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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, 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.

ACKNOWLEDGEMENTS

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.

4.2.2 Assessment Results for DBA events

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** 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})	**Fidelity Adequacy** FA-1 *Geometric Representation* Score (N_{score})	FA-2 *Physics Modeling* Score (N_{score})	**Support** *Code Support Status* 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*		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})
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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