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# Advanced Sodium Fast Reactor Accident Source Terms: Research Needs

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# Advanced Sodium Fast Reactor Accident Source Terms: Research Needs

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## 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.

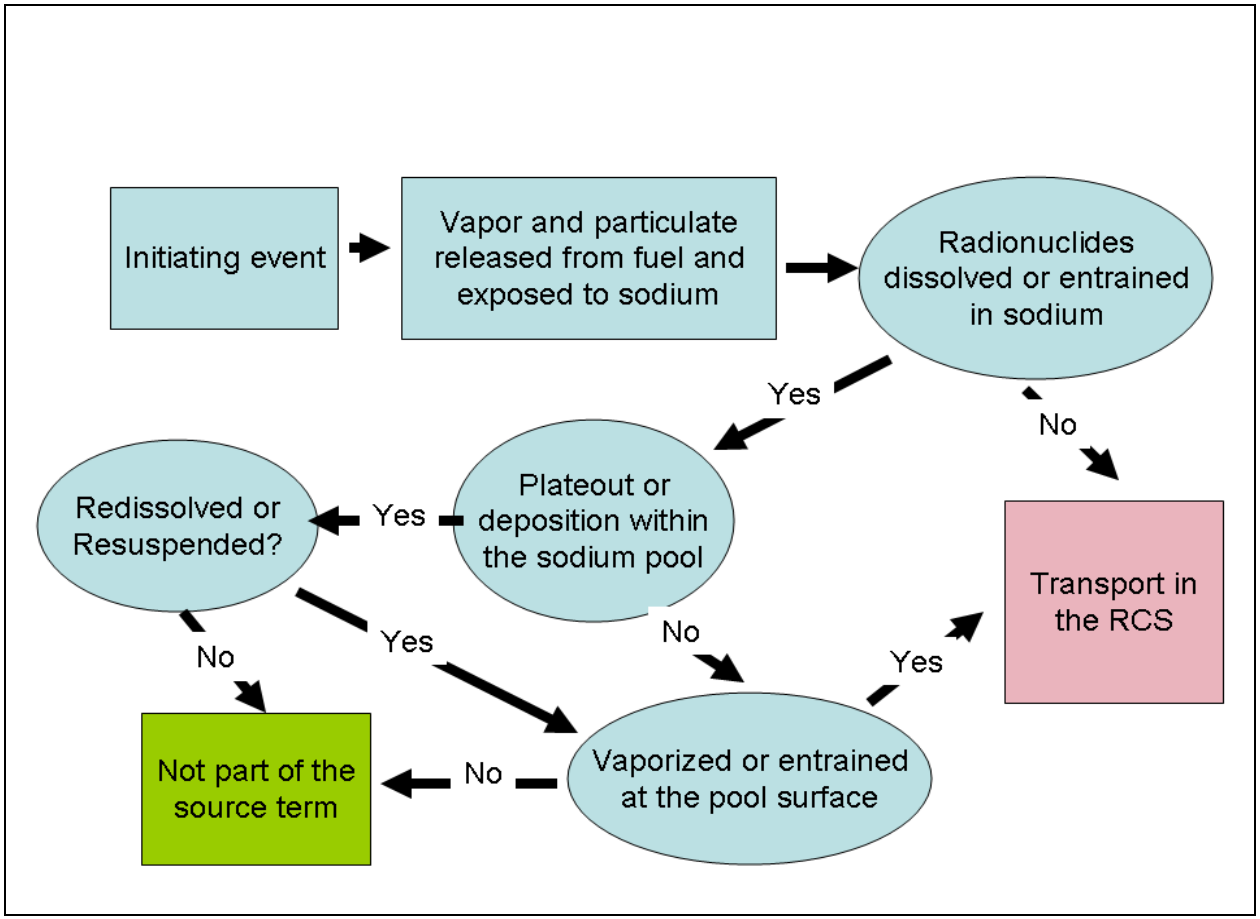


**Figure 1. Schematic representation of the major events of radionuclide behavior following an initiating event that damages fuel.**

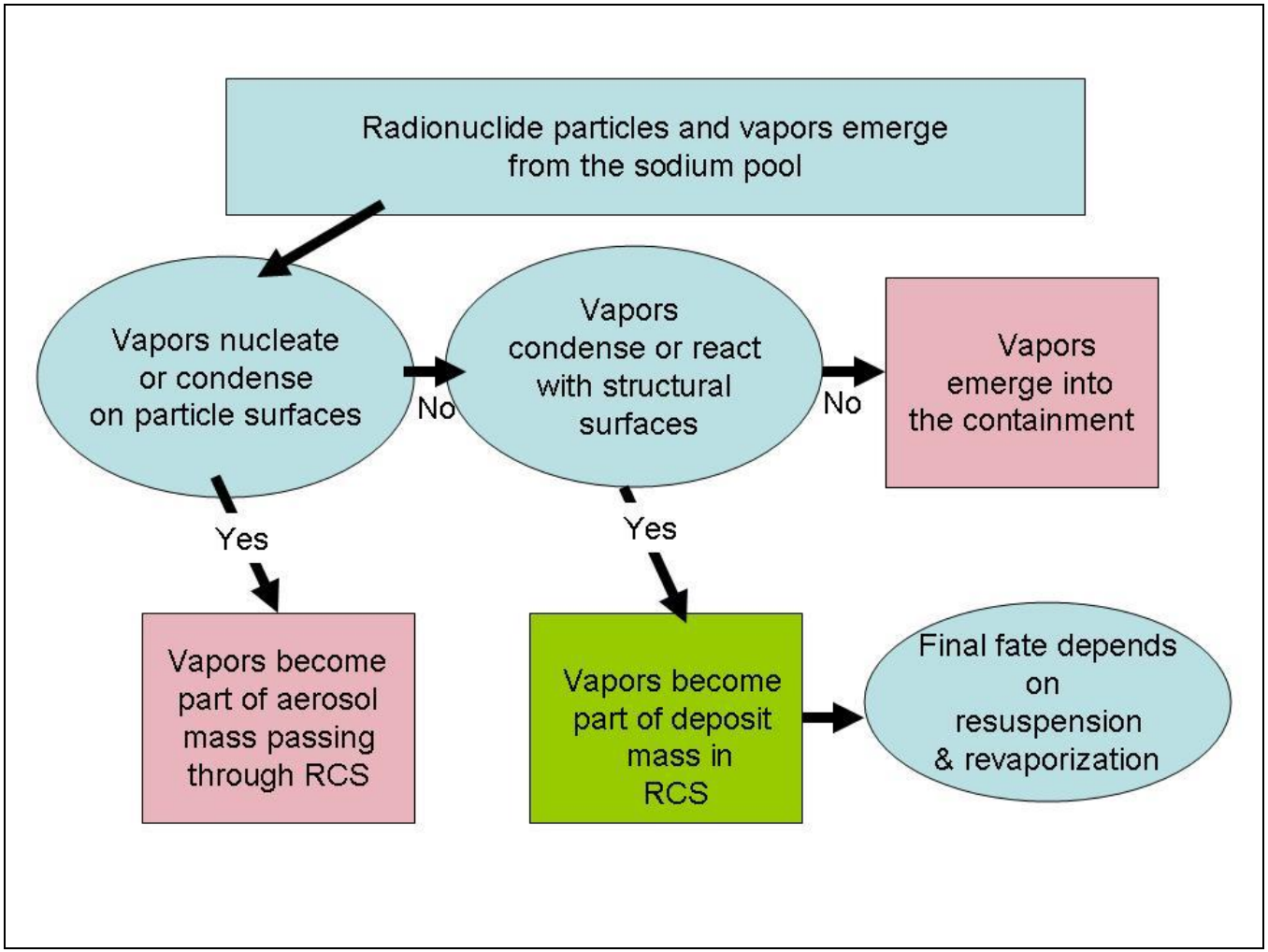


**Figure 2. Behavior of vapors released from the surface of the sodium pool.**

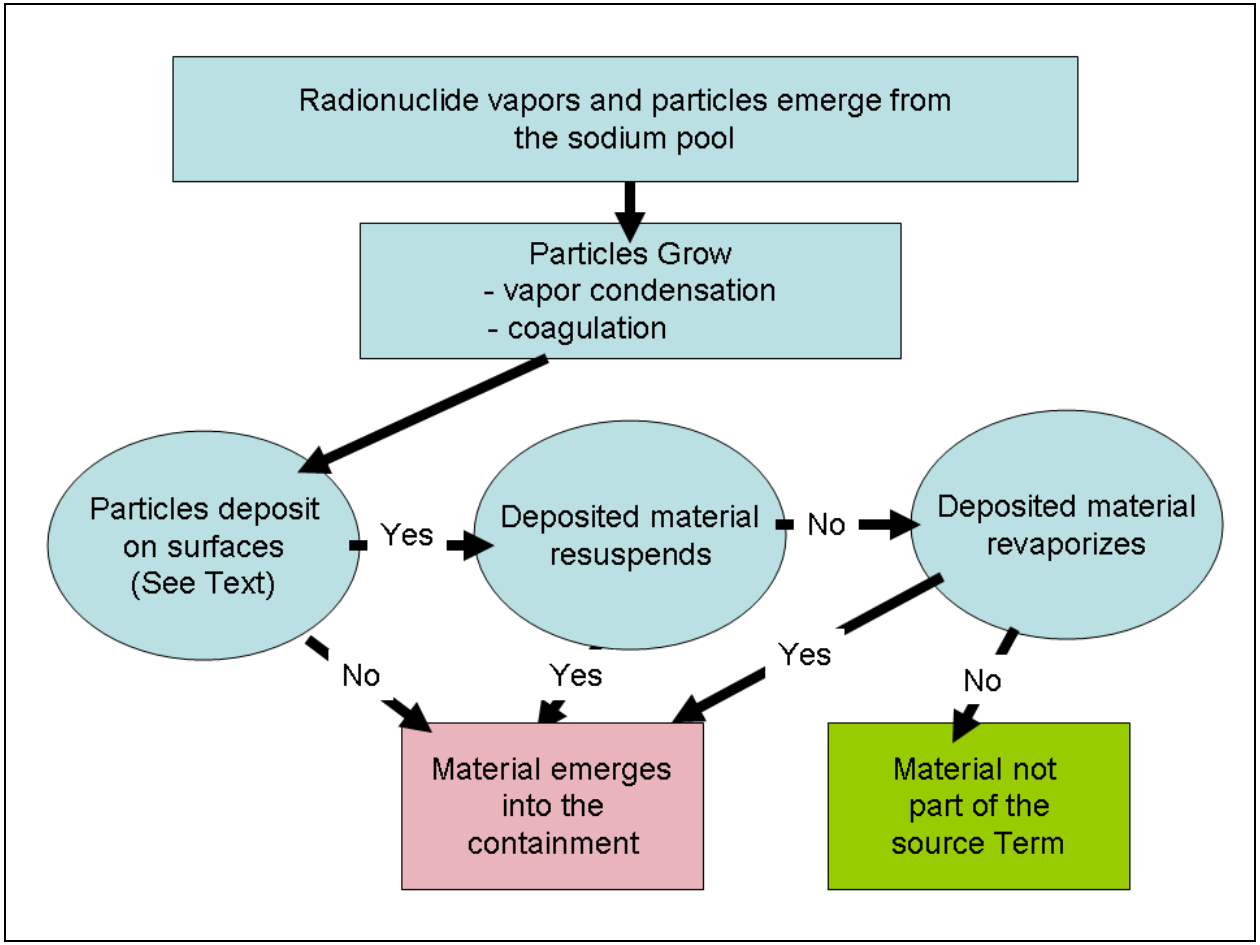


**Figure 3. Behavior of particles released from the surface of the sodium pool.**

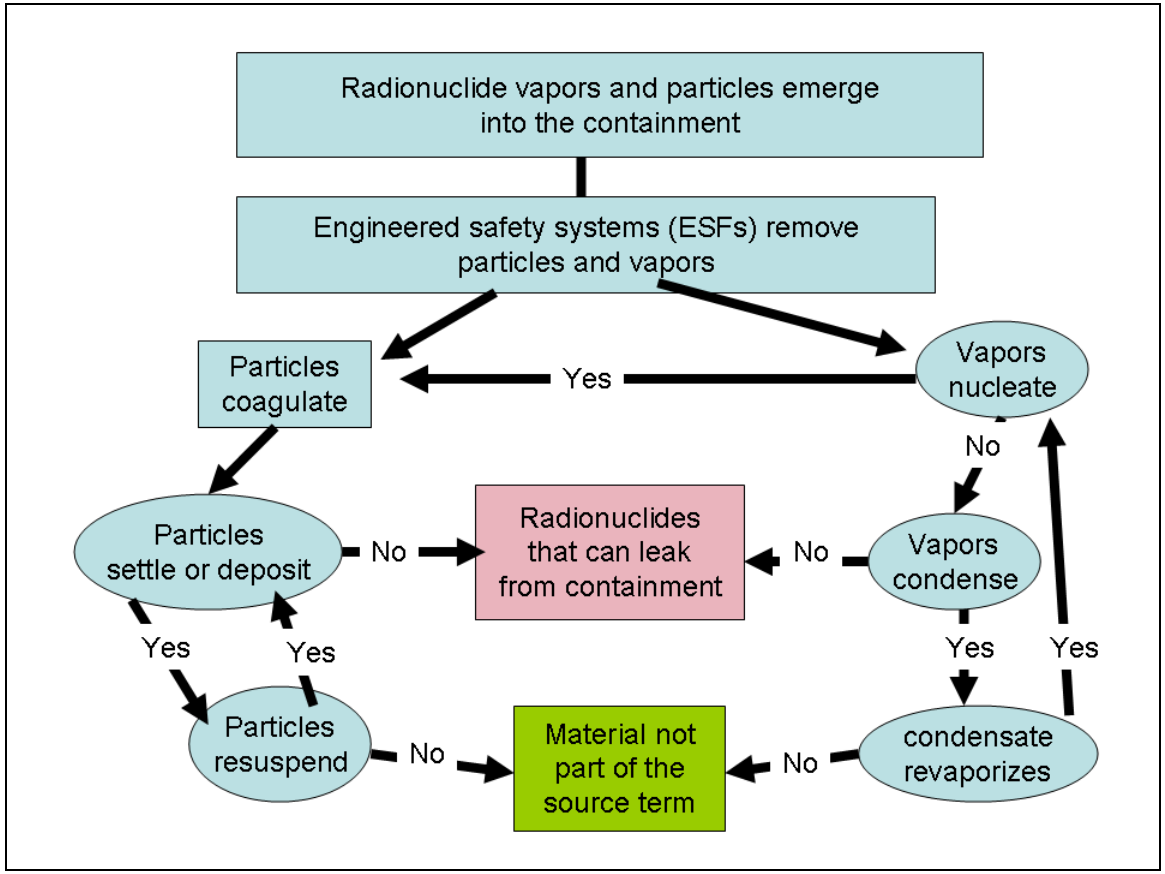


**Figure 4. Behavior of radionuclides in the reactor containment.**

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<br>**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<br>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<br>**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<br>**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<br>**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 51rinates, 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<br>**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 | Importance Ranking by Experts | | | | | | Comment |
|---|---|---|---|---|---|---|---|
| **Vaporization release from the free surface of a sodium pool** | 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<br>**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<br>**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 | Importance Ranking by Experts | | | | | | Comment |
|---|---|---|---|---|---|---|---|
| **Radionuclide transport from sodium pool through the reactor coolant system** | 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<br>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<br>**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<br>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<br>Importance rank: **High** | | Med | Low | Med | Low | **Med** | |
| Final morphology of the fuel debris<br>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<br>**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<br>**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<br>**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 | 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** | |
| 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<br>**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 | Research Need Ranking by Experts | | | | | | Comment |
|---|---|---|---|---|---|---|---|
| **Vaporization release from the free surface of a sodium pool** | 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<br>**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<br>**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 | Research Need Ranking by Experts | | | | | | Comment |
|---|---|---|---|---|---|---|---|
| **Radionuclide transport from sodium pool through the reactor coolant system** | 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 | Research Need Ranking by Experts | | | | | | Comment |
|---|---|---|---|---|---|---|---|
| **Radionuclide transport from sodium pool through the reactor coolant system** | Expert A | Expert B | Expert C | Expert D | Expert E | **Consensus** | |
| 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 | Research Need Ranking by Experts | | | | | | Comment |
|---|---|---|---|---|---|---|---|
| **Radionuclide transport from sodium pool through the reactor coolant system** | 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<br>**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<br>**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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