the containment. A rupture of the reactor coolant system will allow a much greater fraction of the radionuclides released from the fuel to reach containment.

The containment atmosphere will contain air and some partial pressure of water vapor. Both air and water vapor are quite reactive toward sodium vapors and aerosol droplets. With a large flux of sodium vapor from the coolant pool to the containment, very high number densities of aerosol particles will form in the containment atmosphere. These aerosol particles will agglomerate rapidly forming very large particles (> 20 micrometers). Fission product vapors can deposit on the surfaces of these particles and fission product particles can coagulate with these particles. The large particles can sediment to the containment floors quite rapidly.

Exothermic reactions of air and water vapor can produce temperatures high enough to cause some thermal decomposition of radionuclide species. A particular example of interest is the decomposition of sodium iodide to form molecular iodine:

$$NaI + H_2O \rightarrow NaOH + HI$$

$$2\,HI \rightarrow I_2 + H_2$$

Molecular iodine produced in this way can react with organic vapors produced in the containment atmosphere by the pyrolysis and radiolysis of cable insulation and other organic materials. The product of reaction can be volatile organic iodides that, like molecular iodine, can persist in the containment atmosphere. In the radiation environment, radiolysis products can convert these iodine vapors into particles of iodine oxides that initially will be quite small (<0.02 micrometers).

Particulate matter suspended in the containment atmosphere will continue to coagulate by a combination of Brownian diffusion and gravitational collision. If the atmosphere is not dried by reaction with sodium vapor, hygroscopic growth of aerosol particles will also take place. The availability of water for hygroscopic growth of particles may be limited, if for no other reason, by the reaction of water vapor with the sodium vapor from the coolant pool.

As the particles grow, they will deposit predominantly by the mechanism of gravitational settling. Initially deposited particles can be resuspended by sufficiently intense changes in gas flow such as during hydrogen deflagration events or containment depressurization. But, with time, deposited materials will absorb water and become more firmly adhered to deposition surfaces.

8. Sodium Fires and Interactions with Concrete

Contaminated liquid sodium can leak from the reactor coolant system and burn in the containment atmosphere by reaction with either air or water vapor. Fires may be either spray fires of sodium droplets reacting with the atmosphere or pool fires of accumulated sodium reacting at the exposed surfaces with the atmosphere. In extreme cases, liners exposed to sodium leaked from the reactor can rupture and sodium can react with the underlying concrete. Associated with both sodium fires and sodium interactions with concrete will be some release of radionuclides dissolved or suspended in the sodium.

The experts agreed that basic phenomena that lead to radionuclide release from sodium in fires and in interactions with concrete are the same as those discussed above especially in connection with a quiescent sodium pool. The phenomena are enhanced by different mass transport conditions and in the case of sodium-concrete interactions by gas sparging by hydrogen and carbon monoxide or dioxide produced by the thermal decomposition of concrete and reaction with sodium. If there is an adequate understanding of release processes during earlier accident phases discussed above, the releases of radionuclides in fires and sodium-concrete interactions can be adequately modeled. Consequently, there was not an opinion elicitation of the phenomena and research needs to model source terms for fires and sodium-concrete interactions.

IV. EXPERT OPINION ELICITATION RESULTS

The expert opinion elicitation was conducted in three parts:

- Identification of phenomena that can affect the source term
- Importance ranking of these phenomena in terms of the figure of merit which was taken to be the radionuclide inventory suspended in the reactor containment atmosphere.
- Ranking of phenomena according to the need for additional research.

No constraints were placed on the identification of phenomena. The phenomena were simply identified and described without regard to importance. There was clearly some prejudice among the experts to concentrate on the more important phenomena and the phenomena identification was not intended to be comprehensive. The experts did try to identify all phenomena each thought to be important. Results of the phenomena identification are listed in Table 4.

A. Phenomena Importance Ranking

The importance rankings of the phenomena identified by the experts as well as the final consensus rankings are also shown in Table 4. In developing the rankings, the experts assumed that much of the information on fuel behavior during accidents would come from models other than the mechanistic source term model. This included models of the melting and expansion of fuel, cladding rupture and the candling of fuel. Similarly, the experts assumed that needed information on the thermal hydraulics of the coolant would come from other models. Consequently, the experts did not devote a great deal of attention to the needs for these models crucial to the support of a mechanistic source term model.

For both the fuel thermal excursion stage of an accident and the extended degradation stage, the experts noted that the release of radionuclides from the fuel was of quite high importance. Rather extensive efforts have been undertaken to characterize experimentally the diffusive release of radionuclides from low enrichment uranium dioxide fuel used in light water reactors [Gauntt, 2010]. Similar work is being done or planned for mixed oxide fuel. In general, these results show that there is a strong dependence of radionuclide release from the fuel on temperature and a weaker dependence on burnup once a threshold level of burnup is reached. The burnup effects are further complicated at burnups in excess of about 45 GWd/t. At higher levels of burnup, a so-called “rim” develops on the fuel. Formation of this rim accentuates the releases of radionuclides from fuel under accident conditions. Low temperature releases of volatile radionuclides such as cesium and iodine differ in low enrichment urania fuels and mixed oxide fuels. The applicability of the results to oxide fuels that will be used in an Advanced Burner Reactor is open to question. Results cannot be applied confidently to metal fuels that might be used in a sodium-cooled advanced reactor. Metal fuels become molten at much lower temperatures than oxide fuels and there is, then, a lower driving force for vaporization. Furthermore, uranium forms a low volatility iodide that could suppress the release of iodine.

The experts found, in fact, that they needed to distinguish between the behaviors of oxide and metal fuels during the extended degradation stage of an accident as well as the thermal excursion stage. The first point of departure in the behavior of the radionuclides under accident conditions occurs when clad ruptures and the fuel depressurizes. Both types of fuel will vent fission gases and vapors that have accumulated during reactor operation in the fuel-cladding gap and the fuel rod plenum. It is known that the flow of gas will also entrain fuel particles in the case of oxide fuels. The experts were not certain that this entrainment of metal fuel particles would occur. They did feel that liquid sodium used in the metal fuel rods to thermally “bond” the fuel to the cladding could be entrained in the gas flow during depressurization. The experts felt further that the liquid sodium would be contaminated with volatile radionuclides released from the fuel during normal operations.

The potential for an extended period of fuel degradation is dependent on the accident scenario. Sodium has excellent heat transfer properties. In many scenarios, it can be expected that fuel damage will be arrested promptly following the initial phase of the accident as sodium quenches high temperature core debris. Further radionuclide release from quenched fuel would come by means of sodium leaching or chemical attack.

For accident scenarios that did involve an extended period of degradation such as the loss of long term heat removal, the experts felt that the behavior of melting fuel would be different for oxide fuel and metal fuel. The candling and accumulation of molten oxide fuel would be much like fuel behavior that is now expected in severe accidents in light water reactors. That is, it is thought possible that candling melt could accumulate and freeze on rod spacers or a core support plate to form eventually a molten pool within the core region.

Such behavior would not be expected for molten metal fuel. As molten fuel drained down a fuel rod, it would melt and entrain steel cladding. The heat of dissolution of cladding into molten metal fuel might be sufficient to prevent freezing and the molten core debris would drip into residual sodium, quench and sediment into the plenum of the reactor vessel to form a debris bed. Energetics of the quenching of molten metal fuel droplets dripping into sodium might determine the debris particle size and the ultimate coolability of the debris bed.

Furthermore, some transfer of radionuclides from fuel to sodium might occur during the rapid quenching process.

While it is possible for molten oxide fuel to drain down rods and drip into the residual coolant much as for the metal fuel, the experts felt it more likely that a pool of debris would accumulate in the core region. This accumulation of core debris raises the possibility of recriticality and another energetic excursion in the fuel debris. If the core debris remains subcritical, natural convection would develop in the molten pool. This natural convection would be expected to impart the most heat on the radial boundaries of the debris. These boundaries would eventually fail and there could be a pour of debris into residual coolant. Energetic interaction between the molten core debris and the fuel could assure that the entire pool of core debris participates in this process. The energetics of these molten fuel interactions with coolant would greatly reduce the size of debris particulate that accumulated in the lower plenum of the reactor relative to the debris size that would be expected to exist if melt simply drained out of the core region and dripped into the sodium. Energetic quenching of molten fuel by sodium could prompt some transfer of radionuclides to the sodium.

Table 4. Phenomena that will affect the source term to the containment and leakage from the containment.

Phenomenon **Fuel Temperature Transient and Failure Phase**	Importance Ranking by Experts*						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
High temperature release of radionuclides from fuel		High	High	High	High	**High**	Applies equally to metal and oxide fuel
Sodium vapor bubble growth and the importance of fission products being scrubbed from the bubble		High	High	High	High	**High**	Includes the importance of the fuel equation of state
Final morphology of the fuel debris		Med	Med	Med	High	**Med**	Self leveling of the debris included as well as the distribution of debris within the coolant system

* A blank entry means the expert chose not to offer an opinion on the issue.

Table 4, cont'd.

Phenomenon Extended Degradation Phase	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	Consensus	
Bubble size distribution and breakup		High	Med	High	Med	**Med**	
Bubble swarm rise velocity		High	Med	High	Med	**Med**	Includes the effects of structures
Mass transport within the bubbles and the deposition of radionuclide particles and vapor into the sodium		High	High	High	High	**High**	
Energetic molten fuel – coolant interactions that could fragment and disperse core debris		High	High	High	High	**High**	Mechanisms are different for molten oxidic and metallic fuels. Radionuclide transfer of primary interest for the source term.
Radionuclide transport from the molten pool in the reactor core region		Low	Low	Med	Low	**Low**	Potential for recriticality is probably the greater concern

Table 4, cont'd.

Phenomenon **Oxide fuel failure release of radionuclides**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Accumulation of radionuclides in the fuel-cladding gap and fuel plenum during operations		High	High	High	High	**High**	
Chemical form of radionuclides		High	High	High	High	**High**	
Chemical activities of radionuclides in fuel		High	Med	High		**Med**	
Mass transport limitations between fuel and the sodium vapor bubble		Med	Med	Med	Low	**Med**	
Entrainment of particulate during depressurization of fuel rod with ruptured cladding		Med	Low	Med	Low	**Low**	

Table 4, cont'd.

Phenomenon Metal fuel failure release of radionuclides	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	Consensus	
Accumulation of radionuclides in the fuel-cladding gap and fuel plenum during operations		High	High			**High**	Important to know what radionuclides have accumulated in the sodium bond between clad and fuel.
Chemical form of radionuclides		High	High			**High**	
Chemical activities of radionuclides within fuel		High	High			**High**	
Mass transport limitations between fuel and sodium vapor bubble		High	Med			**Med**	
Entrainment of particulate during depressurization of fuel rod with ruptured cladding		High	High			**High**	Includes entrainment of sodium used for thermal bonding of clad and fuel since the sodium could contain dissolved radionuclides

Table 4, cont'd.

Phenomenon **Radionuclide release from fuel debris into a quiescent sodium pool**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Mass flow of liquid and vapor sodium through the debris	High	High	High	High	High	**High**	Boiling of sodium is less likely for metal fuels.
Chemical attack on the fuel to form sodium 244rinates, plutonates, etc.	Med	Med	High	Med	Med	**Med**	Depends on the oxygen potential of the sodium; in the case of oxide fuel also depends on fuel stoichiometry
Grain extraction from the fuel		Med	Med		High	**Med**	
Fuel dissolution or ablation rates	Med	High	High	High	High	**High**	
Radionuclide leaching rates	High	High	High	High	High	**High**	
Fission gas bubble nucleation	High	Low	Low	Low	Low	**Low**	
Fission gas bubble transport	High	Low	High	High	Med	**High**	Includes bubble growth by collision
Diffusion of radionuclides in liquid sodium	Med	High	High	High	Med	**High**	
Solubility of radionuclides in liquid sodium	High	High	High	High	High	**High**	Especially the effect of dissolved oxygen
Recoil injection of radionuclides into sodium	Low	Low	Low		Low	**Low**	Depends on the surface to volume ratio of the core debris

Table 4, cont'd.

Phenomenon **Radionuclide transport within a sodium pool**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Condensation or interaction of dissolved radionuclides with structures within a sodium pool	High	Med	Med	High	Med	**Med**	Includes the effect of low temperatures at the perimeter of pool and the heterogeneous nucleation of deposits on surfaces
Nucleation and growth of particles within the sodium	High	Med	High	Med	High	**High**	Includes formation of insoluble compounds such as UI_3
Particle size, shape factors, drag, sintering and fractal growth	Med	Low	Low	Low	Low	**Low**	
Particle sedimentation rates in sodium	Med	Low	Low	Low	Low	**Low**	
Particle inertial deposition rates in sodium	Med	Low	Med	Low	Low	**Low**	
Thermophoretic deposition of particles in sodium	Med	Low	Low	Low	Low	**Low**	

Table 4, cont'd.

Phenomenon **Vaporization release from the free surface of a sodium pool**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Chemical activity of dissolved radionuclide	High	High	High	High	High	**High**	
Liquid sodium mass transport rate at surface	Med	High	High	High	High	**High**	
Radionuclide diffusion coefficients in liquid sodium	Med	High	High	High	High	**High**	
Surface enrichment of sodium with radionuclides such as that due to surface tension effects		High	High	High		**High**	
Gas phase velocity over the pool surface	High	High	High	High	High	**High**	Convective mass transport coefficient for vapors at the surface
Presence of a sodium oxide film on the surface	Low	High	Med	Med	Med	**Med**	Limiting effect on both vaporization and physical entrainment
Multicomponent gas phase diffusion across surface boundary layer	High	High	High	Med		**High**	
Sodium and radionuclide gas phase diffusion coefficients	Med	High	Med	High		**Med**	
Chemical potential gradients especially of oxygen in the gas phase (fog-line formation)		Low	High			**Med**	Accentuates gas phase mass transport

Table 4, cont'd.

Phenomenon **Mechanical release of radionuclides from the surface of a sodium pool**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Bubble burst entrainment of contaminated liquid sodium	High	Med	High	High	High	**High**	Limited bubble production unless the sodium is boiling
Presence of oxide film on surface	Low	High	Med	Med	Med	**Med**	
Size distribution of droplets produced by bubble bursting	High	Low	High	Med	Med	**Med**	
Surface enrichment of sodium with radionuclides	Low	High	High	High		**High**	
Quiescent surface release observed in Germany	Med	Med	High	Med	Med	**Med**	

Table 4, cont'd.

Phenomenon **Radionuclide transport from sodium pool through the reactor coolant system**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Particle nucleation in cooler environment	High	Low	Low	Low	Low	**Low**	Very high vapor and particle concentrations driven by behavior of sodium and less influenced by radionuclides
Heterogeneous nucleation of radionuclide particles on sodium particles or droplets	High	Low	Med	Med	Low	**Med**	
Particle growth; Brownian diffusion and fractal growth	High	High	High	Med	Low	**High**	
Vapor transport to and deposition on surfaces in the reactor coolant system	High	Low	High	Med	Low	**Med**	
Vapor condensation on particle surfaces and issue of dissipation of the heat of condensation	High	Low	Med	Low	Low	**Low**	
Chemical reaction of radionuclide vapors with surfaces	Med	Low	Med	Med	Low	**Med**	
Inertial deposition of particles	Med	High	High	High	Low	**High**	Especially at bends and discontinuities in flow

Table 4, cont'd.

Phenomenon **Radionuclide transport from sodium pool through the reactor coolant system**	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Thermophoretic deposition of particles	High	Low	Med	Med	Low	**Med**	
Sodium vapor driven diffusiophoretic deposition of particles	Med	Low	Med	Med	Low	**Med**	
Sedimentation of particles in flow path	Med	Low	Low	Low	Low	**Low**	
Resuspension of deposited particles in reactor coolant system	Low	Low	Low	High	Low	**Low**	Sodium vapor condensate film is likely to inhibit resuspension
Revaporization of deposited radionuclides due to decay heating.		High	High	High	Med	**High**	Modeling is possible, but experiments needed to validate models.
Electrostatic charging of particles	Low	Med	Med		Low	**Low**	

Table 4, concluded.

Phenomenon Radionuclide behavior in containment	Importance Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	Consensus	
Combustion of sodium vapor and mist that encounter air	High	High	High	High	High	**High**	Includes vaporization of radionuclides dissolved or entrained in mist droplets
Thermal decomposition of NaI to produce gaseous iodine, I_2	High	Med	High	High	High	**High**	
Reaction to form volatile organic iodides such as CH_3I		Low	High	High	High	**High**	
Radiolytic decomposition of molecular iodine, I_2, to form I_2O_5 particles	Low	Low	High	High	High	**Med**	
Deposition of gaseous iodine on surfaces in containment	Med	Med	High	High	High	**Med**	
Gaseous tellurides		Low	Med	Med		**Med**	
Aerosol particle growth	Med	High	High	High	High	**High**	
Thermophoretic deposition of aerosol particles	Med	Med	Med	Med	Med	**Med**	
Inertial deposition of aerosol particles	Med	Med	Low	Med	Low	**Med**	
Sedimentation of aerosol particles	Med	High	High	High	High	**High**	
Resuspension of deposited particles	Low	Low	Low	Low	Low	**Low**	Formation of sodium hydroxide liquid will inhibit resuspension

The experts expected that there would be a substantial flux of sodium vapor along fuel rods throughout the extended degradation phase of a hypothetical accident. Much of the radionuclide release from the fuel during this stage of the accident would be in the form of gases and vapors. The flows of mixtures of vaporized coolant and radionuclide vapors would be expected to breakup into bubbles. The rise of the bubbles assures that there is some circulation of gases within the bubbles. This circulation would provide a mass flux of vapors soluble in sodium to the sodium surfaces defining the bubbles. Dissolution of the vapors in sodium could mitigate substantially the amount of radioactive material that eventually reached the top of the sodium pool and vented into the gas space above the coolant pool. It was felt that this mitigation needed to be considered in the source term modeling.

The experts felt that many accidents would not have either an especially energetic fuel temperature excursion phase or a prolonged phase of extended degradation. Many accidents would simply involve fuel heatup to the point of clad rupture and then the accident would be arrested. Fuel would be exposed to the action of the sodium circulating by natural convection if not by forced convection. Even those accidents that involved an energetic excursion phase or an extended degradation phase would evolve into a period where fuel debris in particulate form would be exposed to molten sodium. Release of radioactivity from the fuel into the sodium and eventually into the containment would be at a low level, but this stage of an accident could be prolonged and needed to be recognized in the mechanistic source term modeling.

The most important long term process for extracting radionuclides from fuel debris is leaching. Temperatures are likely to be too low in the debris for diffusion of radionuclides to the surfaces of the debris to be significant. Leaching could be enhanced by chemical attack of the coolant on the debris which is known to occur in the case of oxide fuel. Dissolution and ablation of metal fuel could also enhance the leaching of radionuclides from the debris.

Chemical attack on oxide fuel and dissolution of metal fuel would be affected by the presence of dissolved oxygen in the sodium. The experts felt that there was some possibility that air or water vapor leakage into the reactor coolant system could occur under accident conditions. At small rates of oxidant leakage into the reactor coolant system, the oxidant would react with sodium vapor to form sodium oxide aerosol particles. These particles would settle to the surface of the sodium pool and the oxygen would dissolve into the sodium. Should this continue long enough, the solubility limit of oxygen in the large sodium mass could be exceeded and an oxide layer would form on the top of the sodium pool. In any event, it is important to consider in developing a mechanistic source term model whether leaching and fuel dissolution rates are significantly affected by having a sodium-oxygen mixture present rather than pure sodium.

The prediction of leaching of radionuclides from fuel debris does require an understanding of the mass transport of sodium into and out of the debris. The experts assumed that information on this mass transport would be available from models of accident progression. Phenomena affecting this mass transport and the coolability of the debris beds were not discussed.

The experts noted that in addition to leaching, radionuclides could be released if the flow of coolant extracted grains of fuel debris or suspended fine particles of fuel debris. Extraction of fuel grains or suspension of fine debris particles could provide a mechanism for the release of radionuclide of low volatility that would not be expected to be released by vaporization processes. Consequently, this mechanical mechanism of transferring material into the sodium needs to be considered in the development of a mechanistic source term model.

The experts envisaged a debris bed (or fuel with ruptured clad) and sodium circulating into the debris which would be quite hot relative to the bulk sodium temperature. The sodium might even approach boiling within the porous structure of the debris bed. Radionuclides would be extracted up to the point of saturation of the sodium. Sodium, contaminated with dissolved radionuclides (as well as some suspended particulate), would cool as it emerged from the debris and was entrained in the general forced or natural circulation of coolant within the reactor vessel. Since the solubilities of radionuclides in sodium should decrease with temperature, the sodium might become supersaturated in solutes. These solutes might precipitate onto surfaces within the reactor vessel. The rates of mass transport to the surface would limit the rate of precipitation. Cooling of the contaminated sodium emerging from the debris bed might be very rapid. If solubility limits were greatly exceeded very rapidly, dissolved radionuclides might homogeneously nucleate particles within the sodium. To be sure, dissolved fission gases would nucleate bubbles within the sodium.

The experts recognized two dominant mechanisms of radionuclide release from the sodium into the gas phase of the reactor coolant system. The most important is simply vaporization of dissolved radionuclides at the free surfaces of the sodium which could include both the top surface of the pool and bubbles of fission gas or sodium vapor (when the pool boils). The process of vaporization at a free surface is relatively well understood overall. It involves the calculation of mass transport on both sides of the interface as well as the analysis of partial pressures of solutes in equilibrium with the contaminated sodium. The experts assumed that the bulk flow characteristics of the sodium coolant flow would be available for a model of the source term. Two complexities pertinent to radionuclide release were identified. Sodium has a very high surface tension and solutes may preferentially accumulate at the surface or preferentially avoid the surface in favor of the bulk sodium liquid. The experts felt it important to consider deviations in the surface composition of sodium from that of the bulk sodium in the analysis of radionuclide vaporization. The experts noted that changes in the surface composition could be estimated if the solution could be modeled simply as an ideal solution (solute activities equal to mole fractions) or no more complicated than a “regular” solution in its behavior. Modeling surface tension effects for solutions with greater deviation from ideality such as the sodium – iodine system is much more difficult.

The second complexity identified in the prediction of vaporization from a contaminated sodium pool arises because the sodium is also vaporizing. The flux of sodium from a free surface is likely to vastly exceed that of radionuclides. Consequently, gas phase mass transport at the surface needs to consider the full Stefan-Maxwell equation set and not just binary Fickian diffusion. Mass transport could be further complicated if oxidant (air or water vapor) were leaking into the reactor coolant system gas space. Sodium vapor evolving from the surface would react with this incoming oxidant to form sodium oxide particulate and this would distort the concentration gradients in the boundary layer adjacent to the surface of the molten sodium. This so-called “fog-line” phenomenon is known to greatly accentuate vaporization of manganese from steel melts [Turkdogen *et al.*, 1962].

The second mechanism of radionuclide release at a free surface considered by the experts is mechanical entrainment of contaminated sodium droplets by the bursting of bubbles at the surface. Bubbles could be composed predominantly of fission gases or sodium vapor if the sodium pool were boiling. Bubble bursting is well known to produce very fine droplets in water systems and even in steel melts [Tomaides and Whitby, 1976; Guézennec, *et al.*, 2005]. It can be expected that bubble bursting would similarly produce droplets at liquid sodium surfaces at least until a solid crust of sodium oxide formed on the surface. The sodium droplets, of course, would be contaminated by dissolved or suspended radionuclides. This mechanical mechanism would become progressively more important as boiling of the sodium became more vigorous.

German experiments [Koch *et al*., 1990] have detected uranium dioxide release from sodium pools at temperatures far too low for any vaporization of uranium oxides. Further, the sodium pools used in these tests were neither boiling nor sparged. There was, however, gas flow over the sodium surfaces. The release is attributed to some unidentified mechanical process. It may be associated with the formation of capillary waves on the surface of the sodium [Lamb, 1994]. The experts recognized the work by Koch *et al*.[1990], and felt that it needed to be confirmed by independent experiments. If confirmed it should be considered in developing a mechanistic source term model.

The behaviors of vapors and particles emerging into the gas phase of the reactor coolant system of a sodium reactor were viewed by the experts as rather similar to the behaviors of vapors and particles in the reactor coolant systems of light water reactors under accident conditions. Vapors can condense on structural surfaces and aerosol surfaces, or vapors can nucleate to form particles. Particles grow by vapor condensation and by coagulation. Both vapors and particles can deposit on structural surfaces. The deposition of vapors is enhanced by chemical reaction with the surfaces.

Radionuclides retained on surfaces may not be permanently prevented from reaching containment. Particulate materials can be resuspended into the flowing gas phase either by sudden increases in gas flow or by shocks and vibrations of the underlying substrate. Deposited radionuclide vapors can be revaporized as deposition surfaces are heated convectively or by the continuing decay heat release from the deposited radionuclides.

A significant feature of processes operative during the transport of vapors in the reactor coolant system unique to sodium-cooled reactors is the presence of copious amounts of sodium vapor. This vapor could condense on surfaces during transport and enhance the deposition of particles on surfaces by diffusiophoresis.

Similarly, the experts noted that many of the processes affecting radionuclides in the reactor containment for a sodium-cooled reactor are quite similar to processes affecting radionuclides in light water reactor containments. Except for the fission gases, these processes are predominantly aerosol processes including particle growth by coagulation and particle removal by sedimentation. A feature unique to sodium-cooled reactors is that as vapors and particles emerge from the reactor coolant system, they enter an environment rich in air and water vapor that will promptly and exothermically react with sodium vapors and particles. If sodium is rapidly injected into the containment, as in a spray fire or within the flame region of a pool fire, very high temperatures can be produced by these reactions. At high temperatures, sodium iodide is known to decompose to form molecular iodine. This molecular iodine can further react with organic vapors in the containment atmosphere to produce volatile organic iodides. Radiolytic processes in the containment atmosphere can produce ozone that will react with both molecular iodine and volatile organic iodides to form iodine oxide particles.

B. Ranking Phenomena in Terms of the Need for Further Research

Once phenomena were identified and ranked according to importance relative to the figure of merit adopted in this work, the experts then ranked the phenomena in terms of the need for further research. Research the experts were addressing was predominantly experimental research. Some phenomena were in need of further modeling research. In the summary of the expert rankings of the phenomena in need of further research (Table 5), a notation is made in the "comments" column if the additional research in mind is just in the modeling.

For the fuel temperature excursion phase and for the extended degradation phase of accidents, the experts felt that the greatest need was for information on the high temperature release of radionuclides from fuel. Data that experts had in mind are similar to data collected in the VERCORS and ORNL experiments [Gauntt, 2010] for light water reactor fuel. The need is especially acute for radionuclide release from metal alloy fuels. There is also a need for experiments to determine the entrainment of liquid sodium that thermally “bonds” the fuel and cladding during the depressurization of fuel rods with metal fuel.

The experts felt that the scrubbing of radionuclides released in a vapor bubble from the fuel was important, but at least approximate models could be devised using analogies to data and models available for aqueous systems [Clift, Grace, and Weber, 1978]. Such approximations were thought adequate for single bubbles. The data base for rise velocities of swarms of bubbles is less developed for all liquids and not at all well developed for molten metals.

The experts recognized that the morphology of debris and especially its surface area and porosity would be important. For debris that simply quenched in sodium and did not undergo energetic interactions, the experts felt that approximations consistent with the feasible level of accuracy could be made with existing data. Of more concern was the formation of debris by energetic interactions between molten fuel and coolant. Available information suggests that energetic fuel-coolant interactions would lead to very fine debris that could be readily leached of radionuclides. There might also be a transfer of radionuclides from the fuel to the coolant during the interactions. Such transfer has not been examined in detail.

The experts felt the most important research need in the quiescent pool phase of the accident was for data on radionuclide leaching rates with both pure sodium and with sodium contaminated with some dissolved oxygen. There is an important need to understand the thermalhydraulics of sodium flow into debris beds and fuel with ruptured cladding, but the experts felt this information should come from models of accident progression.

With respect to the transport of radionuclides in sodium within the reactor vessel, the experts felt that most aspects of the problem could be modeled in at least a first order way with existing technology. The one exception is the possible enrichment or depletion of sodium surfaces of dissolved radionuclides. Such enrichment or depletion could affect radionuclide vaporization models and mechanical entrainment model predictions and is difficult to predict for solutions with large deviations from ideality such as sodium-iodine solutions.

As might be expected, experts were relatively comfortable with the state of technology concerning the transport of radionuclides within the reactor coolant system. They were especially confident in the ability to model aerosol phenomena when the characteristics of the aerosol particles are known. Shape factors and charged aerosol effects are areas of considerable uncertainty, however. The experts did not identify any high research needs in this area. Depending on the location of a breach in the reactor coolant system, there could be little retention of radionuclides in the pathway from the cover gas above the sodium pool to the containment.

The understanding of aerosol phenomena in the reactor containment was also thought to be quite mature. There was confidence in the data available to support modeling of combustion of vapor and aerosol mixtures that emerged into the containment and subsequent growth and deposition of aerosol particles. There was less confidence in the understanding of chemical process that might be associated with localized high temperature combustion events. Needs for research were noted concerning the potential for sodium iodide decomposition to decompose to form molecular iodine and the subsequent behavior of this iodine including reactions to form volatile organic iodides and the decomposition to form iodine oxide particles.

The experts did consider whether there were radionuclides other than iodine that could be chemically affected by sodium combustion. Of those radionuclides that are released extensively from the fuel, only tellurium and the formation of gaseous hydrogen telluride was thought to be susceptible to such chemical transformations from a particle to a gas in the reactor containment.

Table 5. Assessment of research needs.

Phenomenon **Fuel Temperature Transient and Failure Phase**	Research Need Ranking by Experts*						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
High temperature release of radionuclides from fuel Importance rank: **High**		High	Med	High	Med	**High**	Includes the uncertainty in the fuel equation of state
Sodium vapor/noble gas bubble growth, dynamics and the importance of fission products being scrubbed from the bubble Importance rank: **High**		Med	Low	Med	Low	**Med**	
Final morphology of the fuel debris Importance rank: **Medium**		Low	Med	Med	Med	**Med**	Prototypic experiments are required to characterize the morphology.

* A blank entry indicates that the expert elected not to offer an opinion on the issue.

Table 5, cont'd.

Phenomenon Extended Degradation Phase	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	Consensus	
Bubble size distribution and breakup. Importance rank: **medium**		High	Med	Med		**Med**	
Bubble swarm rise velocity. Importance rank: **medium**.		High	High	Med		**High**	Includes the effects of structures
Mass transport within the bubbles and the deposition of radionuclide particles and vapor into the sodium. Importance rank: **high**.		Med	Low	Med	Med	**Med**	Substantial data available for water and other fluids.
Energetic molten fuel – coolant interactions that could fragment and disperse core debris. Importance rank: **high**.		High	Med	Med	High	**High**	Some data are available but modeling is not predictive.
Radionuclide transport from the molten pool in the reactor core region. Importance rank: **low**.		Low	Low	Med	Low	**Low**	Potential for recriticality is probably the greater concern

Table 5, cont'd.

Phenomenon Oxide fuel failure release of radionuclides	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	Consensus	
Accumulation of radionuclides in the fuel-cladding gap and fuel plenum during operations. Importance rank: **High**.		Med	Low		Low	**Low**	Experimental data are now sufficient to provide an adequate prediction.
Chemical form of radionuclides Importance rank: **High**		Low	Low	Med	Med	**Low**	
Chemical activities of radionuclides in fuel. Importance rank: **Medium**		Med	Low	High		**Med**	
Mass transport limitations between fuel and the sodium vapor bubble. Importance rank: **Medium**		Med	Med	High		**Med**	Time scale for release from fuel could be very short.
Entrainment of particulate during depressurization of fuel rod with ruptured cladding. Importance rank: **Low**.		Low	Low	High	Low	**Low**	

Table 5, cont'd.

Phenomenon **Metal fuel failure release of radionuclides**	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Accumulation of radionuclides in the fuel-cladding gap and fuel plenum during operations. Importance rank: **High**.		Med	Med			**Med**	Data on noble gas releases are adequate. Important to know what radionuclides have accumulated in the sodium bond between clad and fuel.
Chemical form of radionuclides. Importance rank: **High**		Med	Med			**Med**	Potential for substantial retention of iodine is high.
Chemical activities of radionuclides within fuel. Importance rank: **High**.		Med	Med			**Med**	Uranium is a good solvent. Behavior of fuel eutectic must be examined
Mass transport limitations between fuel and sodium vapor bubble. Importance rank: **Medium**.		Med	High			**Med**	No data are available under prototypic conditions. Analyses would have to use analogies to data for water and steel.
Entrainment of particulate during depressurization of fuel rod with ruptured cladding. Importance rank: **High**.		High	High			**High**	High ranking for research need is focused on entrainment of sodium bond.

Table 5, cont'd.

Phenomenon **Radionuclide release from fuel debris into a quiescent sodium pool**	Research Need Ranking by Experts Expert A	 Expert B	 Expert C	 Expert D	 Expert E	 **Consensus**	Comment
Mass flow of liquid and vapor sodium through the debris. Importance rank: **High**	Low	Low	Low	Low	Low	**Low**	Experts felt this information could now be provided adequately accurately.
Chemical attack on the fuel to form sodium uranates, plutonates, etc. Importance rank: **medium.**	Low	Low	Low	Low	Low	**Low**	Adequate data to predict are now available.
Grain extraction from the fuel. Importance rank: **medium.**		Med	Med		Med	**Med**	If extracted grains are suspended in sodium, non-volatile release possible
Fuel dissolution or ablation rates. Importance rank: **high**.	Low	Med	High		Med	**Med**	Exposes surface for leaching
Radionuclide leaching rates. Importance rank: **high**.	Med	High	High		High	**High**	
Fission gas bubble nucleation. Importance rank: **low**.	Med	Low	Low		Low	**Low**	
Fission gas bubble transport. Importance rank: **High**	Med	Low	Med	Med	Med	**Med**	Includes bubble growth by collision

Table 5, cont'd.

Phenomenon	Research Need Ranking by Experts						Comment
Radionuclide release from fuel debris into a quiescent sodium pool	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Diffusion of radionuclides in liquid sodium. Importance rank: **high**.	Low	Low	Med	Low	Low	**Low**	Experts felt that correlations with existing data adequate.
Solubility of radionuclides in liquid sodium. Importance rank: **high**.	Med	Med	High	Med	Med	**Med**	Substantial uncertainty when sodium contaminated with oxygen.
Recoil of radionuclides into sodium. Importance rank: **Low**.	Med	Low	Low		Low	**Low**	Adequate estimates can now be made.

Table 5, cont'd.

Phenomenon **Radionuclide transport within a sodium pool**	**Research Need Ranking by Experts**						**Comment**
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Condensation or interaction of dissolved radionuclides with structures within a sodium pool. Importance rank: **Medium**.	Med	Med	High	Med	Med	**Med**	Includes the effect of low temperatures at the perimeter of pool and the nucleation of deposits on surfaces
Nucleation and growth of particles within the sodium. Importance rank: **High**.	Med	Low	High		Med	**Med**	Includes formation of insoluble compounds such as UI_3
Particle size, shape factors, drag, sintering and fractal growth. Importance rank: **Low.**	Low	Low	Low	Low	Low	**Low**	Now predictable to adequate accuracy
Particle sedimentation rates in sodium. Importance rank: **Low**.	Low	Low	Low	Low	Low	**Low**	
Particle inertial deposition rates in sodium. Importance rank: **Low**.	Low	Low	Low	Low	Low	**Low**	
Thermophoretic deposition of particles in sodium. Importance rank: **Low**.	Low	Low	Low	Low	Low	**Low**	Talbot correlation well established.

Table 5, cont'd.

Phenomenon **Vaporization release from the free surface of a sodium pool**	**Research Need Ranking by Experts**						**Comment**
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Chemical activity of dissolved radionuclide. Importance rank: **High**.	Med	High	High	Med	Med	**Med**	Crude estimates are now possible.
Liquid sodium mass transport rate at surface. Importance rank: **High**.	Low	Med	Low	Med	Low	**Low**	Some data are available
Radionuclide diffusion coefficients in liquid sodium. Importance rank: **High**.	Low	Low	Med	Low	Low	**Low**	Estimates possible where data are not available.
Surface enrichment of sodium with radionuclides such as that due to surface tension effects. Importance rank: **High**.		Med	High	High		**High**	Complicated to predict for solutions that are sub-regular in behavior
Gas phase velocity over the pool surface. Importance rank: **High**.	Med	High	High	Low	Med	**Med**	Convective mass transport coefficient for vapors at the surface
Presence of a sodium oxide film on the surface. Importance rank: **Medium**.	Low	High	Low	Med	Med	**Med**	Limiting effect on both vaporization and physical entrainment
Multicomponent gas phase diffusion across surface boundary layer. Importance rank: **High**.	Med	High	High	High		**Med**	Viewed as a modeling issue.

Table 5, cont'd.

Phenomenon **Vaporization release from the free surface of a sodium pool**	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Sodium and radionuclide gas phase diffusion coefficients. Importance rank: **Medium**	Low	Med	Med	Med		**Med**	Simple modeling may be adequate for source term. Some sophisticated studies have been done by Japanese investigators. Need for further work can be shown by uncertainty analysis of model.
Chemical potential gradients especially of oxygen in the gas phase ("fog-line" formation). Importance rank: **Medium**		Low	High			**Med**	Accentuates gas phase mass transport

Table 5, cont'd.

Phenomenon **Mechanical release of radionuclides from the surface of a sodium pool**	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Bubble burst entrainment of contaminated liquid sodium. Importance rank: **High**.	Med	Low	Med	Med	Med	**Med**	Correlations for water and other fluids may be adequate. More prototypic data can be obtained if the release mechanism is found significant.
Presence of oxide film on surface. Importance rank: **Medium**.	Low	High	Low	Med	Med	**Med**	Inhibits mechanical release
Size distribution of droplets produced by bubble bursting. Importance rank: **Medium.**	Med	Low	Med	Med	Low	**Med**	Existing data and models can be used initially.
Surface enrichment of sodium with radionuclides. Importance rank: **High**.	Low	Med	High	High		**Med**	Modeling and uncertainty analysis required.
Quiescent surface release observed in Germany. Importance rank: **Medium**.		Med	High	Med		**Med**	Experimental confirmation needed before modeling.

Table 5, cont'd.

Phenomenon **Radionuclide transport from sodium pool through the reactor coolant system**	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Particle nucleation in cooler environment. Importance rank: **Low**.	Med	Low	Low	Low	Low	**Low**	May not need to be modeled. If it is, existing technology adequate.
Heterogeneous nucleation of radionuclide particles on sodium particles or droplets. Importance rank: **Medium**.	Med	Low	Low	Med	Low	**Low**	
Particle growth; Brownian diffusion and fractal growth. Importance rank: **High**.	Low	Low	Low	Low	Low	**Low**	Though important, the processes are well understood for the purposes of safety analysis
Vapor transport to and deposition on surfaces in the reactor coolant system. Importance rank: **Medium**.	Med	Low	Low	Med	Low	**Low**	Existing technology adequate.
Vapor condensation on particle surfaces and issue of dissipation of the heat of condensation. Importance rank: **Low.**	Med	Low	Low	Low	Low	**Low**	Process should be well modeled and uncertainty analysis will show if more data are needed

Table 5, cont'd.

Phenomenon **Radionuclide transport from sodium pool through the reactor coolant system**	Research Need Ranking by Experts Expert A	 Expert B	 Expert C	 Expert D	 Expert E	 **Consensus**	Comment
Revaporization of deposited radionuclides due to decay heating. Importance rank: **High**.		Low	Med	Med	Low	**Med**	Modeling is possible, but experiments needed to validate models.
Electrostatic charging of particles. Importance rank: **Low**.		Med	Low		Low	**Low**	
Chemical reaction of radionuclide vapors with surfaces. Importance rank: **Medium**.	Low	Low	Med	Med	Low	**Low**	
Inertial deposition of particles. Importance rank: **High**.	Low	Low	Low	Med	Low	**Low**	Especially at bends and discontinuities in flow.

Table 5, cont'd.

Phenomenon **Radionuclide transport from sodium pool through the reactor coolant system**	**Research Need Ranking by Experts**						**Comment**
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Thermophoretic deposition of particles. Importance rank: **Medium**.	Low	Low	Low	Low	Low	**Low**	General confidence that the Talbot correlation was adequate.
Sodium vapor driven diffusiophoretic deposition of particles. Importance rank: **Medium**.	Low	Low	Low	Med	Low	**Low**	Can be predicted adequately accurately using existing models
Sedimentation of particles in flow path. Importance rank: **Low**.	Low	Low	Low	Low	Low	**Low**	Little sedimentation anticipated in RCS
Resuspension of deposited particles in reactor coolant system. Importance rank: **Medium**.		Med	Low	Med	Low	**Low**	Sodium vapor condensate film is likely to inhibit resuspension

Table 5, cont'd.

Phenomenon **Radionuclide behavior in containment**	**Research Need Ranking by Experts**						**Comment**
	Expert A	Expert B	Expert C	Expert D	Expert E	**Consensus**	
Combustion of sodium vapor and mist that encounter air. Importance rank: **High**.	Low	Low	Low	Med	Low	**Low**	Includes vaporization of radionuclides dissolved or entrained in mist droplets
Thermal decomposition of NaI to produce gaseous iodine, I_2. Importance rank: **High**.	High	High	Med		Med	**High**	
Reaction to form volatile organic iodides such as CH_3I. Importance rank: **High**.			High	Med	High	**High**	
Radiolytic decomposition of molecular iodine, I_2, to form I_2O_5 particles. Importance rank: **Medium**.	Low		Low	Med	Low	**Low**	Well established process in LWR safety research.
Deposition of gaseous iodine on surfaces in containment. Importance rank: **Medium**.	Low	Low	Low	Med	Low	**Low**	Extensive study in light water reactor field.
Gaseous tellurides. Importance rank: **Medium**.			High	Med		**Med**	Modeling investigation needed.
Aerosol particle growth. Importance rank: **High**.	Low	Low	Low	Low	Low	**Low**	Current technology adequate.

Table 5, concluded.

Phenomenon Radionuclide behavior in containment	Research Need Ranking by Experts						Comment
	Expert A	Expert B	Expert C	Expert D	Expert E	Consensus	
Thermophoretic deposition of aerosol particles. Importance rank: **Medium**.	Low	Low	Low	Low	Low	**Low**	Talbot correlation well established.
Inertial deposition of aerosol particles. Importance rank: **Medium**.	Low	Low	Low	Med	Low	**Low**	Crucial features of a design leading to inertial dependence can be identified using existing correlations. The need for further experiments can be shown by uncertainty analysis.
Sedimentation of aerosol particles. Importance rank: **High.**	Low	Low	Low	Low	Low	**Low**	Existing models adequate if particle characteristics known.
Resuspension of deposited particles. Importance rank: **Low**.	Low	Low	Low	Low	Low	**Low**	Formation of sodium hydroxide liquid will inhibit resuspension if there is sufficient water vapor present in the atmosphere. Water vapor may be eliminated by reaction with sodium.

V. SUMMARY OF RESULTS

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. These are listed in Table 6. 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 experts identified 14 such areas and these areas are listed in Table 7.

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 even 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.

Table 6. Phenomena judged to have high importance and high need for research.

- High temperature release of radionuclides from fuel during a temperature excursion event
- Energetic interactions between molten oxide fuel and the sodium coolant
- Entrainment during fuel rod depressurization of radionuclide-contaminated, liquid sodium making up the "sodium bond" between metal fuel and the cladding.
- Fuel morphology and the rates of radionuclide leaching by liquid sodium
- Enrichment of free surfaces of sodium by dissolved or suspended radionuclides
- Thermal decomposition of sodium iodide in the containment to form molecular iodine
- Reaction of iodine species in the containment to form volatile organic iodides

Table 7. Phenomena judged to have high importance and a medium need for research.

- Sodium vapor bubble growth and scrubbing of radionuclides from the bubble during a thermal excursion and fuel failure.
- Mass transport within a rising sodium vapor and noble gas bubble that results in the deposition of radionuclide particles and vapor into liquid sodium
- Accumulation during normal operations of radionuclides in the sodium bond in metal fuel
- Chemical form of radionuclides in the fuel and the fuel-cladding gap
- Chemical activities of radionuclides in the fuel
- Rates of fuel dissolution or ablation in a liquid sodium pool
- Fission bubble transport in the sodium pool
- Solubility of radionuclides in sodium containing various amounts of dissolved oxygen
- Chemical activities of radionuclides dissolved in sodium especially when some dissolved oxygen is present
- Nucleation and growth of radionuclide particles in liquid sodium
- Gas phase velocity over the sodium pool (Thermal hydraulics issue.)
- Multicomponent gas phase diffusion of radionuclides across the boundary layer at the gas-liquid sodium interface
- Entrainment of liquid sodium into the gas phase by the bursting of bubbles at the sodium surface
- Revaporization of radionuclide deposits in the reactor coolant system.

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Appendix A. Biographies of Experts

Bernard Clément

Bernard Clément is a graduate from the French Ecole Centrale de Paris in 1974. He joined the ″Institut de Protection et de Sûreté Nucléaire″, predecessor of the French ″Institut de Radioprotection et de Sûreté Nucléaire″ in 1976.
As a first job, he participated to the start-up of the Phébus reactor devoted to safety experiments. As a test director, he was a member of the team that achieved the first experiments of the Phébus LOCA programme on single rods and 25-rods bundles at Cadarache. He participated in particular in the development and calibration of two-phase flow measurements used for the blowdown phase of the experiments.

He was then responsible for the development and calibration of the in-pile instrumentation for the Phebus, Cabri and Scarabee experiments (Cabri and Scarabee that were used for in-pile safety experiments for sodium-cooled fast breeder reactors). His duties were extended to the realisation of in-pile test sections, a challenging task given the extreme conditions of the experiments both for sodium reactors and for the Phébus CSD programme on LWR severe accidents. He was the co-chair of the Test Section and Instrumentation Working Group of the CABRI 1 programme. At the end of that period, he actively participated to the design of the Phébus FP test section, experimental circuits and instrumentation.

In 1991, B. Clément became experimental Project leader for the Phebus FP programme and then scientific Project leader of the Phebus FP and the International Source Term programmes. He chairs the Scientific Analysis Working Group of both programmes. He is also member of the OECD/CSNI Working Group on Analysis and Management of Accidents and of the Contact Expert Group on Severe Accident Management of the European Commission. This group monitors EC-funded projects realised in countries from the former USSR.

B. Clément is presently Senior Expert at the IRSN.

Richard S. Denning

Education

B.E., Physics, Cornell University (1963)
M.S.E., Nuclear Engineering, University of Florida (1965)
Ph.D., Nuclear Engineering, University of Florida (1967)

Positions

September 2004-August 2006, NRC Advisory Committee on Reactor Safeguards
Committee member
1999-Present, Ohio State University, Nuclear Engineering Program
Adjunct Professor, Program Chair from 1999-2001; April 2006-June 2007.
1987-1991 DOE Advisory Committee on Nuclear Facility Safety
Committee member.
1967-2009 Battelle Memorial Institute
Held technical positions from Principal Research Scientist to Sr. Research Leader.

Qualifications

Dr. Denning has performed research for over 40 years in the fields of risk analysis, nuclear analysis, nuclear safety, and severe accident behavior of nuclear reactors. He has managed studies of the safety and risk of a variety of nuclear facilities including commercial nuclear power plants and a number of DOE's non-reactor nuclear facilities. Beginning in 1973 with the WASH-1400 study and later with NUREG-1150, he has been a primary contributor to the development of Level II methods of Probabilistic Risk Assessment. He assisted the NRC in the development and oversight of its severe accident research program and was a principal contributor to the NRC's source term reassessment study. He was a consultant to the TMI Special Inquiry Group. He was a member of DOE's Advisory Committee on Nuclear Facility Safety 1987-1991. From 1995 to 2007, he had responsibility for the oversight of safety hardware upgrades in DOE's program to improve the safety of former Soviet Union reactors (PNNL was the DOE lead laboratory). He was a member of the NRC's Advisory Committee on Reactor Safeguards from September 2004 to August 2006. He chaired the Nuclear Engineering Program at The Ohio State University on an interim basis from July 1999 to June 2001 and from March 2006 to June 2007.

In a joint program with MIT and ISU, he is performing safety/economic tradeoff studies for sodium cooled fast reactors. Severe accident scenarios are being analyzed using the SAS4A and MELCOR codes augmented by source term models and sodium fire models, to quantify generic Licensing Basis Events. A dynamic approach is being taken to event tree analysis in which the simulation codes are driven by the OSU-developed code ADAPT.

Dana A. Powers

D. A. Powers received his Bachelor of Science degree in chemistry from the California Institute of Technology in 1970. He received a Ph.D. degree in Chemistry, Chemical Engineering and Economics in 1975 from the California Institute of Technology. His research for this degree program included magnetic properties of basic iron compounds, catalyst characterization and the rational pricing of innovative products. In 1974, Powers joined Sandia National Laboratories where he worked in the Chemical Metallurgy Division. His principal research interests were in high temperature and aggressive chemical processes. In 1981, he became the supervisor of the Reactor Safety Research Division and conducted analytic and experimental studies of severe reactor accident phenomena in fast reactor and light water reactors. These studies included examinations of core debris interactions with concrete, sodium interactions with structural materials, fission product chemistry under reactor accident conditions, aerosol physics, and high temperature melt interactions with coolants. In 1991, Powers became the acting Manager of the Nuclear Safety Department at Sandia that was involved in the study of fission reactor accident risks and the development of plasma-facing components for fusion reactors. Powers has also worked on the Systems Engineering for recovery and processing of defense nuclear wastes and has developed computer models for predicting worker risks in Department of Energy nuclear facilities. Dr. Powers was promoted to Senior Scientist at Sandia in 1997. Dr. Powers is the author of 103 technical publications. He is a Fellow of the American Nuclear Society.

From 1988 to 1991, Dr. Powers served as a member of the Department of Energy=s Advisory Committee on Nuclear Facility Safety (ACNFS). In 1994, he was appointed to the Advisory Committee on Reactor Safeguards (ACRS) for the U.S. Nuclear Regulatory Commission. He was Vice Chairman of the ACRS in 1997 and 1998. He was elected Chairman in 1999 and 2000. In 2001, Dr. Powers received the Distinguished Service Award from the US Nuclear Regulatory Commission. Dr. Powers has served on committees for the National Research Council involved with the safety of Department of Energy facilities and the nuclear safety of reactors in the former Soviet Union. He has been an instructor for courses on reactor safety and accident management held by the International Atomic Energy Agency in several countries. Dr. Powers was voted the Theos J. (Tommy) Thompson Award for Nuclear Safety in 2007 by the American Nuclear Society “in recognition of outstanding contributions to the field of nuclear reactor safety”.

Shuji Ohno

S. Ohno received his Bachelor of Engineering degree in nuclear engineering from the University of Tokyo in 1987. He joined Power Reactor and Nuclear Fuel Development Corporation (PNC) in 1987 where he has been engaged for about 20 years in the study on sodium leak and fire for Fast Breeder Reactor (FBR) safety. While the PNC changed its name to Japan Nuclear Cycle Development Institute (JNC) in 1998 and while JNC and JAERI were integrated to the Japan Atomic Energy Agency (JAEA) in 2005, he has worked to date in various fields of research in addition to the sodium fire such as:

- evaluation of ex-vessel sodium-related behaviors for FBR level 2 PRA,
- experimental study on equilibrium evaporation of radioactive impurities in high temperature liquid metal lead-bismuth eutectic, and
- numerical simulation study of sodium thermal-hydraulics in FBR reactor vessel with the emphasis on thermal stratification behavior in an upper plenum region.

In 2008, S. Ohno received a Ph.D. degree in Engineering from the University of Tokyo with the above-mentioned experimental research on evaporation of impurities in lead-bismuth eutectic. Further, he has been contributing in several specialists committees, continuously from 1993 to now, which were organized in the Atomic Energy Society of Japan for discussing thermal-hydraulics and radioactive materials behavior in the containment vessel of light water reactors.

Roland ZEYEN

Roland ZEYEN, citizen of the Grand-Duchy of Luxembourg, has a degree of Dipl. Engineer at the Rheinisch-Westfählische Technische Hochschule Aachen (Germany) in 1973. Post graduate thesis work at EC/JRC Ispra on “High temperature fluidized bed technology for the pyrolytic coating of nuclear fuel kernels for the HTR reactors”.

After a short stay at the International Patent Institute (IIB) at The Hague (the Netherlands) he joined the OECD team of the DRAGON High Temperature Gas Cooled Reactor Reactor (UK) team for the development of high temperature coating techniques of UO_2 particles, in particular for the development of an automatic pyrolytic coating system.

Since 1975 R. Zeyen is working within the laboratories of the European Commission’s (EC) Joint Research Centre (JRC) at Ispra (Italy), in the field of nuclear safety and in particular for a light water reactor severe accident programme called SUPER-SARA to be installed in the Ispra ESSOR experimental reactor. During the same time he performed the EOLO in-pile single rod ballooning experiments also in the ESSOR reactor. In 1983 he became responsible for instrumentation and control of the FARO test equipment (FARO is a melting furnace for 100 kg uranium oxide, with release mechanism for interaction studies with sodium and/or water coolants). Design of new instrumentation (Ultrasonic thermometers) and data acquisition: Participation to the in-pile PAHR (Post-accident heat removal) test series at CEA Grenoble. He also became responsible for JRC-Ispra KROTOS project on fuel/coolant interactions between molten UO_2 and water.

Since 1989 he is the JRC representative at the international Phébus FP programme at Cadarache (France) as EC Deputy Programme Manager to this programme, as well as to the follow-up International Source Term Programme (ISTP). He plays a major role in data management and long term experimental data storage within the EC Severe Accident Research Network, SARNET1&2 of the EC framework programmes.

Appendix B. Background Information Available to the Experts

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Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety

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

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

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Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety

Rodney C. Schmidt, Panel Chair
Sandia National Laboratories
Albuquerque, NM 87185 USA

Tanju Sofu, Tom Wei, and Justin Thomas
Argonne National Laboratory
Argonne, IL 60439 USA

Roald Wigeland
Idaho National Laboratory
Idaho Falls, ID 83415 USA

Juan Carbajo
Oak Ridge National Laboratory
Oak Ridge, TN 37831 USA

Hans Ludewig
Brookhaven National Laboratory
Upton, NY 11973 USA

Michael Corradini
University of Wisconsin
Madison, WI 53706 USA

Hae-Yong Jeong
Korea Atomic Energy Research Institute
Daejeon, 305-352 KOREA

Frederic Serre
Centre d'études nucléaires de
Cadarache – CEA
FRANCE

Yoshiharu Tobita and Hiroyuki Ohshima
Japan Atomic Energy Agency
Ibaraki-ken, 311-1393 JAPAN

ABSTRACT

This report summarizes the results of an expert-opinion elicitation activity designed to qualitatively assess the status and capabilities of currently available computer codes and models for accident analysis and reactor safety calculations of advanced sodium fast reactors, and identify important gaps. The twelve-member panel consisted of representatives from five U. S. National Laboratories (SNL, ANL, INL, ORNL, and BNL), the University of Wisconsin, the KAERI, the JAEA, and the CEA. The major portion of this elicitation activity occurred during a two-day meeting held on Aug. 10-11, 2010 at Argonne National Laboratory.

There were two primary objectives of this work:

- Identify computer codes **currently** available for SFR accident analysis and reactor safety calculations.
- Assess the status and capability of current US computer codes to adequately model the required accident scenarios and associated phenomena, and identify important gaps.

During the review, panel members identified over 60 computer codes that are currently available in the international community to perform different aspects of SFR safety analysis for various event scenarios and accident categories. A brief description of each of these codes together with references (when available) is provided.

An adaptation of the Predictive Capability Maturity Model (PCMM) for computational modeling and simulation [1] is described for use in this work. The panel's assessment of the available US codes is presented in the form of nine tables, organized into groups of three for each of three risk categories considered: anticipated operational occurrences (AOOs), design basis accidents (DBA), and beyond design basis accidents (BDBA). A set of summary conclusions are drawn from the results obtained. At the highest level, the panel judged that current US code capabilities are adequate for licensing given reasonable margins, but expressed concern that US code development activities had stagnated and that the experienced user-base and the experimental validation base was decaying away quickly.

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This work was overseen and managed by Jeffrey L. LaChance (Sandia National Labs), who provided guidance on the approach taken, attended the expert elicitation panel meeting, and provided useful input during the report preparation.

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1.0 BACKGROUND AND INTRODUCTION

1.1 The "Gaps Analysis" Project

The U.S. DOE is currently evaluating advanced Sodium-cooled Fast-Reactor (SFR) designs to provide the capability to transmute actinides and enhance the long-term fissile fuel-supply for fission reactors. An essential element in this evaluation concerns the development of the safety case and appropriate licensing approaches. Development of the safety case for an advanced SFR requires the evaluation of the status of the existing technology base — both experimental as well as computer modeling and simulation — in order to identify gaps where additional information is required. To accomplish this task, the DOE is funding this gap-analysis project under the Fuel Cycle Research and Development program.

The SFR gap-analysis work is divided into several topical areas, including

1. Accident Initiators/Sequences
2. Sodium Technology
3. Source Term
4. Computer Codes and Models for Accident Analysis and Reactor Safety Calculations
5. Fuels and Materials

The approach taken involves expert elicitation and incorporates familiar features of a traditional Phenomena Identification and Ranking Table (PIRT) process incorporated to identify SFR safety relevant phenomena, evaluate the knowledge base, and rank potential gaps in the specific areas of SFR safety technologies. The information developed is intended to enhance our ability to evaluate the safety implications of SFR design options, identify the high priority R&D needs to support SFR safety evaluation, and inform the process of fully integrating safety into SFR design activities.

1.2 Assessment of Computer Codes and Models for Accident Analysis and Reactor Safety Calculations

The work described here concerns topical area 4: Computer Codes and Models for Accident Analysis and Reactor Safety Calculations. Of interest here are the computational tools used to determine if, from a safety standpoint, the response of a reactor system is acceptable during all normal, off-normal, and potential reactor accident conditions that must be considered in order for the NRC to license a reactor. A full assessment of these tools requires a tremendous amount of background information, including

1. A knowledge of SFR physics and all associated reactor plant components and systems,
2. The types of normal, off-normal, and reactor accident conditions and scenarios that could potentially occur and that must be analyzed for reactor licensing,
3. An understanding of all important physical processes that may occur during the accident scenarios of interest,
4. The important safety related concerns and safety metrics used to quantify the performance of a reactor during an accident scenario, and

5. A knowledge of, and detailed information about, the capabilities and limitations inherent in the actual computer codes and models which are currently available.

The purpose of assembling a group of experts was to leverage their collective knowledge about these various topics, and use this as a basis for conducting the assessment. The twelve-member panel (hereafter simply called "the panel") consisted of representatives from five U. S. National Laboratories (SNL, ANL, INL, ORNL, and BNL), the University of Wisconsin, the KAERI, the JAEA, and the CEA in France.

It should be recognized that there are inherent limitations in any expert-elicitation-based assessment activity. This type of assessment, by its very nature, has a subjective quality. Instead of relying on a set of uniformly tested well-defined quantitative metrics, this approach relies on the personal knowledge, experience, and judgment of individual panel members. Furthermore, because this particular assessment activity was so broad in scope, none of the panel members can be considered experts in all relevant areas and topics. However, as a whole, the panel members assembled brought a large amount of experience and depth to the table, and the results and insights that have been produced should prove valuable.

It should be noted that this work also benefited from the earlier expert elicitations for topical areas 1–3 completed previously (see Appendix A for highlights from these earlier activities).

1.3 Description of the expert-elicitation process

The expert elicitation panel met together for two days. Prior to the meeting panel members were provided a description of the elicitation objectives and invited to review relevant reports and papers, including draft reports of the earlier expert elicitations for topical areas 1–3 completed previously [2, 3, 4]. In addition, the panel members were asked to consider the following eight questions in preparation for the elicitation activities.

1. What are the safety metrics of importance for an advanced SFR?
2. What accident analysis and reactor safety calculations will be (or are expected to be) required/needed to license a future advanced SFR?
3. What are the metrics that will determine if a particular computer code or model is acceptable for use in an accident analysis or reactor safety calculation used to support the licensing of a future advanced SFR?
4. What computer codes and associated models are currently available which can perform the accident analysis and reactor safety calculations specified above in the answer to question 2?
5. Are there any accident analysis and reactor safety calculations identified in 2 for which no potentially acceptable computer codes or models are currently available?
6. To what degree do the computer codes and associated models identified in 4 meet the criteria for acceptability described in 3?

7. Based on 5 and 6, what gaps and or weaknesses exist in currently available computer codes and models that would be required/needed to license a future advanced SFR?
8. Are there any other areas of concern or weakness not discussed in 7 relating to currently available computer codes and models that are worthy of note.

During the meeting, the elicitation process had effectively three parts. The first part consisted of introductions, a review of meeting objectives, an initial discussion of the above-mentioned guiding questions, and the refinement of how, as a group, we might best accomplish the panel objectives. This led naturally into the final two parts. Part two involved the active discussion and review of a representative set of generic safety related event scenarios for three types of accident categories: anticipated operational occurrences (AOOs), design basis accidents (DBA), and beyond design basis accidents (BDBA). An important objective of this part was the identification of currently available computer codes that might be used to perform the safety analysis required to assess the consequences of these different event scenarios. The last part of the elicitation process concerned the assessment of computer codes, and involved a significant discussion of the different aspects of a computer code assessment that are important.

The remainder of this report is organized as follows. Section 2 reviews the safety-relevant events and potential accidents that can be envisioned as hypothetically possible during the operation of a SFR nuclear power plant. Section 3 focuses on the identification of computer codes potentially applicable for use in performing the associated safety analysis for each of the scenario/events identified. Section 4 describes the assessment methodology adopted, and then presents the results of the code assessment in tabular form with discussion. Section 5 summarizes the elicitation effort and lists several key conclusions.

2.0 SAFETY-RELEVANT EVENTS AND ACCIDENT SCENARIOS

A broad spectrum of safety-relevant events and potential accidents can be envisioned as hypothetically possible during the operation of a nuclear power plant. As part of the gaps analysis project, a previous expert-elicitation panel identified general reactor transient and accident sequences that are important for establishing the overall safety characteristics of a sodium fast reactor design [3]. For licensing purposes, these events and accidents are typically classified as belonging to one of three "risk categories" that are characterized by the event likelihood (quantified in terms of event frequency per reactor year) and potential consequences. Table 1, derived from reference [3], provides a brief description of three basic risk categories; Anticipated Operation Occurrences (AOO), Design Basis Accidents (DBA), and Beyond Design Basis Accidents.

Table 1 Risk-based classification of safety-relevant events and accidents

Risk Category	**Frequency** (events per reactor year)	**Current NRC Allowable Consequences**
Anticipated Operational Occurences (AOO)	$F > 10^{-2}$ Note: These are expected during the lifetime of a plant	None; maintain margin to fuel damage
Design Basis Accidents (DBA) Note: typically associated with the failure of one safety-grade system	$10^{-2} > F > 10^{-5}$ Note: These are not expected during the lifetime of a plant, but anticipated in the design probability for the design class.	Minor fuel damage permissible at lower probability (< 10^{-4} per reactor year); allowable individual exposure to public < 25 rem at site boundary
Beyond Design Basis Accidents (BDBA) Note: typically associated with multiple safety-grade system failures	$F < 10^{-5}$ Note: These accidents have very low probability and are not considered as part of the design basis for the plant.	Substantial fuel damage permissible; allowable individual exposure > 25 rem to public at lower probability (<10^{-6} per reactor year)

Reference [3] notes that the frequency and allowable consequences shown in Table 1 reflect the higher safety standards that NRC is expected to require for any new reactor system design.

In addition to risk categories, reactor accidents are usefully described in terms of whether or not the safety systems controlling reactor scram operate properly. "Protected" accidents denote that the reactor system successfully scrams, whereas "unprotected" accidents denote failure to scram and are BDBA based on the scram system failure probabilities. Furthermore, reference [3] identifies the following three general types of upset conditions as the important initiating event categories for an accident;

- Loss or reduction of core cooling,
- Addition (or insertion) of reactivity into the core, and
- Reduction or loss of heat removal capacity from the reactor.

It should be recognized that there can be design specific aspects to accident initiators, sequences, mitigating actions, and the ultimate consequences. In this assessment, only general accident scenarios are considered in the absence of a specific design description.

Finally, “severe accidents”, a special sub-category of the BDBA classification, are of importance. Hypothetical severe accidents are typically defined as any type of accident that leads to substantial core melting. In SFRs, such scenarios include the potential for re-criticalities as core materials relocate from their original locations within the core. As a result, these accidents are also known as hypothetical core disruptive accidents (HCDAs).

3.0 IDENTIFICATION OF CODES

For code identification and assessment purposes, the panel constructed three different sets of event tables based on the three risk categories described above. These tables were derived from similar tables provided in Ref. [3], but have been modified for our purposes. Each table describes a set of generic event-scenarios that, taken together, cover the spectrum of safety-related events or accident scenarios identified as important to that risk category. In these tables some of the key relevant phenomena are also listed, and the names of computer codes that might be used to perform the associated analysis are identified. The computer codes named in these tables are those codes that panel members were aware of, including those developed and used in the international community. A separate complete listing of all of the codes mentioned in Tables 2-4 is provided in Table 5.

As shown in Table 2, each of the four generic event/scenarios associated with anticipated operational occurrences assume the reactor successfully scrams. These "protected" events include two reactivity insertion events (one due to seismic), a loss of core cooling event, and a loss of normal heat sink event. The third column in Table 2 lists the code sets that were identified by the panel as potentially applicable for use in performing the associated safety analysis. Each set typically contains a collection of codes that, in aggregate, could be used to model the physical phenomena and reactor systems for the scenario of interest. However, the methodology that might be used for code interactions (e.g. coupled or non-coupled physics, mode of data transfer, etc.) is not denoted and was not addressed in the panel discussions. Code sets are color coded to reflect the country where those codes are available or used (black denotes USA, red denotes France, green denotes Japan, blue denotes Korea).

Table 3 contains eight distinct DBA type event scenarios. (Note that the table continues for two pages.) The first six are protected accidents that reflect several variations of the reactivity insertion (DBA-1, DBA-2), loss of core cooling (DBA-3, DBA-4), and loss of normal heat sink (DBA-5, DBA-6) accidents. The remaining two event scenarios are sodium leakage accidents. The key distinguishing factor between these two is that DBA-7 is at high pressure and DBA-8 is at low pressure. As in Table 2, the third column in Table 3 lists the code sets that were identified by the panel as potentially applicable for use in performing the associated safety analysis. Once again, each color-coded set typically contains a collection of codes that, in aggregate, could be used to model the physical phenomena and reactor systems for the scenario of interest.

Table 4 lists a collection of ten generic beyond design basis accident event scenarios. (Note that Table 4 extends over three pages.) The first six (BDBA-1 through BDBA-6) correspond directly with the DBA-1 through DBA-6 in Table 3, except that the system fails to scram. BDBA-7 and BDBA-8 are simply more severe forms of DBA-7 and DBA-8. BDBA-9 generically represents any unprotected hypothetical event/scenario that leads to substantial core melting, and would thus be considered a "severe accident." In fast reactors, this type of accident scenario can hypothetically lead to core disruption events that would require modeling a host of associated physical processes. BDBA-10 is a variant of BDBA-9 that has historically only been a PRA question in Japan. This is a "protected" event with a complete loss of heat rejection capability that eventually leads to substantial core melting, and therefore would occur over significantly longer time-scales than BDBA-9 (i.e. because the system scrams).

Table 5 lists each of the computer codes mentioned in column 3 of Tables 2 through 4. A brief description of each code, together with references (up to five if available) is also provided.

Table 2 Generic Anticipated Operational Occurrence (AOO) event/scenarios that computer codes would be used to simulate

Event/scenario Description	Relevant Phenomena	Code(s)
AOO-1: Protected Reactivity Insertion event (e.g. control rod withdrawal or drop) and subsequent system response to SCRAM	**Reactivity Effects Prior to Scram** * reactivity feedback at high power * end-of-life prediction of reactivity feedback * burnup control swing / control rod worth * integrity of fuel with breached cladding * integrity of fuel with load following	$MC^2$2/DIF3D/REBUS-3 + SE2 + SAS4A/SASSYS-1 ERANOS2 + GERMINAL CATHARE-V2.5/TRIO-U[1] Super-COPD + FINAS SSC-K
AOO-2: Protected Reactivity Insertion event due to seismic event and subsequent system response to SCRAM	**Relative motion of core and control rods** **Reactivity Effects Prior to Scram** * reactivity feedback at high power * end-of-life prediction of reactivity feedback * burnup control swing / control rod worth * integrity of fuel with breached cladding * integrity of fuel with load following	ANSYS + $MC^2$2/DIF3D/REBUS-3 + SE2+ SAS4A/SASSYS-1 CAST3M + ERANOS2 + GERMINAL CATHARE-V2.5/TRIO-U Super-COPD + FINAS ANSYS + SSC-K
AOO-3: Protected Loss of Core Cooling due to equipment failure or operator error, and subsequent system response to SCRAM	**Thermal-hydraulics** * single phase transient sodium flow * thermal inertia * pump coast-down profiles * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena **Reactivity Effects Prior to Scram** * mechanical changes in core structure * intact fuel expansion * fuel/coolant/structure temperatures	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 ERANOS2 + CATHARE-V2.5/TRIO-U *or* ERANOS2 + FLICA[2] Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) MARS-LMR
AOO-4: Protected loss of normal heat sink due to equipment failure or operator error, and subsequent system response to SCRAM	**Thermal-hydraulics** * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 ERANOS2 + CATHARE-V2.5/TRIO-U Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) MARS-LMR

[1] Calculation of the system response to the SCRAM
[2] Study limited to the core

Table 3 Generic Design Basis Accident (DBA) event/scenarios that computer codes would be used to simulate

Event/scenario Description	Relevant Phenomena	Code(s)
DBA-1: Protected Reactivity Insertion event (e.g. accident due to rapid withdrawal of control rods) and subsequent system response to SCRAM	**Same as AOO-1 case (see Table 2)** ***plus*** * reactivity effects of gas bubble entrainment	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 ERANOS2 + GERMINAL CATHARE-V2.5/TRIO-U[3] NERGAL + Super-COPD + VIBUL(from CEA) SSC-K
DBA-2: Protected Reactivity Insertion event due to seismic event, and subsequent system response to SCRAM	**Same as AOO-2 case (see Table 2)** ***but*** * larger relative motion of core and control rods	ANSYS + $MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 CAST3M + ERANOS2 + GERMINAL CATHARE-V2.5/TRIO-U Super-COPD + FINAS ANSYS + SSC-K
DBA-3: Protected Loss of Core Cooling due to equipment failure or operator error and subsequent system response to SCRAM	**Same as AOO-3 case (see Table 2)**	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 ERANOS2 + CATHARE-V2.5/TRIO-U GALILEE/ERANOS2 + FLICA Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) MARS-LMR
DBA-4: Protected Loss of local core cooling due to a partial internal flow blockage, and subsequent system response to SCRAM	**Thermal-hydraulics** * Effect of subassembly flow redistribution * single phase transient sodium flow * thermal inertia * pump-coast down pump coast-down profiles * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation	SAS4A/SASSYS-1 FLICA + GERMINAL CATHARE-V2.5/TRIO-U ASFRE MATRA-LMR/FB
DBA-5: Protected Loss of normal heat sink due to power-conversion system tube rupture, and subsequent system response to SCRAM	**Thermal-hydraulics** * sodium-steam chemical reaction * CO_2-sodium chemical reaction * pressure-pulse impacts from chemical reaction * sodium stratification * transition to natural convection core cooling core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena • reaction product formation and deposition	SAS4A/SASSYS-1 + SWAAM-II DEBIDO + EUROPLEXUS + REACNOV + PROPANA + MECTUB + REPSO/CALHYPSO + GVNOV CATHARE-V2.5/TRIO-U Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) SWACS (pressure pulse) SERAPHIM+TACT+RELAP (for the sodium-H2O reaction) MARS-LMR+SPIKE

[3] TRIO-U is used for the gas entrainment calculation

Table 3 Generic Design Basis Accident (DBA) event/scenarios that computer codes would be used to simulate (continued)

Event/scenario Description	Relevant Phenomena	Code(s)
DBA-6: Protected Loss of normal heat sink due to equipment failure other than steam-generator tube rupture, and subsequent system response to SCRAM	**Thermal-hydraulics** * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation * decay heat removal system phenomena	SAS4A/SASSYS-1 CATHARE-V2.5/TRIO-U Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) MARS-LMR
DBA-7: Sodium leakage from the primary or intermediate cooling system **at high pressure** (~1 MPa) into a compartment of the reactor containment.	* Sodium spray dynamics * Sodium-pool fire on an inert substrate * Aerosol dynamics * Sodium-cavity-liner interactions * Sodium-concrete-melt interactions	MELTSPREAD (pool behavior) NACOM (spray phenomena) FEUMIX (spray/jet fire) PULSAR (spray/jet fire) PYROS-1 (pool fire) SORBET (Sodium-concrete) RESSORT(Sodium-concrete) CONTAIN-LMR-J SPHINCS + AQUA-SF BISHOP (chemical reactions) NACOM (spray phenomena) ORIGEN-2/CONTAIN-LMR-K /MACCS
DBA-8: Sodium leakage from the primary or intermediate cooling system **at low pressure** (~0.1 MPa) into a compartment of the reactor containment;	* Sodium jet dynamics * Sodium-pool fire on an inert substrate * Aerosol dynamics * Sodium-cavity-liner interactions * Sodium-concrete-melt interactions	MELTSPREAD (pool behavior) FEUMIX (spray/jet fire) PULSAR (spray/jet fire) PYROS-1 (pool fire) SORBET (Sodium-concrete) RESSORT(Sodium-concrete) CONTAIN-LMR-J SPHINCS + AQUA-SF BISHOP (chemical reactions) ORIGEN-2/CONTAIN-LMR-K /MACCS

Table 4 Generic Beyond Design Basis Accident (BDBA) event/scenarios that computer codes would be used to simulate

Event/scenario Description	Relevant Phenomena	Code(s)
BDBA-1: ATWS unprotected Reactivity Insertion event (e.g. Accident due to rapid withdrawal of control rods), not leading to severe accident case.	**Same as for DBA-1 protected event** ***plus*** **Thermal-hydraulics** * heat removal path/capacity **Reactivity Effects** * reactivity feedback at high power * coolant heating and margin to boiling * core reactivity feedback * core thermal and structural effects **Material Behavior** * fuel cladding structural integrity at elevated temperatures * cooling systems structural integrity at elevated temperatures * containment structure integrity	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 + ANSYS ERANOS2 + GERMINAL + CATHARE/TRIO + CAST3M Super-COPD+FINAS SSC-K
BDBA-2: Unprotected Reactivity Insertion event due to seismic event, not leading to severe accident case.	**Same as DBA-2 case** ***but*** * even larger relative motion of core and control rods	ANSYS + $MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 CAST3M + ERANOS2 + GERMINAL Super-COPD+FINAS ANSYS + SSC-K
BDBA-3: ATWS unprotected loss of Core Cooling due to equipment failure or operator error, not leading to severe accident case.	**Same as for DBA-3 protected event** ***plus*** **Thermal-hydraulics** * margin to boiling at peak temperature * core thermal and structural effects * heat removal path and capacity **Reactivity Effects** * core reactivity feedback > fuel motion in intact fuel pins >core restraint system performance * reactor shutdown mechanism **Material Behavior** * long-term performance of structures at elevated temperatures * fuel cladding integrity at elevated temperatures	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 + ANSYS ERANOS2 + CATHARE-V2.5/TRIO-U + + CAST3M GALILEE/ERANOS2 + FLICA Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) SSC-K
BDBA-4: Unprotected Loss of local core cooling due to a partial internal flow blockage, not leading to severe accident case.	**Thermal-hydraulics** * Effect of subassembly flow redistribution * single phase transient sodium flow * thermal inertia * pump-coast down profiles * sodium stratification * transition to natural convection core cooling * core flow redistribution in transition to natural convection * decay heat generation	SAS4A/SASSYS-1 FLICA + GERMINAL ASFRE(+SPIRAL) MATRA-LMR/FB

Table 4   Generic Beyond Design Basis Accident (BDBA) event/scenarios that computer codes would be used to simulate (continued)

Event/scenario Description	Relevant Phenomena	Code(s)
BDBA-5: Unprotected Loss of normal heat sink due to **power-conversion system tube rupture,** not leading to severe accident case.	**Same as for protected events** ***plus*** **Thermal-hydraulics** * thermal inertia * core thermal and structural effects **Reactivity Effects:** * core reactivity feedback * fuel motion in intact fuel pins (metal fuel) * core restraint system performance * reactor shutdown mechanism **Material behavior** * long-term performance of structures and piping at elevated temperatures * fuel cladding structural integrity at elevated temperatures * containment structure integrity	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 + ANSYS + SWAAM-II ERANOS2 + CATHARE-V2.5/TRIO-U (or SAS4A for metallic fuel[4]) + DEBIDO + EUROPLEXUS + REACNOV + PROPANA + MECTUB + CALHYPSO + GVNOV + CAST3M Super-COPD/AQUA (+ ASFRE/BAMBOO) + FINAS SWACS (pressure pulse) SERAPHIM+TACT+RELAP (for the sodium-H2O reaction) SPIKE (pressure pulse)
BDBA-6: ATWS Unprotected Loss of normal heat sink due to equipment failure **other than steam-generator tube rupture**, not leading to severe accident case.	**Same as for protected events** ***plus*** **Thermal-hydraulics** * thermal inertia, core thermal / structural effects **Reactivity Effects:** * core reactivity feedback fuel motion in intact fuel pins core restraint system performance * reactor shutdown mechanism **Material behavior** * long-term performance of structures at elevated temperatures * fuel cladding structural integrity at elevated temperatures * containment structure integrity	$MC^2$2/DIF3D/REBUS-3 + SAS4A/SASSYS-1 + ANSYS ERANOS2 + CATHARE-V2.5/TRIO-U (or SAS4A for metallic fuel) + CAST3M Super-COPD/AQUA (+ ASFRE/BAMBOO+SPIRAL) + FINAS SSC-K
BDBA-7: Sodium leakage from the primary or intermediate cooling system **at high pressure** (~1 MPa) into a compartment of the reactor containment.	* Sodium spray dynamics * Sodium-pool fire on an inert substrate * Aerosol dynamics * Sodium-cavity-liner interactions * Sodium-concrete-melt interactions	MELTSPREAD (pool behavior) NACOM (spray phenomena) FEUMIX (spray/jet fire) PULSAR (spray/jet fire) PYROS-1 (pool fire) SORBET (Sodium-concrete) RESSORT(Sodium-concrete) CONTAIN-LMR-J SPHINCS + AQUA-SF BISHOP (chemical reactions) NACOM (spray phenomena) ORIGEN-2/CONTAIN-LMR-K /MACCS

[4] For In-pin fuel motion calculation

Table 4 Generic Beyond Design Basis Accident (BDBA) event/scenarios that computer codes would be used to simulate (continued)

Event/scenario Description	Relevant Phenomena	Code(s)
BDBA-8: Sodium leakage from the primary or intermediate cooling system **at low pressure** (~0.1 MPa) into a compartment of the reactor containment.	* Sodium jet dynamics * Sodium-pool fire on an inert substrate * Aerosol dynamics * Sodium-cavity-liner interactions * Sodium-concrete-melt interactions • Plant Dynamics	MELTSPREAD (pool behavior) FEUMIX (spray/jet fire) PULSAR (spray/jet fire) PYROS-1 (pool fire) SORBET (Sodium-concrete) RESSORT(Sodium-concrete) CONTAIN-LMR-J SPHINCS + AQUA-SF BISHOP (chemical reactions) ORIGEN-2/CONTAIN-LMR-K /MACCS
BDBA-9: Severe Accidents – Substantial Core Melting, such as: Severe loss of core cooling event Severe reactivity addition event, Severe loss of heat rejection capability (**but not** including protected complete loss of heat rejection capability, i.e. BDBA-10)	**Essentially the same as other BDBAs:** ***plus*** **Fuel and Core Behavior:** * sodium voiding effects > temporal and spatial incoherence * fuel pin failure * fuel dispersal and coolability * re-criticality > potential for energetic events (oxide) * primary vessel thermal and structural integrity (oxide fuel) * radiation release and transport (oxide fuel)	$MC^2$2/DIF3D/REBUS-3 + MCNP + SAS4A/SASSYS-1 (+ SIMMER-III + CONTAIN-LMR + MACCS for oxide) + ANSYS ERANOS2 + TRIPOLI + SAS4A/SASSYS-1 (+ SIMMER-III/IV + CONTAIN-LMR for oxide) + EUROPLEXUS DIF3D + PERKY + SAS4A/SASSYS-1 (+ SIMMER-III/IV + CONTAIN-LMR-J for oxide) + AUTODYN + Super-COPD MELT-III + VENUS-II + CONTAIN-LMR-K
BDBA-10: Protected complete loss of heat rejection capability leading to a severe accident (substantial core melting). NOTE: This has been a PRA question in Japan.	**Same as for above BDBA-9** ***but*** **accident time-scale is longer**	Super-COPD + APPLOHS + SIMMER-III/IV + CONTAIN-LMR-J for oxide + AUTODYN + FINAS

Table 5 List of currently available computer codes relevant for LMR Safety Analysis

Code Name	Brief Description	Notes	References
ANSYS	Generic reference to a suite of engineering simulation software tools developed and marketed by ANSYS Corp. Of particular note here are the structural mechanics and the explicit dynamics tools/codes.	Commercial software	6
APPLOHS	--	JAEA code	No public refs. found
AQUA	Multi-purpose multi-dimensional single-phase thermal-hydraulic analysis code (FDM with porous media approach)		7
AQUA-SF	Advanced simulation using Quadratic Upstream differencing Algorithm - Sodium Fire version (Sodium fire analysis code with three-dimensional gas thermal hydraulics.)		8
ASFRE	Single-phase subchannel analysis code for wire-wrapped fuel pin bundle of sodium-cooled fast reactor with distributed resistance model and flow blockage models		9
AUTODYN	Explicit analysis tool (ANSYS suite) for modeling the non-linear dynamics of solids, fluids, gas and their interaction.	Commercial software	6
BAMBOO	Analysis code to simulate wire-wrapped fuel pin bundle deformation under bundle-duct-interaction conditions		10
BISHOP	Bi-Phase, Sodium-Hydrogen-Oxygen System Chemical Equilibrium Calculation Program		11
CAST3M	FEM analysis of structures as well as Computational Fluids Dynamics. Developed by the French Atomic Energy Commission (CEA). The goal of CAST3M development was to build a high level instrument able to be used as a valid support for the design, dimensioning and the analysis of structures and components, both in the nuclear field as well as in the more traditional industrial sector.		12, 13
REPSO/CALHYPSO	1-D Code for modeling the evolution and the diffusion of reaction product, and modeling of the hydrogen detection performance according to the leak characteristics	EDF code	14
CATHARE V2.5	CATHARE 2 is a multi-purpose multi-reactor concept system code. CATHARE 2 was originally devoted to best estimate calculations of thermal-hydraulic transients in Water-Cooled Reactors such as PWR, VVER or BWR. New developments extend the code to Sodium-Cooled Reactors. CATHARE 2 can now describe several circuits with various fluids either in single-phase gas or liquid, or in two-fluid conditions possibly with non-condensable gases, which allows simulating any kind of reactor concept and any kind of accidental transient.		15
CONTAIN-LMR/1B-Mod1	Containment analysis of accidents in liquid-metal-cooled nuclear reactors	Original SNL-developed code	16
CONTAIN-LMR-J	Containment analysis of accidents in liquid-metal-cooled nuclear reactors with revisions made at JAEA (original version developed at SNL)	CORCON and VANESSA are included and modified for sodium fast reactors	17
CONTAIN-LMR-K	Containment analysis of accidents in liquid-metal-cooled nuclear reactors with revisions made at KAERI, including a sodium pool fire flame sheet model.	original version developed at SNL	No reference found

Table 5 List of currently available computer codes relevant for LMR Safety Analysis

Code Name	Brief Description	Notes	References
DEBIDO	Calculation of 1-D fast 2-phase transients in a pressurized water tube : Computes water flow rate due to guillotine break of the tube	AREVA code	No public refs. found
DIF3D	DIF3D's nodal option solves the multigroup steady state neutron diffusion and transport equations in two- and three-dimensional hexagonal and cartesian geometries. One-, two- and three-dimensional orthogonal (rectangular and cylindrical) and triangular geometry diffusion theory problems are solved by DIF3D's finite difference option. Both transverse leakage and variational nodal transport options are available in hexagonal and Cartesian geometries. Eigenvalue, adjoint, fixed source and criticality (concentration) search problems are permitted.	Developed at ANL	18 - 21
ERANOS2	Deterministic Transport. The European Reactor ANalysis Optimized calculation System, ERANOS, for reliable neutronic calculations. Includes nuclear data libraries, a cell and lattice code (ECCO), reactor flux solvers (diffusion, Sn transport, nodal variational transport), a burn-up module, various processing modules, tools related to perturbation theory and sensitivity analysis, core follow-up modules (connected in the PROJERIX procedures), a fine burn-up analysis subset MECCYCO (mass balances, activities, decay heat, dose rates).		22-25
EUROPLEXUS	General FE software for the non-linear dynamic analysis of fluid-structure systems subjected to fast transient dynamic loading, such as: • explosions in enclosures; • shocks and impacts of projectiles on structures; • analysis of pipelines in transient mode; • safety evaluations of complex Fluid-Structure systems under accidental situations. Jointly developed since 2000, by the CEA, the Joint Research Centre (EC) and SAMTECH.		26, 27
FEUMIX	Code for modeling spray/jet fire and calculation of consequences in the room; simplified modeling with a combustion model taking into account the Na-Air contact surface; a 2 zones modeling is used in the room: a hot zone and a cold zone; Results of combustion efficiency calculated by PULSAR are used as input in FEUMIX	IRSN code	28
FINAS	Finite element nonlinear structural analysis system		29
FLICA-4	3D 2-phase flow thermal hydraulic code dedicated to flow in nuclear reactors or experimental facilities. The main features of FLICA4 code are: (1) single and two-phase flow 3D calculations for transient and steady regimes; (2) transient and steady-state calculations of the fuel temperature field (1D model); and (3) point kinetics model.		30
GALILEE	Nuclear data processing code		31
GERMINAL V1	GERMINAL is a code for fuel pin thermal and mechanical behaviour, both during steady-state and incidental conditions, up to high burn-up. The main models are fuel evolution, high burn-up models, fuel-cladding heat transfer, and fuel-cladding mechanical interaction. The validation data base is wide - more than 50 exps. from PHENIX, SUPERPHENIX, PFR, CABRI reactors Currently under active development to improve some models and to make Germinal more predictive.		32

Table 5 List of currently available computer codes relevant for LMR Safety Analysis

Code Name	Brief Description	Notes	References
GVNOV	Thermal-hydraulics for transient and steady states for SG with overheating	AREVA code	No public refs. found
MACCS	MELCOR Accident Consequence Code System.		33, 34
MACCS2	MELCOR Accident Consequence Code System Version 2		35
MARS-LMR	System analysis code for general transients. 1-D or 3-D is possible for a large plenum.		36
MATRA-LMR/FB	Subchannel code mainly for the analysis of subchannel blockage		37
MC^22	A code to calculate fast neutron spectra and multigroup cross sections.	Developed at ANL	38 – 40
MCNP	A general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori.	Developed at LANL	41, 42
MECTUB	Code for assessment of the swelling and tube bursting risk, linked with wastage	CEA code	43
MELTSPREAD	A transient, 1-D, finite difference computer code to predict spreading behavior of high temperature melts flowing over concrete and/or steel surfaces submerged in water, or without the effects of water if the surface is initially dry.	Developed at ANL	44, 45
MELT-III	Computer program to investigate the transient behavior of a fast reactor during postulated accident conditions.		46
NACOM	Analysis of large-scale sodium spray fires.	Developed at BNL	47
NERGAL	High-precision numerical simulation method for gas-liquid two-phase flows (interface tracking)		48
ORIGEN-2	A computer code for calculating the build up, decay and processing of radioactive materials. The program has a very flexible input scheme that allows user to calculate the burn-up and the fission products fuel inventory for a given reactor power and history as well as the reactor decay power after the reactor scram.	Developed at ORNL. Version 2.2 released June 2002	49-51
PERKY	The code calculates reactivity worth on the multi-group diffusion perturbation theory in two or three dimensional core model and kinetics parameters such as effective delayed neutron fraction, prompt neutron lifetime.		52
PROPANA	Micro leak and leak evolution modeling: empirical correlations based on CEA and EdF experiments for rupture diameter evolution calculation with A800 material Wastage empirical and parametric modeling, calculation of tube damaging, calculation of the hydrogen detection system answer	CEA-AREVA code	53
PULSAR	Code for modeling spray/jet fire; Bi-dimensional meshed modeling with a combustion model taking into account droplets	IRSN code	No public refs. found
PYROS-1	Code for modeling pool-fire	IRSN code	No public refs. found
REACNOV	Code for calculation of consequences of mass transfer and long term effects on secondary circuit	AREVA code	No public refs. found
REBUS-3	System of codes for the analysis of reactor fuel cycles. Two types of problems 1) the infinite-time, or equilibrium, conditions of a reactor operating under a fixed fuel	Developed at ANL	54, 55

Table 5 List of currently available computer codes relevant for LMR Safety Analysis

Code Name	Brief Description	Notes	References
	management scheme, or, 2) the explicit cycle-by-cycle, or nonequilibrium operation of a reactor under a specified periodic or non-periodic fuel management program.		
RELAP5	Light water reactor transient analysis code	INL code	56, 57
RESSORT	Code for modeling sodium-concrete interaction	IRSN code	58
SAS4A/SASSYS-1	Deterministic analysis of design basis, beyond-design basis, and severe accidents in liquid metal cooled reactors (LMRs). Detailed, mechanistic models of steady-state and transient thermal hydraulic, neutronic, and mechanical phenomena are employed to describe the response of the reactor core (including its coolant, fuel elements, and structural members), the reactor primary and secondary coolant loops, the reactor control and protection systems, and the balance-of-plant to accidents caused by loss of coolant flow, loss of heat rejection, or reactivity insertion. The initiating phase of the accident is modeled, including coolant heating and boiling, fuel cladding failure, and fuel melting and relocation. Analysis is terminated upon loss of subassembly hexcan integrity.	Developed at ANL	5, 59-67
SERAPHIM	Computer program for multidimensional multiphase flow involving sodium-water chemical reaction during heat transfer tube failure accident in a steam generator of sodium cooled fast reactors		68
SE2	SE2-ANL is a modified version of the SUPERENERGY-2 thermal-hydraulic code, which is a multi-assembly, steady-state sub-channel analysis code developed at MIT for application to fast reactor (wire-wrapped and ducted) rod bundles. At Argonne, the code was coupled to heating calculation methods based on the DIF3D code system, and models were added for hot spot analysis, fuel element temperature calculations, and allocation of coolant flow subject to thermal performance criteria.		69, 70
SIMMER-III	A 2-D 8-velocity-field, multi-phase, multi-component, Eulerian fluid dynamics code coupled with space-dependent reactor kinetics. Tailored to core disruptive accidents (CDAs) in LMFRs, but flexible for non-LMFR materials to be modeled.		71 – 75
SIMMER-IV	A 3-D 8-velocity-field, multi-phase, multi-component, Eulerian fluid dynamics code coupled with space-dependent reactor kinetics. Tailored to core disruptive accidents (CDAs) in LMFRs, but flexible for non-LMFR materials to be modeled.		76, 77
SORBET	Code for modeling sodium-concrete interaction	IRSN code	78
SPHINCS	Sodium fire analysis code with zone model in multi-cell system		79
SPIKE	Assessment of pressure wave propagation		80
SPIRAL	Computer program to simulate detailed local flow and temperature fields in a wire-wrapped fuel pin bundle (FEM with RANS models)		81
SSC-K	System code for the analysis of reactivity insertion accidents and ATWS.		82
Super-COPD	Plant dynamics code to simulate rated and transient behaviors of sodium-cooled fast reactors		83, 84
SUPERENERGY-2	A Multi-assembly Steady-State Computer Code for LMFBR Core Thermal-Hydraulic Analysis		85

Table 5 List of currently available computer codes relevant for LMR Safety Analysis

Code Name	Brief Description	Notes	References
SWACS	Large leak sodium-water reaction analysis code (pressure pulse)		86
SWAMM-II	Sodium-Water Reaction Code. SWAMM-II models the dynamics of a sodium/water reaction bubble in the bulk of liquid sodium in the steam generator of a liquid metal reactor.		87, 88
TACT	Computer program to evaluate temperature and stress distributions in a heat transfer tube and westage rate on the tube surface due to sodium-water reaction jet in a steam generator of sodium-cooled fast reactors	Under development at JAEA	No reference at present
TRIO-U	CFD reference code of the CEA which is designed for incompressible, turbulent flows in complex geometries. Boussinesq's approximation is used to account for density effects. The code is especially designed for industrial large eddy simulations (LES) on structured and non-structured grids of several tens of millions of nodes.		89 - 91
Tripoli4	General purpose Monte Carlo-based radiation transport code to simulate neutron and photon behaviour in three-dimensional geometries. The main areas of applications include but are not restricted to: radiation protection and shielding, nuclear criticality safety, fission and fusion reactor design, nuclear instrumentation. In addition, it can simulate electron-photon cascade showers.		92 - 94
VARI3D	A generalized perturbation theory code that allows calculation of the effects on reactivity and reaction rate ratios of alterations in microscopic cross sections and/or material number densities. VARI3D is most frequently used to compute the reactivity coefficient distributions and kinetics parameters employed in reactor dynamics and safety analyses. The flux and adjoint distributions required to compute these quantities are provided by DIF3D.	Developed at ANL	95, 96
VIBUL	Plant dynamics code to evaluate the concentration distribution of the dissolved gas and the free gas bubble in primary coolant system of sodium cooled fast reactor	(Originally developed by CEA)	97
VENUS-II	Hypothetical Core Disruptive Accident (HCDA) energy release calculation.	Developed at ANL	98

4.0 RESULTS AND DISCUSSION OF EXPERT-ELICITATION-BASED ASSESSMENT OF US CODES

Tables 2 through 5 identify a large number of computer codes available within the international community to address reactor safety issues in SFRs. However, only those codes that have been used in the US were considered in the actual code assessment process discussed here. Because international panel members were also familiar with these codes, they participated in and contributed to the discussions that occurred during the process. However, they did not provide any scores for the assessment tables.

4.1 Assessment-Methodology and Scoring

There are several different aspects that were considered as part of the code assessment. These are shown in Figure 1 where three distinct assessment categories are defined, each with subheadings. The contents of this figure formed the basis for the code assessment and guided the scoring that was requested. These assessment categories were generated as part of the panel discussions and are strongly influenced by the Predictive Capability Maturity Model (PCMM) for computational modeling and simulation described in reference [1]. However, the categories and approach finally adopted reflect significant adaptations that are felt to be important in this setting. For example, the fidelity assessment scores are directly associated to an "adequacy" standard that is tied to licensing.

A good understanding of Figure 1 is essential to properly interpret the results of the assessment presented later.

The panel first considered three parts of what we call a computer code's "maturity level."

The first part (denoted ML-1) concerns two key aspects of verification: code verification and solution verification. When you verify a code, you insure that the source code exactly represents the physics and modeling equations as intended. When you verify code solutions, you are verifying that the linear and/or nonlinear solution algorithms do indeed provide a correct solution to the discrete equation sets, and that the numerical order-of-accuracy of the discretized equations is understood and realized by the code.

The second part of code maturity (denoted ML-2) concerns software quality engineering. Here we consider software configuration management practices such as configuration identification, configuration and change control, and configuration status accounting. It also includes procedures for software analysis and testing such as regression testing, black box testing, and glass box testing.

The final part of code maturity (ML-3) concerns the degree of model validation, uncertainty quantification and sensitivity studies. Model validation involves quantification of the accuracy of the computational model results by comparing the system response quantities (SRQs) of interest with experimentally measured data. This includes addressing issues about experimental error, data availability and/or applicability, phenomenological scaling, and so forth. It also includes the degree to which results are sensitive to the real-life uncertainty ranges of things such

- **"Maturity Level" assessment**
  - ML-1: Code and Solution Verification
    - *Source code exactly represents the intended models (no bugs)*
    - *Linear and nonlinear matrix solutions are always accurate (no numerical corruption), converged, and the "order of accuracy" of numerical approximations is understood, documented, and verified.*
  - ML-2: Software Quality Engineering (SQE) level
    - *A key aspect of SQE is software configuration management, which is composed of configuration identification, configuration and change control, and configuration status accounting. It also includes procedures for software analysis and testing such as regression testing, black box testing, and glass box testing. (See SAND2007-5948)*
  - ML-3: Model Validation, U.Q. and sensitivity studies
    - *Quantification of the accuracy of the computational model results by comparing the system response quantities (SRQs) of interest with experimentally measured SRQs. This includes addressing issues about experimental error, data availability and/or applicability, phenomenological scaling, and so forth. It also includes the degree to which results are sensitive to the uncertainty ranges in specified boundary conditions, material properties, model coefficients, and so forth. (See SAND2007-5948).*
      The score denotes the quality of the quantification, <u>not</u> the accuracy of the model itself.
  - 3-level assessment scoring
    - *Low (score = 0)*
    - *Medium (score = 1)*
    - *Hi (score = 2)*

- **"Fidelity Adequacy" assessment.**
  - FA-1: Representation and geometric fidelity
    - *Adequacy, for its intended use, of the spatial dimensionality and level of detail included in the spatial definition of all constituent elements of the system being analyzed. Implies an assessment about impact on system response quantities of interest.*
  - FA-2: Physics and material model fidelity
    - *Adequacy, for its intended use, of the physics modeling fidelity, per se. Models can vary from empirical models that are based on the fitting of experimental data (empirical models) to those that might be called "first-principles physics." Assessment requires some judgment about intended use and impact on system response quantities of interest.*
  - 3-level assessment score
    - *0 inadequate for licensing*
    - *1 adequate for licensing as long as margins are significant*
    - *2 adequate for licensing even if margins are small*

- **Code Support Status (CSS) assessment.**
  - Current status of code support, knowledgeable and experienced user base, etc
  - 3-level assessment score
    - *0 not currently supported, no experienced users*
    - *1 partially supported (e.g. maintenance only), few experienced and knowledgeable users*
    - *2 fully supported, many experienced and knowledgeable users*

Figure 1 Description of Code Assessment Scoring Used as the specified boundary conditions, material properties, and model coefficients. Of note is that the ML-3 score is intended to reflect the quality of the quantification, not the fidelity of the model itself (which is addressed separately).

Based on their knowledge of the codes, their development, and use, panel members were asked to use their personal judgment to rate the maturity level as either Low, Medium, or High for each of the maturity level categories.

The second assessment area is "Fidelity Adequacy." A central point here is that the adequacy of a model in this context is to be judged relative to its intended use, which in this case is considered reactor licensing. This implies an assessment about a models impact on system response quantities of interest to licensing. FA-1, titled "Representational and Geometric Fidelity" focuses on the spatial dimensionality and level of detail included in the spatial definition of all constituent elements of the system being analyzed. FA-2 concerns the physics modeling itself. Here, for example, models can vary from empirical models that are based on the fitting of experimental data (empirical models) to those that might be called "first-principles" based physics models. Once again a three-level assessment scoring system was used, but here they are designated numerically as 0 for "inadequate for licensing", 1 for "adequate for licensing as long as margins are significant," and 2 for "adequate for licensing even if margins are small."

The third and final assessment area is about the current status of code support (denoted CSS). This concerns whether there are knowledgeable and experienced users to run a code, and whether current programs are being funded to maintain, use, and/or develop the code. A score of 0 denotes that there are no experienced users and that the code is not supported in any current programs, 1 indicates partially support (e.g. maintenance only) with few experienced and knowledgeable users, and 2 means the code is fully supported and has many experienced and knowledgeable users.

The panel felt that each of these assessment areas was important and relevant when considering potential gaps in the status and capabilities of currently available computer codes. Figure 2 shows the format of a blank code assessment table that lists each of the different code assessment areas. As shown, separate rows are provided for each of the different problems defined in Tables 2 through 4. Each panel member who provided scores completed one of these tables for each of risk categories described earlier (i.e. AOO, DBA, and BDBA).

Prob. ID	**Maturity Level**			**Fidelity Adequacy**		**Support**
	ML-1: *Code and Solution Verification* Rating (N_{score})	ML-2 *Software Quality Eng.* Rating (N_{score})	ML-3 *Validation with UQ/SS* Rating (N_{score})	FA-1 *Geometric Representation* Score (N_{score})	FA-2 *Physics Modeling* Score (N_{score})	*Code Support Status* Score (N_{score})

Figure 2 Format of the Code Assessment Table

Because assessment questions are posed relative to (1) a specific event scenario, (2) a particular set of computer codes that would be used, and (3) with assumptions about the skill of the user/analyst, panel members were forced to make "broad-brush" subjective judgments. For several reasons this means that some measure of inconsistency is inevitable. First, because the different codes identified in a "code set" may have important differences in their maturity, fidelity, or code support characteristics. And second, because event scenarios themselves involve many different physical phenomena, and different models within a particular code may have different maturity or fidelity characteristics for these phenomena.[**] However, this expert-judgment based context is also of value because the results can be presented in a manageable form that can be more easily processed and understood. The results must simply be interpreted and used with perspective and with these limitations in mind.

In addition to filling out the assessment tables, panel members were invited to answer the following summary question for each of the corresponding risk categories:

> *"In your opinion, what is the weakest aspect (or most significant gap) associated with the current US computer code(s) available for simulating AOO, DBA, and BDBA safety events for a SFR?"*

Reponses to this question are presented in a separate table for each of the risk categories.

4.2 Results

This section presents the assessment results in the form of nine tables, where groups of three tables are associated with each risk category. For each risk category the first table (i.e. Table 6, 9, or 12) summarizes the assessment ratings and scores from the panel members. All results are presented as average values. All numerical averages are arbitrarily shown with three significant figures. Because Maturity Level questions were assessed using the terms Low, Medium, or High, these were first translated to numerical scores (0, 1, 2), averaged, and then reported as follows:

Avg. Score S	Rating
0	L
0.0 < S < 0.5	L+
0.5	L/M
0.5 < S < 1.0	M-
1	M
1.0 < S < 1.5	M+
1.5	M/H
1.5 < S < 2	H-
2	H

[**] Of course the complexities that realities like these bring to the assessment process probably make the organization and conduct of an ideally comprehensive, systematic, and fully consistent assessment activity a practical impossibility.

Since not all panel members felt qualified to provide a meaningful assessment for all categories, the actual number of panel scores (or ratings) is also shown in parenthesis. Note that if all "scoring" panel members provided an assessment, then the number of values used to compute the average (denoted N_{score}) would be eight.

The second table in each set (i.e. Table 7, 10, or 13) provides a compilation of short notes that panel members added for context or clarification. They are identified by a numerical ID valued 1 to 12, with the first eight corresponding to "scoring" panel members.

The third table in each set (i.e. Table 8, 11, or 14) is a compilation of the brief reviewer responses to the question posed about the most significant gap or weakness (limited to US computer codes) in each risk category. These are identified by the same panel-member IDs as explained above so that the responses of individual panel members can be compared among tables.

4.2.1 Assessment Results for AOO events

Tables 6, 7 and 8 present assessment results for the generic AOO events described in Table 2.

In the Maturity Level area, panel members uniformly rated the Verification and SQE categories as high, with the Validation with UQ/SS category (ML-3) somewhat lower. Scenario AOO-2, which concerns seismic events, was the only scenario where some concern is evident by panel members. As indicated in the Table 7 notes and Table 8 comments, this is due to some degree of concern about relevant seismic data.

Table 6 Summary of Assessment Results for US Computer Codes used to simulate AOO events.

Prob. ID (Table 2)	**Maturity Level**			**Fidelity Adequacy**		**Support**
	ML-1: *Code and Solution Verification*	ML-2 *Software Quality Eng.*	ML-3 *Validation with UQ/SS*	FA-1 *Geometric Representation*	FA-2 *Physics Modeling*	*Code Support Status*
	Rating (N_{score})	Rating (N_{score})	Rating (N_{score})	Score (N_{score})	Score (N_{score})	Score (N_{score})
AOO-1	H- (6)	H- (6)	M+ (7)	1.14 (7)	1.86 (7)	1.14 (7)
AOO-2	H- (6)	H- (6)	L/M (6)	1.00 (7)	1.71 (7)	1.14 (7)
AOO-3	H- (6)	H- (6)	M+ (7)	1.14 (7)	1.86 (7)	1.14 (7)
AOO-4	H- (6)	H- (6)	M+ (7)	1.14 (7)	1.86 (7)	1.14 (7)

Table 7 Reviewer notes associated with AOO code assessment

ID	Note or Comment
2	On A00-2: ANSYS is not evaluated. For ANSYS, my assessment would be H H M, 2, 2, 2
5	On Maturity Level for Validation with UQ/SS: Rated medium since specific case may not be validated although phenomena has been validated for similar events
7	On A00-2 – ANSYS evaluation is H H M 2 2 2, SAS4A/SASSYS evaluation is H M M 1 1 1.
8	On A00-2 – Support for CSS rated 2 for ANSYS and 1 for SAS4A/SASSYS. Also note there is no seismic data associated with LMR. On A00-2 and A00-3: Exp. Data from EBR-II and FFTF General: Exp data on small reactors compared to power reactors. More data from prototype tests needed.

Table 8 Reviewer responses to the following question: *"In your opinion, what is the weakest aspect (or most significant gap) associated with the current US computer code(s) available for simulating AOO safety events for a SFR?"*

ID	Response
1	Lack of experienced user/analysts who are supported by an active experimental program. Multi-physics simulation codes of complex phenomena must be used/applied by users who understand both the code (numerics, models, limitations, etc.) and the underlying physics being simulated (insights from exp.s, etc.).
2	Using SASSYS-1/SAS4A as part of a driver for sensitivity analysis to quantify uncertainties will be needed for AOO analysis since they require higher degree of certainty for higher frequency events and therefore need a more rigorous treatment.
3	Highest priority: Transition to natural convection / V&V data for complex reactor geometry
4	WORK FORCE: Preserving knowledge and experimental data bases. PHYSICS: Thermal stratification in hot & cold pool with multi-dimensional effects. Experimental basis for turbulent sodium flow and heat transfer. CODE: Continued development is hindered by aging code structure. New users are hindered by archaic input, leading to modeling errors.
5	Fidelity in A00-1 due to ex-core effects during SCRAM, especially thermal stratification/natural convection
6	Because core geometry is maintained in these transients, the most significant gap in my view is the common cause effects of a seismic event on the reactor systems, specifically oscillatory motion of the structure of the core and reactivity feedback given physics uncertainties
7	Need for better/more data for validation

With respect to code fidelity, the consensus was that the geometric representation, although relatively crude by current computational engineering standards, was adequate for licensing purposes, and that the fidelity of the physics modeling was quite high.

CSS scores uniformly reflect that the US codes are only partially supported, and that the number of experienced and knowledgeable users is an area of some concern.

Overall, the assessment results for the AOO events do not suggest any significant gaps. However, a survey of the responses in Tables 7 and 8 suggest several areas of possible concern. They include some seismic event issues, the modeling of transient natural convection processes in the reactor system, and diminished code support having led to out-dated codes and the loss of knowledgeable and experienced users.

<u>4.2.2 Assessment Results for DBA events</u>

Tables 9, 10 and 11 present assessment results for the generic DBA events described in Table 3.

We begin by noting that only one panel member felt qualified to provide assessment results for the sodium leakage scenarios DBA-7 and DBA-8. Furthermore, even this expert was not able to provide an assessment of ML-1 and ML-2 issues for the two codes of relevance here (MELTSPREAD and NACOM). Although Reference [4] assesses the knowledge-level currently available to address sodium leakage, actual codes were not evaluated. Thus additional efforts may need to be pursued in another setting to obtain a more satisfactory assessment of codes for the sodium leakage scenarios. For this reason the results in the DBA-7 and DBA-8 row are italicized and the text is shown in grey.

Table 9 Summary of Assessment Results for US Computer Codes used to simulate DBA events.

Prob. ID (Table 3)	**Maturity Level**						**Fidelity Adequacy**				**Support**	
	ML-1: *Code and Solution Verification*		ML-2 *Software Quality Eng.*		ML-3 *Validation with UQ/SS*		FA-1 *Geometric Representation*		FA-2 *Physics Modeling*		*Code Support Status*	
	Rating	(N_{score})	Rating	(N_{score})	Rating	(N_{score})	Score	(N_{score})	Score	(N_{score})	Score	(N_{score})
DBA-1	H-	(5)	H-	(6)	M+	(5)	1.29	(7)	1.71	(7)	1.17	(6)
DBA-2	H-	(5)	H-	(6)	M-	(5)	1.00	(7)	1.57	(7)	1.17	(6)
DBA-3	H-	(5)	H-	(6)	M+	(5)	1.14	(7)	1.86	(7)	1.20	(6)
DBA-4	H-	(4)	H-	(6)	L/M	(5)	0.71	(7)	1.50	(6)	1.14	(5)
DBA-5	H-	(5)	H-	(5)	M-	(5)	1.14	(7)	1.29	(7)	1.17	(6)
DBA-6	H-	(5)	H-	(6)	L/M	(5)	1.14	(7)	1.86	(7)	1.17	(6)
DBA-7	-----	*(0)*	-----	*(0)*	*L*	*(1)*	*1.00*	*(1)*	*1.00*	*(1)*	*1.00*	*(1)*
DBA-8	-----	*(0)*	-----	*(0)*	*M*	*(1)*	*1.00*	*(1)*	*1.00*	*(1)*	*1.00*	*(1)*

For all other scenarios panel members uniformly rated the Verification and SQE categories as high in the Maturity Level area, with the "Validation with UQ/SS" category (ML-3) somewhat lower. Specifically, there were four scenarios (2, 4, 5, and 6) where the maturity level of the validation category is rated as below medium. This suggests that Validation with UQ/SS is an area where greater attention should probably be paid.

Concerning code fidelity, the consensus was that the geometric representation, although relatively crude by current computational engineering standards, was adequate for licensing purposes, and that the fidelity of the physics modeling was high. The one exception is DBA-4, where the average geometric representation score was 0.71. Concerning this, panel member 2 suggests the need for an improved subchannel + multi-pin analysis capability, and panel member 8 suggests this scenario may not apply to US SFR designs.

Finally, the CSS scores once again uniformly reflect that the US codes are only partially supported, and that the number of experienced and knowledgeable users is an area of some concern.

Overall, the assessment results for the DBA events do not suggest any major gaps. However, in addition to the areas already mentioned in the AOO assessment (seismic, natural convection, code support), several additional areas of possible concern are noted in Tables 10 and 11. These include the need for improved sub-channel + multi-pin analysis capability, the modeling of sodium-steam/water interactions (see notes about the SWAMM-II code in Table 10), and gas bubble entrainment modeling. Finally, a note from panel members 9 and 10 suggests that, for high-burnup conditions potentially considered in future SFRs, the fuel-pin bundle deformation effects might have to be considered in the safety assessment.

Table 10 Reviewer notes associated with DBA code assessment

ID	Note or Comment
2	DBA-2: See AOO-2 note about ANSYS. No specific tests on reactivity implications of an earthquake, but for a bounding case this event is similar to DBA-1 DBA-4: An improved sub-channel + multi-pin analysis capability (to simulate entire sub assembly) would be beneficial as an additional modeling option under SASSYS-1 DBA-5: Ratings for SWAMM-II code are separate and different from other codes, would be M, L, L, 1, 1, 0 DBA-5 & 6: are identical scenarios other than the complication due to sodium fire in steam generator for DBA-5
3	DBA-1: The gas entrainment event controlled the ratings. DBA-4: The experiments are better than CFD. DBA-5: CO2-sodium controlled the ratings.
4	DBA-5: SWAMM code brings down the scores for DBA-5
5	Everything is very similar to the AOOs, same weaknesses. Effects of sodium–steam/water interaction are much more complex to model, so physics modeling is not as developed; SWAMM-II, BUT this analysis can be outside of the SASSYS/SAS4A context
8	DBA-1: Gas bubble entrainment not credible! EBR-II and FFTF data DBA-2: No seismic data associated with LMRs, ANSYS support better than other codes DBA-4: May not apply to US design, no foreign object – only marginally credible, worst case could lead to local pin failure. Oxide fuel generates “crud” which causes blockage DBA-5: No exp. data for CO2 power conversion, ANSYS support better than other codes. SWAAM essentially not supported, SWAAM needs to be upgraded for CO2 General Comment: Same as AOO case - need prototype data for validation of codes.
9, 10	If Advanced Burner Reactor will aim for high burn-up ratio, then fuel pin bundle deformation effects (e.g. radial expansion, bowing, ovalization due to thermal expansion, swelling, irradiation creep and mechanical interaction) might have to be considered in the safety assessment. In JAEA, coupling use of ASFRE and BAMBOO can simulate such phenomena.

Table 11 Reviewer responses to the following question: *“In your opinion, what is the weakest aspect (or most significant gap) associated with the current US computer code(s) available for simulating DBA safety events for a SFR?”*

ID	Response
1	Lack of experienced user/analysts who are supported by an active experimental program. Multi-physics simulation codes of complex phenomena must be used/applied by users who understand both the code (numerics, models, limitations, etc.) and the underlying physics being simulated (insights from exp.s, etc.).
2	Weakest link: An improved sub-channel + multi-pin analysis capability (to simulate entire sub assembly) would be beneficial as an additional modeling option under SASSYS-1
3	Highest priority: Gas entrainment / V&V data for complex reactor geometry
4	Sub-channel and multi-pin channel heat transfer modeling for flow blockages
5	Same as AOO case. Fidelity in DBA-1 due to ex-core effects during SCRAM, especially thermal stratification/natural convection.
6	The most significant gap for this set of accidents is again focused on areas where the geometry is not well known or directly affected by the accident initiation. This can result in uncertainties in reactivity feedback in the reactor core (seismic events or flow blockages) or in the effect on containment or compartment pressurization from sodium leakage and subsequent combustion and fires.
7	No code for water/sodium reaction and better codes for sub-channel analysis. These specific codes have not been included into the *(system)* codes.

4.2.3 Assessment Results for BDBA events

Tables 12, 13 and 14 present assessment results for the generic BDBA events described in Table 4. Note that the first six entries (BDBA-1 through BDBA-6) in Table 4 correspond directly with the DBA-1 through DBA-6 in Table 3, except that the system fails to scram. Also, BDBA-7 and BDBA-8 are simply more severe forms of DBA-7 and DBA-8. BDBA-9 generically represents any unprotected hypothetical event/scenario that leads to substantial core melting, and would thus be considered a "severe accident." BDBA-10, is a variant of BDBA-9 that has historically been a PRA question in Japan (but not in the U.S.).

Table 12 Summary of Assessment Results for US Computer Codes used to simulate **BDBA** events.

Prob. ID (Table 4)	**Maturity Level**			**Fidelity Adequacy**		**Support**
	ML-1: *Code and Solution Verification* Rating (N_{score})	ML-2 *Software Quality Eng.* Rating (N_{score})	ML-3 *Validation with UQ/SS* Rating (N_{score})	FA-1 *Geometric Representation* Score (N_{score})	FA-2 *Physics Modeling* Score (N_{score})	*Code Support Status* Score (N_{score})
BDBA-1	H- (3)	M+ (3)	H- (3)	1.20 (5)	1.80 (5)	1.25 (4)
BDBA-2	H- (3)	M+ (3)	L/M (3)	1.00 (5)	1.40 (5)	1.25 (4)
BDBA-3	H- (3)	M+ (3)	H- (3)	1.20 (5)	1.80 (5)	1.25 (4)
BDBA-4	M/H (2)	M (2)	M- (3)	1.00 (4)	1.50 (4)	1.33 (3)
BDBA-5	H- (3)	M+ (3)	M- (3)	1.20 (5)	1.60 (5)	1.25 (4)
BDBA-6	H- (3)	M+ (3)	H- (3)	1.20 (5)	1.80 (5)	1.25 (4)
BDBA-7	----- *(0)*	------ *(0)*	*L (1)*	*1.00 (1)*	*1.00 (1)*	*1.00 (1)*
BDBA-8	----- *(0)*	------ *(0)*	*M (1)*	*1.00 (1)*	*1.00 (1)*	*1.00 (1)*
BDBA-9	M+ (3)	M+ (3)	M (3)	0.80 (5)	1.00 (5)	1.00 (4)
BDBA-10	H (1)	H (1)	H (1)	1.00 (2)	1.33 (3)	1.00 (2)

As with the DBA risk category assessment, only one panel member felt qualified to provide assessment results for the sodium leakage scenarios BDBA-7 and BDBA-8, and no assessment is given for ML-1 and ML-2 issues. Additional efforts may need to be pursued in another setting to obtain a more satisfactory assessment for the sodium leakage scenarios.

Beyond Design Basis Accident events are considered extremely unlikely and are the most difficult and challenging scenarios for which to obtain high quality experimental data or to model computationally. Providing general assessment scores are especially difficult here because of these issues and the corresponding lower degree of knowledge about the physical processes. Only three panel members provided Maturity-level assessment results and only five did so for the other two assessment categories. This reflects the fact that relatively few people are familiar with the codes, models, and phenomena for these types of scenarios and conditions.

Compared to Table 9 (for DBA events), the ratings and scores shown in Table 12 are similar although slightly lower. In general, the lowest scores are for BDBA-9, the generic "severe accident" scenario. Fidelity scores were generally 1.0 or higher (with BDBA-9 being the one exception), but must be interpreted in light of the “adequacy” criteria. For extremely unlikely events, the fidelity needed for licensing purposes is felt to be significantly less than for events of higher probability.

Reviewer notes listed in Table 13 include important details that add perspective to the ratings provided and should be read. Note in particular that several reviewers comment on the differences between ceramic and metallic fuels, and that the U.S. program on ceramic fuels ended many years ago. Thus U.S. codes may not treat some of the phenomena that must be considered if the reactor contains oxide fuel.

Table 13 Reviewer notes associated with BDBA code assessment

ID	Note or Comment
1	Because US program on ceramic fuels ended in ~1982, severe accident codes to treat phenomena related to ceramic fuels are not supported in the US. However, Japan and France have tools to consider this. Also note that source terms are essentially bounding estimates.
2	BDBA-2: No specific tests on reactivity implications of an earthquake, but for a bounding assumption, this event is similar to BDBA-1. BDBA-4: An improved sub-channel + multi-pin analysis capability (to simulate entire sub assembly) would be beneficial as an additional modeling option under SASSYS-1 BDBA-5: Ratings for SWAMM-II code are separate & different from other codes, would be M, L, L, 1, 1, 0 BDBA-5 & 6: are identical scenarios in terms of primary system response. The difference is the question of how to deal with a sodium-fire in the steam generator in BDBA-5 BDBA-9: Relevant test data from TREAT. Ratings for SIMMER-III + CONTAIN-LMR path is M,L,L,1,1,0
5	BDBA-2: lack of good data to validate seismic response makes ML-3 assessment difficult Early part of the transient is calculated in detail. For metallic fuel, neutronic shutdown is achieved and subsequent events are governed by fuel/steel melting + relocation under decay heat until a coolable geometry is achieved within the reactor vessel. The latter part of the transient is calculated with an experimental/phenomenological discussion, possibly supplemented with small stand-alone models. Given probability of < 10^{-7} per reactor year or smaller, this is likely adequate. The key is no energetic recriticality. For oxide fuel, the accident progression can be substantially different and may involve energetic recriticalities. If one decides that computing these effects are necessary, the first step is to go to Japan *(because of their technical experience in this area).*
7	Note that SIMMER was started in the US and now is a Japanese/German/French code – changed extensively and renamed SIMMER IV. Inclusion of this code in the US group of codes is not appropriate. Concerning source term: There are codes like ORIGEN-2 that can calculate the total source term inside the fuel/core – but the problem is to predict how much will be released for each specific accident. There are aerosols and sodium coolant that complicate the releases. MELCOR can do this job in LWR – a version for LMRs does not exist.
9, 10	If Advanced Burner Reactor will aim for high burn-up ratio, then fuel pin bundle deformation effects (e.g. radial expansion, bowing, ovalization due to thermal expansion, swelling, irradiation creep and mechanical interaction) might have to be considered in the safety assessment. In JAEA, coupling use of ASFRE and BAMBOO can simulate such phenomena.

Table 14 Reviewer responses to the following question: *"In your opinion, what is the weakest aspect (or most significant gap) associated with the current US computer code(s) available for simulating BDBA safety events for a SFR?"*

ID	Response
1	Lack of experienced user/analysts who are supported by an active experimental program. Multi-physics simulation codes of complex phenomena must be used/applied by users who understand both the code (numerics, models, limitations, etc.) and the underlying physics being simulated (insights from exp.s, etc.).
2	Lack of advanced fuel behavior models to predict the margin to pin failure for fuels with high actinide content
3	Core Passive feedback mechanisms / V&V data
6	The most significant gap for this set of accidents is again focused on areas where the geometry is uncertain or changes with time due to fuel rod failure, blockage or voiding with large reactivity insertions, and directly affected by the accident initiation. This can result in large changes in reactivity feedback in the reactor core (seismic events, flow blockages, voiding). These physics are most apparent in BDBA-2, BDBA-4, BDBA-7, BDBA-8, BDBA-9 (not sure of what is in BDBA-10)
7	The biggest gap is in the codes that predict source term releases from fuel in LMR accidents. These codes are not available.

5.0 SUMMARY AND CONCLUSIONS

A two-day expert-opinion elicitation was conducted to qualitatively assess currently available computer codes and models for accident analysis and reactor safety calculations of advanced sodium fast reactors. The expert panel consisted of twelve members representing five U. S. National Laboratories, the University of Wisconsin, the KAERI, the JAEA, and the CEA.

As context for the assessment, safety related event scenarios for three types of accident categories were reviewed: anticipated operational occurrences (AOOs), design basis accidents (DBA), and beyond design basis accidents (BDBA) (See Table 1). During this review, panel members identified computer codes potentially applicable for use in performing the associated safety analysis for each of the scenario/events. Tables 2 through 5 summarize this activity and list 58 computer codes that are currently available in the international community to perform SFR safety analysis. However, only those codes that have been used in the US were reviewed as part of the subsequent assessment.

As detailed in Figure 1, 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 are presented in the form of nine tables (Tables 6 through 14), 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 US 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.

Details of the assessment results are discussed in Section 4 above. The following is a bulleted list of notable conclusions that can be drawn from the assessment:

- Although current US 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” sub-category than to the other sub-categories. This sub-category 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 V&V 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 due to fuel rod failure, blockage or voiding with large reactivity insertions, and 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 sub-channel 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 US 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, establish a more vigorous code verification and QA plan for code maintenance, configuration management/control, and testing of software through improved SQE practices. In their view modernization of the code system was needed to (1) support updating the memory management scheme to remove various nodalization limits, (2) support parallel applications, and (3) 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 the automated design optimization, uncertainty quantification, and sensitivity analysis schemes. It was suggested that if an 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.

Finally, it must be recognized that the conclusions drawn from this assessment activity are relatively general in nature and reflect the personal knowledge, experience, and judgment of individual panel members. A more extensive and involved process would be required to provide a detailed assessment of the each of the individual codes for each of the applicable accident scenarios and physical phenomena that have been identified here.

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Appendix A. Highlights from Previous Gaps Analysis Expert Elicitations

A.1 Accident Initiators/Sequences [3]

This work “identified general reactor transient and accident sequences that are important for establishing the overall safety characteristics of a particular reactor design.”

Three general categories of accidents were defined

- protected,
- unprotected,
- severe with core melting

together with three general types of upset conditions

- reduction or loss of core cooling,
- addition (or insertion) of reactivity into the core,
- reduction or loss of heat removal capacity from the reactor

Several key tables were prepared which summarized the results.

- Table 1: Event Descriptions and Relevant Phenomena
- Table 2: Classification of Events and Consequences for Reactor Licensing
- Table 4: Evaluation of Phenomena and Their Importance

Computer codes mentioned or referenced in the report included the following:

HOTCHAN, SASSYS-1LMFBR, SAS4A, COMMIX, SSC Rev 2., NATDEMO, FRAS3

A.2 Sodium Technology [4]

This effort “focused on phenomena that would occur after a leak,” where the “location and extent of the sodium leak is provided”

Three general accident areas were defined:

- Sodium leakage from primary or intermediate loops at high-pressure,
- Sodium leakage from primary or intermediate loops at low-pressure,
- Coolant leakage into sodium within the power-cycle heat exchanger,

and a group of seven general phenomena identified:

- 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

A summary of the “key gaps” indentified is found in Table 5.1

Codes mentioned or referenced in the report included the following:

NACOM code, MELTSPREAD-1, ABOVE code, CORCON, STAR-CCM, FLUENT, CONTAIN-LMR

A.3 Source Term [2]

This effort only considered "accidents involving substantial fuel damage to the reactor core."

Focused on "research needed to develop a predictive, mechanistic model of the source term for use in the licensing and risk analysis"

Developed "a hypothetical scenario"…"to serve as a framework for identification of phenomena…"

Identification of Phenomena (Table 4), Research needs (Table 5) and "seven phenomena that are of high importance and had a high need for additional experimental research" (Table 6)

- high temperature release of radionuclides from fuel during energetic event
- Energetic interactions between molten reactor fuel and sodium coolant and associated transfer of radionuclides from fuel to coolant
- Entrainment of fuel and sodium bond material during the depressurization of a fuel rod with breached cladding
- Rates of radionuclide leaching from fuel by liquid sodium
- Surface enrichment of sodium pools by dissolved and suspended radionuclides
- Thermal decomposition of sodium iodide in the containment atmosphere
- Reactions of iodine species in the containment to form volatile organic iodides

Computer codes mentioned or referenced in the report included the following:

Source Term Code Package, MAAP4, MAEROS, CONTAIN LMR, MELCOR TRACER

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