ERI/NRC 11-211

PEER REVIEW OF ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS USING HIGH-BURNUP AND MIXED OXIDE FUEL

Work Performed under the Auspices of the
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. 20555

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LIST OF ACRONYMS AND ABBREVIATIONS

Item	Definition
ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFW	Auxiliary Feed Water
AST	Accident Source Term; sometimes also referred to as Alternative Source Term
ATWS	Anticipated Transient Without Scram
B/U	Burnup
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CE	Combustion Engineering
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CRDA	Control Rod Drop Accident
CSARP	Cooperative Severe Accident Research Program
CV	Control Volume
DBA	Design Basis Accident
EOP	Emergency Operating Procedures
EQ	Equipment Qualification
ERI	Energy Research, Inc.
FHA	Fuel Handling Accident
FM	Fuel Melt
HBU	High Burnup
HEU	High Enriched Uranium
HPCI	High Pressure Coolant Injection
IBRAE	Institute of Nuclear Safety of the Russian Academy of Sciences

Item	Definition
IC	Internal Cooler
ICP-AES	Inductively Coupled Plasma Atomic Emission Spectrometry
IPE	Individual Plant Examination
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
LBU	Low Burnup
LEU	Low Enriched Uranium
LOCA	Loss Of Coolant Accident
LRF	Large Release Frequency
LWR	Light Water Reactor
MCAP	MELCOR Cooperative Assessment Program
MCCI	Molten-Core Concrete Interaction
MOX	Mixed Oxide Fuel
MSLB	Main Steam Line Break
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSC	(Japan Secretariat of the) Nuclear Safety Commission
ORNL	Oak Ridge National Laboratory
OSU	Ohio State University
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling (System)
RCS	Reactor Coolant System
REA	Rod Ejection Accident
RES	Office of Nuclear Reactor Research
RG	Regulatory Guide

Item	Definition
RIA	Reactivity Initiated Accident
RN	Radionuclide
RPV	Reactor Pressure Vessel
S/V	Safety Valve
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SMR	Small Modular Reactor
SNL	Sandia National Laboratories
SOARCA	State-of-the-Art Reactor Consequence Analysis
SRP	Standard Review Plan
SSC	Structures, Systems and Components
ST	Source Term
STCP	Source Term Code Package
TEDE	Total Effective Dose Equivalent
TEPCO	Tokyo Electric Power Company
TGT	Thermal Gradient Tube
TID	Technical Information Document
TMI	Three Mile Island

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EXECUTIVE SUMMARY

The characteristics of the radionuclide releases from the core into the containment for the currently licensed nuclear power plants are set forth in Regulatory Guides (RGs) 1.3 and 1.4. These regulatory guides are derived from the 1962 Atomic Energy Commission (AEC) report, Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," which is intended for application to offsite dose calculations as part of the reactor licensing process.

Using the results from research conducted on severe accidents and radionuclide releases since the publication of TID-14844, NUREG–1465 sets forth proposed source terms for the assumed releases into the containment in light-water reactors (LWRs) following an event that results in significant core damage. The NUREG–1465 source terms are based on more realistic assumptions than those used in TID-14844. The revised source terms in NUREG–1465 are primarily based on experiments and calculations using the Source Term Code Package (STCP), which is applicable to low-burnup UO_2 fuel (i.e., less than 40 gigawatt days per metric ton of uranium [GWd/MTU]).

In 2001 and 2002, the U.S. Nuclear Regulatory Commission (NRC) formed a review panel to examine the applicability of NUREG–1465 to reactors using a high-burnup (HBU), low-enriched uranium (LEU) fuel and to reactors using mixed oxide (MOX) fuel (i.e., mixed uranium and plutonium). The panel proposed changes to the NUREG–1465 source term for such applications. In addition, the review panel drew attention to changes in understanding LEU fuel fission product release behavior that have emerged after NUREG–1465 was issued.

Since the publication of NUREG–1465, the NRC has sponsored the development and incorporation of improved models into the MELCOR computer code for use in severe accident and source term studies. These models have been assessed using limited available experimental data including those from France and Japan—that encompass the effects of fuel burnup up to 60 GWd/MTU for UO_2 and MOX fuels. Based on the recommendations of the 2002 panel review, the NRC commissioned Sandia National Laboratories (SNL) to use MELCOR and determine a more defensible basis for source term estimates applicable to LWRs operating with HBU and MOX fuel.

It became apparent during the course of this work that advances in modeling the progression of severe accidents would result in source terms for low-enriched fuel significantly different from those found in NUREG–1465. This program also observed that changes between the NUREG–1465 source terms and those generated for the HBU and MOX fuels resulted primarily from advances in modeling and not from differences in fuel types or burnup.

Energy Research, Inc. (ERI), under NRC sponsorship, organized and assembled the present peer review committee with the objectives of (1) providing an independent assessment of the suitability of the SNL-recommended source terms for reactors with HBU and MOX fuel for equipment qualification and reactor siting dose calculations; and (2) issuing a final report describing the technical findings of the committee, including conclusions, observations and recommendations.

The committee agreed that there are ample justifications for changing the source terms in NUREG–1465. The committee also reached the following conclusions:

- The proposed source terms in the referenced report are technically justified and appropriate.
- Although the revised source terms are state-of-the-art, they are based on limited experimental data.
- The assumed magnitude of the gaseous iodine into the containment that is referenced in the report (i.e., 5 percent) is highly uncertain. This assumption should be revised pending the completion of ongoing research programs.
- The preliminary observations from the Fukushima Dai-ichi events do not change the findings and conclusions of the peer review committee.

In addition, the peer review committee reached the following recommendations:

1. Modify the revised source terms in the referenced document to
    a. eliminate the gap-release phase;
    b. consider the peer review committee comments on the potential under-prediction of the plutonium release and if warranted, modify such releases; and
    c. consider the preliminary information available from Fukushima Dai-ichi and state whether the preliminary data confirm or contradict any assumptions or results in the proposed accident source terms or methodology.
2. Issue a separate report that provides
    a. an expanded discussion of iodine chemical forms to include a summary of recent experimental and analytical work;
    b. a more complete explanation of the source terms for applications where late releases are important;
    c. additional information made available by SNL to the committee during the peer review meetings and technical exchanges (e.g., graphs of the different distribution functions);
    d. additional information about the progression of fuel degradation calculated with MELCOR for the different accident sequences; and
    e. tables indicating the quantity and composition of core debris at the time of reactor pressure vessel lower head failure.

1. INTRODUCTION

1.1 Background

The reactor siting criteria in Title 10 to the *Code of Federal Regulations*, Part 100 (10 CFR Part 100), "Reactor Site Criteria," require (1) the postulation of release of radionuclides resulting from a "substantial meltdown" of the core into the containment due to an accident; and (2) an evaluation of the potential radiological consequences with the assumption that the containment remains intact but leaks at its maximum allowable leak rate. Radioactive material escaping from the containment is often referred to as the "radiological release into the environment." The radiological release is obtained from the containment leak rate and knowledge of the airborne radioactive inventory in the containment atmosphere. The radioactive inventory within the containment is referred to as the "in-containment accident source term." In addition to site suitability, the regulatory applications of this source term affect the design of a wide range of plant systems.

In 1962, the AEC published TID-14844 [1]. This report symbolizes the first time that the AEC provided licensees with basic information and guidance pertaining to calculations that were necessary to support the siting criteria specified in 10 CFR Part 100. The TID-14844 "source terms" were based on the understanding of radiological release and transport mechanisms at that time. This release consisted of (1) 100 percent of the core inventory of noble gases, (2) 50 percent of the core inventory of iodine (assuming that half will be deposited on interior surfaces), and (3) 1 percent of the remaining solid fission products. It is important to note that the latter was removed from consideration in RG 1.3 [2] and RG 1.4, [3]. However, the 1 percent of the solid fission products is considered in certain areas, such as equipment qualification.

Subsequent to the accident at Unit 2 of Three Mile Island in 1979, the NRC and the nuclear industry sponsored major research initiatives to reassess the technical bases for severe accident source terms. In 1985, the NRC published NUREG–0956 [4], which documented the results of the NRC reassessment process, including the NRC's analytical procedures for estimating accident source terms for nuclear power plants. These analytical procedures and associated computer codes and models [5] formed the basis for the risk reassessment study of five representative nuclear power plants. The study was published as part of NUREG–1150 [6]. Subsequently, these studies formed the basis for the revised source term recommendations for LWR regulatory actions that were published in NUREG–1465 [7]. NUREG–1465 provided what the NRC referred to as a "source term" or "accident source term" released into the containment following a substantial meltdown of the reactor core. This source term was based on an improved level of understanding of LWR accidents and radionuclide release and transport behavior compared to the information available at the time that TID-14844 was published. NUREG–1465 as published, was proposed for use in LWR designs and was intended to form the basis for the development of regulatory guidance. It is important to recognize that the in-vessel release phase of an accident is the only phase considered in the most typical regulatory application (i.e., assessment of offsite and control room dose). Furthermore, the evolution of the accident source term led the NRC to issue 10 CFR 50.67, so as to specify the accident source term as part of the design basis for plants and to specify the total effective dose equivalent (TEDE) (versus only for siting, as described in 10 CFR 100).

That regulatory guidance came into being with the publication in 2000 of RG 1.183 [8], "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." RG 1.183 provided guidance to licensees of operating nuclear power reactors on (1) acceptable applications of alternative accident source terms; (2) the scope, nature, and documentation of associated analyses and evaluations; (3) consideration of the effects on analyzed risk; and (4) content of submittals to the NRC. RG 1.183 established an acceptable alternative source term that identifies the significant attributes of other source terms that may be used in an application. RG 1.183 also identifies acceptable assumptions to be used in a radiological analysis in conjunction with alternative source terms.

From a regulatory viewpoint, the publication of RG 1.183 was significant. A series of regulatory guides and NUREG–0800 [9] (Standard Review Plan [SRP]) chapters were predicated on the use of 10 CFR 100.11 [10], which discusses exclusion areas, low population zones, and population distances, and on the use of TID-14844 [1]. RG 1.183 states:

> [m]any of those analysis assumptions and methods are inconsistent with the accident source terms and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST [acceptable alternative source term]. This guidance supersedes corresponding radiological analysis assumptions provided in other regulatory guides and in SRP chapters when used in conjunction with an approved AST [acceptable alternative source term] and the TEDE criteria provided in 10 CFR 50.67. The affected guides will not be withdrawn as their guidance still applies when an AST [acceptable alternative source term] is not used.

The following regulatory guides were affected by the publication of RG 1.183:

- RG 1.3, which discusses assumptions used for evaluating potential radiological consequences of a loss-of-coolant accident (LOCA) for boiling-water reactors (BWRs) [2].

- RG 1.4, which discusses assumptions used for evaluating potential radiological consequences of a LOCA for pressurized water reactors (PWRs) [3].

- RG 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)" [11].

- RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)" [12].

- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" [13].

- RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" [14].

In 2001 and 2002, the NRC convened an expert panel to revise the reactor accident source terms described in NUREG–1465 [6]. This undertaking was prompted by interest in having source terms that are applicable to conventional reactor fuel at HBUs (i.e., more than 55 GWd/MTU) and to MOX fuel that includes weapons-grade plutonium dioxide. In formulating the proposed changes to the NUREG–1465 source term for application to reactor accident analyses for HBU and MOX fuel, the review panel drew attention to changes in understanding LEU fuel radionuclide release behavior. These changes have emerged since the publication of NUREG–1465, as a result of improved computer codes and analytic capabilities and the findings of major experimental investigations of radionuclide release and transport behavior under reactor accident conditions (e.g., Phébus-FP tests, VERCORS experiments, etc.).

Based on the results and recommendations of the 2002 panel review [15], NRC initiated a program at SNL to determine a more defensible basis for source term estimates that are applicable to LWRs operating with HBU and MOX fuel. The NRC-sponsored activities at SNL that are documented in SAND2011-0128, "Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel," [16] include research on HBU and MOX fuels that is applicable to both BWRs and PWRs. This work analyzed dominant sequences leading to core damage using a version of the MELCOR severe accident code that was modified for and benchmarked against recent radiological releases and transport experiments, including experiments involving HBU and MOX fuel.

Tables generated by the expert panel for NUREG–1465 were updated in Reference [16] using the results of the SNL program. It became apparent during the course of this work that advances in modeling the severe accident progression could allow changes in the low-enriched fuel source terms in NUREG–1465 [6] (see Tables 3.1 and 3.2 of this report that compare NUREG–1465 results to the results in Reference [16]). Observations in this study noted that changes between the NUREG–1465 source terms and those generated for the HBU and MOX fuel during this program resulted primarily from advances in modeling and not from differences between the fuel types. The results of the SNL activities that form the basis for the proposed revisions to the HBU and MOX source terms are documented in several SNL reports [17,18,19,20,21, 22], which form the basis for the synthesis report [16] that is the subject of the present review.

If adopted by NRC, the licensees may choose to use the updated source term primarily for application to radiological design basis accidents (DBAs), as related to offsite and control room dose calculations for licensing.

1.2 Committee Charter

The charter of the peer review committee is to (1) provide an independent assessment of the suitability of the recommended revised source terms for reactors with HBU and MOX fuel, as documented in the source term synthesis report for application to containment equipment qualification and siting dose calculations; and (2) issue a final report describing the technical findings of the Committee, including conclusions and observations.

It is important to point out that even though several committee members commented in their early evaluations that MELCOR deficiencies should be also addressed by the committee. Even though it was concluded that a full review of MELCOR models was beyond the scope of the committee charter, some peer review members considered the impact of MELCOR modeling uncertainties on the source term results.

1.3 Peer Review Committee Membership

In establishing this peer review committee, the NRC brought together a group of subject matter experts to review the proposed source terms in the source term synthesis report. Committee membership includes experts who have performed source term experiments, analytic modeling of fission product release and transport, and regulatory applications of accident source terms. Several members are intimately familiar with the MELCOR models and their experimental assessment bases. These experts are also fully cognizant of the regulatory basis for applying source terms in modeling severe accident progression.

The peer review committee consisted of the following members:

- Dr. Bernard Clément, Insitut de Radioprotection and Surete Nucleaire (IRSN)
- Dr. Richard Denning, Ohio State University (OSU)
- Dr. Akihide Hidaka, Japan Atomic Energy Agency (JAEA)
- Dr. Tom Kress, Consultant
- Dr. David Leaver, Worley Parsons
- Dr. Sudarshan Loyalka, University of Missouri–Columbia

Dr. Mohsen Khatib-Rahbar chaired the committee and prepared the summary report.

1.4 Peer Review Process

The process for conducting the peer review follows the objectives and applications defined by the NRC. The revised source terms are designed to extend the existing approach for defining releases into the containment for both PWRs and BWRs operating under an HBU or using MOX fuel. The peer review committee was responsible for assessing whether the proposed source terms are applicable to containment equipment qualification and siting dose calculations.

The peer review process considered the following general areas, (1) the adequacy of the proposed approach, (2) the adequacy of the models used to reproduce the results of the available experimental data, and (3) the representation of uncertainties in release quantities and other characteristics associated with the proposed source terms.

The peer review process included an evaluation of whether (a) the PHEBUS-FP program, the VERCORS tests, and the VEGA tests in combination provide an adequate basis for determining the likely radionuclide releases from HBU and MOX fuel; (b) the CORSOR-Booth model in MELCOR adequately reproduces the results of these tests and programs; (c) appropriate severe accident sequences (for severe accidents involving MOX or HBU fuel) were chosen as input to the determination of the revised source terms; (d) the proposed source term is representative or typical, rather than conservative or bounding; (e) the application of the revised

source term is adequate for determining containment equipment qualification parameters and performing reactor siting dose calculations; (f) the documented technical basis, including the evaluation of uncertainties regarding the proposed source term, is adequate; and (g) what is known about the Fukushima Dai-ichi events based on observations to date (see References [23] and [24]) would alter the findings and conclusions of this report.

The peer review committee met together twice in Rockville, Maryland at NRC headquarters. At the initial meeting, the committee interacted with NRC and SNL and was briefed on the objectives of its peer review, the contents of the source term synthesis report, and the SNL program to update source terms. Subsequent to this meeting, committee members developed requests for additional information (RAIs) for SNL regarding the source term synthesis report and supporting documentation. NRC transmitted these RAIs to SNL and SNL responded.

At another meeting of committee members, they discussed the SNL responses to the RAIs; tentative conclusions; and how the committee should proceed. In addition, the committee held a number of conference calls with ERI, NRC, and SNL to resolve specific questions and issues. After resolving the issues, committee members documented their evaluations of the SNL responses to RAIs and the proposed source terms. Committee members then submitted their final evaluation, findings, and recommendations regarding the proposed source terms. During the committee evaluation process, the committee received preliminary information from the Japanese and other sources and studies (References [23 and 25]) issued in the aftermath of the Fukushima Dai-ichi events. The committee considered the information on Fukushima Dai-ichi, and each committee member amended his final report, as needed, to discuss the implications of the available Fukushima Dai-ichi information. Subsequent to completion of committee member inputs, the Japanese Government released Reference [24], which the committee members also received for their consideration.

ERI assembled the various RAIs; the SNL responses to the RAIs; and the committee evaluations, findings, and recommendations as documented here. The committee also reviewed and endorsed this report.

1.5 Report Outline

Section 2 summarizes NUREG–1465 source terms and regulatory uses of radiological source terms. Section 3 summarizes the proposed source terms for HBU and mixed oxide fuels. Section 4 contains a summary of the peer review committee findings and observations relative to review objectives based on the individual committee member reports that are documented in Appendix C. Section 5 contains the conclusions and recommendations. Section 6 lists the references. Appendix A contains the NRC's position on the impact of NUREG–1465 on RG 1.183 [7]. Appendix B lists the committee's questions (RAIs), the SNL responses to those questions, and the committee's' evaluation of the SNL responses. Appendix D discusses preliminary information about the Fukushima Dai-ichi events. Appendix E discusses plutonium releases during severe accidents. Appendix F discusses the presentations at the two meetings at NRC headquarters with the NRC, SNL, and committee members. Appendix G includes short biographies of the committee members.

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2. NUREG–1465 SOURCE TERMS AND REGULATORY CONSIDERATIONS

2.1 NUREG–1465 Source Terms

Radiological releases into containment under severe accident conditions can be generally categorized in terms of four (4) phenomenological phases associated with the degree of core damage and degradation, reactor pressure vessel integrity, and, as applicable, attack on the structural concrete in the reactor cavity by molten core materials.

1. **Gap Release Phase**: The release, following a clad rupture, of fission gases and radionuclide vapors that accumulate during operation and follow the initial phase of accidents in the fuel-cladding gap and fuel rod plena.

2. **In-Vessel Release Phase**: The release of radionuclides during core degradation and prior to core debris penetration of the reactor pressure vessel lower head.

3. **Ex-Vessel Release Phase**: The release of radionuclides from core debris relocated from the reactor pressure vessel onto the containment floor following a failure of the reactor pressure vessel lower head. This release is predominantly due to core debris interactions with concrete, but could include releases caused by high-pressure melt ejection phenomena and even by ex-vessel steam explosions.

4. **Late In-Vessel Release Phase:** This phase starts once the core debris has penetrated the reactor pressure vessel lower head. Radionuclide releases come from fuel that was not relocated/expelled from the reactor vessel and from the revaporization of radionuclides that were deposited on surfaces within the reactor coolant system.

The core inventory release fractions for the NUREG–1465 [6] accident source terms are listed in Table 2.1 for BWRs and in Table 2.2 for PWRs by radionuclide groups. It should be noted that the late in-vessel release fractions for halogens and alkali metals for BWRs listed in Table 2.1 are different from those reported erroneously in the final version of NUREG–1465.

Table 2.1 BWR Inventory Release Fractions for the NUREG–1465 Accident Source Term

Group	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Noble Gases	0.05	0.95	0	0
Halogens	0.05	0.25	0.30	0.1
Alkali Metals	0.05	0.20	0.35	0.1
Tellurium Group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Lanthanides	0	0.0005	0.005	0
Cerium Group	0	0.0002	0.005	0

Table 2.2 PWR Inventory Release Fractions for the NUREG–1465 Accident Source Term

Group	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Noble Gases	0.05	0.95	0	0
Halogens	0.05	0.35	0.25	0.1
Alkali Metals	0.05	0.25	0.35	0.1
Tellurium Group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Lanthanides	0	0.0005	0.005	0
Cerium Group	0	0.0002	0.005	0

NUREG–1465 [6] recognized that for degraded core or core-melt accidents, the gap release phase may overlap to some degree with the early in-vessel release phase. Therefore, the release magnitude was taken as an initial release of 3 percent of the volatile fission products (noble gases, iodine, and cesium), which are in the gap between the fuel pellet and the cladding, plus an additional release of 2 percent for the duration of the gap release phase.

The early in-vessel release fractions presented in Tables 2.1 and 2.2, except for the low-volatile nuclides, are generally the mean values of the uncertainty distributions for a typical low-pressure core-melt accident scenario documented in NUREG/CR–5747 [26]. The 75th percentile value was selected for the low volatile nuclides on the basis that it bounds most of the range of values, without undue influence by the upper tail of the distribution.

The ex-vessel release fractions for the revised accident source terms are generally the mean values of the uncertainty distributions for releases associated with core-concrete interactions documented in NUREG/CR–5747 [26]. The ex-vessel releases of tellurium and the low-volatile nuclides are reduced as a result of comments received at the time.

The late in-vessel release fractions presented in Tables 2.1 and 2.2 are also generally the mean values of the uncertainty distributions for a typical low-pressure core melt accident scenario documented in NUREG/CR–5747 [26]. These releases are associated with the revaporization of radionuclides retained in the reactor coolant system and their subsequent release into the containment after the vessel failed.

Table 2.3 shows the onset and duration of each release phase in the NUREG–1465 source terms for BWRs and PWRs. The specified onset is the time following the initiation of an accident (i.e., time = 0). It should be also noted that the rate of release of fission products into the containment was assumed to be constant during the duration shown.

The selected timing was intended to be typical of a core melt scenario, except for the onset of the release of gap activity (the duration of coolant activity), which was selected to reflect the earliest calculated time of fuel rod failure under design-basis accident (DBA) conditions (i.e., a large-break LOCA).

Table 2.3 Timing of Release Phases

Release Phase	BWRs		PWRs	
	Onset	Duration	Onset	Duration
Gap Release (seconds)	30	0.5	10–30	0.5
Early In-Vessel (hours)	0.5	1.5	0.5	1.3
Ex-Vessel (hours)	2	3	1.8	2
Late In-Vessel (hours)	2	10	1.8	10

2.2 Attributes of an Acceptable Alternative Accident Source Term

RG 1.183 [7] uses NUREG–1465 as its basis, and it offers licensees an option that they can use in place of TID-14844. These attributes include the following:

- The accident source term must be based on major accidents and hypothesized for the purposes of design analyses or consideration of possible accidental events, which could result in hazards not exceeded by those from other accidents considered credible. The accident source term must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.

- The accident source term must be expressed in terms of times and rates of appearance of radioactive fission products released into the containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

- The accident source term must not be based upon a single accident scenario. It must instead represent a spectrum of credible severe accident events. Insights from risk assessments may be used (i.e., not a single risk-significant accident but a range of events to be considered). Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

- The accident source term must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.

- The accident source term must be peer reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the accident source term.

During their deliberations, the peer review committee members considered the existing regulatory basis for allowing the use of accident source terms and the full spectrum of applications in which accident source terms are used.

2.3 Regulatory Use of NUREG–1465

The NRC clarified for the peer review committee the manner in which NUREG–1465 [6] is currently used in licensing decisions. These decisions include site suitability to establish boundaries, control room habitability, equipment qualification, and post-accident vital area access.

The following regulatory requirements pertain to these topics:

- 10 CFR Part 100 discusses siting criteria [27]
- 10 CFR 50.67, "Accident Source Term" [28]
- 10 CFR 50.34, "Contents of Applications; Technical Information" [29]
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion–19, "Control room" [30]

The accident source term issue is further elaborated in RG 1.183 [7], which discusses the applicability of the source terms in TID-14844 [1] and NUREG–1465 [6].

There are specific regulatory considerations that are important to understand when reviewing the proposed source terms [16]. The following sections discuss these considerations in some detail using information supplied by the NRC staff (see Appendix A).

2.3.1 Use of the Gap Phase

In the analysis of core meltdown accidents, cladding failure occurs at different times across the core and there is not a distinct gap release phase. Therefore, the Committee proposes to incorporate this into the in-vessel release.

When evaluating LOCA radiological DBA submittals, some containment designs have release pathways that isolate during the time frame commonly referred to as the gap release phase (e.g., the containment purge system). Consequently, the timing of the onset of the gap release and the magnitude of the release during this phase influence the dose analysis. The assumed time within which these pathways isolate does not extend beyond the current NUREG–1465 gap release duration, and the release magnitude is only a fraction of the total source term. Therefore, given the methods used to calculate releases for regulatory purposes, the NRC anticipates that combining the gap and the in-vessel phase releases will have a conservative but small impact on the LOCA dose analyses and will therefore be acceptable.

For non-LOCA DBAs where the fuel cladding is postulated to be breached, but the fuel melt is not postulated, NUREG–1465 [6] gap release fractions are not used unless the event involves the entire core. Typically, accidents within the design basis that are non-LOCA do not involve the entire core—which would be the case for a full core meltdown assumed for the LOCA—but involve only those fuel pins that experience a departure from nucleate boiling. Also for non-LOCA DBAs, the timing of the gap release into the reactor coolant system and the containment

is conservatively assumed to be instantaneous. Therefore, for these accidents, RG 1.183 [7] does not rely upon NUREG–1465 gap release fractions or on the timing of the releases.

Generally, the peer review committee concludes from the previous staff analysis that there is no regulatory reason for maintaining the gap release phase.

2.3.2 Other Uses of NUREG–1465 for Non-LOCA Design-Basis Accidents

Because of the caveats in NUREG–1465 [6] regarding its applicability to existing nuclear power plants and to fuel burnup levels in excess of 40 GWd/MTU, the NRC only uses selected information from NUREG–1465, most of which is associated with the isotopic mix and release fractions for accidents involving fuel damage.

RG 1.183 assumes that the released iodine from the reactor coolant system into the containment in a postulated accident is consistent with the speciation assumed in NUREG–1465 [6] (i.e., 95 percent cesium-iodide aerosol, 4.85 percent elemental iodine and 0.15 percent organic iodine). RG 1.183 uses RG 1.77 release fractions for the two non-LOCA DBAs where fuel melt is typically considered. These accidents include the BWR control rod drop accident (CRDA) and the PWR rod ejection accident (REA). Releases from fuel due to melting are assumed to be instantaneous, and they are therefore not dependent on NUREG–1465 [6].

The amounts of the radio-iodine (gaseous and otherwise) contribution to an accident source term can directly affect containment designs that must cope with the design basis source term (such as the allowed leakage rate, containment atmosphere removal concepts, water chemistry, etc.).

The peer review committee generally concludes from the earlier staff analysis that there is significant importance and uncertainty attached to the chemical forms and release fractions of iodine applicable to accident conditions.

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3. PROPOSED SOURCE TERMS FOR HIGH-BURNUP AND MIXED OXIDE FUELS

Reference [16] considers two BWR plants—Peach Bottom, a BWR/4 with Mark I containment, and Grand Gulf, a BWR/6 with Mark III containment. That reference also considers two PWR plants – Surry, a 3-loop Westinghouse design with a large-dry containment and Sequoyah, a 4-loop Westinghouse design with an ice-condenser containment. The BWRs considered are two of the three considered in NUREG–1465 [6], and the PWRs considered are two of the five considered in NUREG–1465.

For each of these plants, data concerning core axial and radial power distributions, end-of-cycle fission product inventories, and decay heat generation were obtained from plant operating data. Low burnup and HBU loading patterns were developed from licensee data for three consecutive early cycles and three recent cycles, respectively. The actual HBU patterns resulted in assembly burnup levels lower than desired for this study. Hence, ORIGEN was used to determine the inventories corresponding to peak assembly average burnup levels of 59 GWd/MTU and peak pin burnup levels of 62 GWd/MTU by extrapolating from the available results.

3.1 Proposed Source Terms

The proposed source terms for HBU and MOX fuels in Reference [16] are listed in Table 3.1 through Table 3.3 for BWRs and PWRs, respectively. These tables also include the corresponding NUREG–1465 [6] values for purposes of comparison.

3.2 Uncertainty

Reference [16] briefly discusses two sources of uncertainty—epistemic and aleatory uncertainty. Aleatory Uncertainty, also referred to as stochastic, is due to natural and unpredictable variations in the performance of the system under study. Epistemic Uncertainty results from/is attributed to a lack of knowledge about the behavior of a system that is conceptually resolvable.

In response to questioning from the peer review committee, SNL stated that there was no attempt to assess epistemic uncertainties in Reference [16]. Furthermore, a reconsideration of the use of the term “uncertainty” led the committee to conclude that its use is probably inappropriate, because it could be misunderstood to imply a level of knowledge about certain parameters that may mislead the reader. Consequently, the committee recommends to substitute the term “variability due to the selected sequences” in place of “uncertainty.”

Therefore, the uncertainties quantified as part of Reference [16] Tables 10, 11, and 15 result from a variability in accident sequences and as such, they do not include uncertainties associated with an incomplete knowledge of severe accident and radiological releases.

The committee finds that there are significant uncertainties associated with the gaseous forms of iodine and the current 5 percent estimate, which was included in the original basis document for NUREG–1456 that relied heavily on expert judgment.

Table 3.1 Proposed Source Term for BWRs Using HBU Uranium Dioxide Fuel NUREG–1465 values are in parentheses)

Release Mode	Gap Release	In-Vessel Release	Ex-Vessel Release	Late In-Vessel Release
Duration (hours)	0.16 (0.5)	8.0 (1.5)	2.9 (3.0)	12 (10)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr, Xe)	0.008 (0.05)	0.96 (0.95)	0.009 (0)	0.016 (0)
Halogens (Br, I)	0.002 (0.05)	0.47 (0.25)	0.013 (0.30)	0.39 (0.01)
Alkali Metals (Rb, Cs)	0.002 (0.05)	0.13 (0.20)	0.01 (0.35)	0.05 (0.01)
Alkaline Earths (Sr, Ba)	-	0.005 (0.02)	0.029 (0.10)	0.005 (0)
Tellurium Group (Te, Se, Sb)	0.002 (-)	0.39 (0.05)	0.002 (0.25)	0.33 (0.005)
Molybdenum [*] (Mo, Tc, Nb)	-	0.02 (0.0025)	0.003 (0.0025)	0.0055 (0)
Noble Metals (Ru, Pd, Rh, etc.)	-	0.0027 (0.0025)	[0.0025]	1.0×10^{-4} (0)
Lanthanides (Y, La, Sm, Pr, etc.)	-	1.4×10^{-7} (2×10^{-4})	5×10^{-5} (0.005)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	1.3×10^{-7} (2×10^{-4})	0.0021 (0.005)	-

[*] The Molybdenum Group releases fractions that NUREG–1465 assumes are identical to the release fractions of the Noble Metals Group in that report; hence, the Molybdenum and Noble Metal Groups listed here have identical values.

Table 3.2 Proposed Source Term for PWRs Using HBU Uranium Dioxide Fuel NUREG–1465 values are in parentheses)

Release Mode	Gap Release	In-Vessel Release	Ex-Vessel Release	Late In-Vessel Release
Duration (hours)	0.22 (0.5)	4.5 (1.5)	4.8 (2.0)	143 (10)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr, Xe)	0.017 (0.05)	0.94 (0.95)	0.011 (0)	0.003 (0)
Halogens (Br, I)	0.004 (0.05)	0.37 (0.35)	0.011 (0.25)	0.21 (0.10)
Alkali Metals (Rb, Cs)	0.003 (0.05)	0.23 (0.25)	0.02 (0.35)	0.06 (0.10)
Alkaline Earths (Sr, Ba)	0.0006 (0)	0.004 (0.02)	0.003 (0.10)	- (-)
Tellurium Group (Te, Se, Sb)	0.004 (0)	0.30 (0.05)	0.003 (0.25)	0.10 (0.005)
Molybdenum [†] (Mo, Tc, Nb)	-	0.08 (0.0025)	0.01 (0.0025)	0.03 (0)
Noble Metals (Ru, Pd, Rh, etc.)	-	0.006 (0.0025)	[0.0025]	-
Lanthanides (Y, La, Sm, Pr, etc.)	-	1.5×10^{-7} (2×10^{-4})	1.3×10^{-5} (0.005)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	1.5×10^{-7} (5×10^{-4})	2.4×10^{-4} (0.005)	-

[†] The Molybdenum Group releases fractions from NUREG–1465 that are assumed to be identical to the release fractions of the Noble Metals Group in that report; hence, the Molybdenum and Noble Metal Groups listed here have identical values.

Table 3.3 Proposed Source Term For PWRs Using HBU and MOX Fuel (NUREG–1465 values[‡] are in parentheses)

Release Mode	Gap Release	In-Vessel Release	Ex-Vessel Release	Late In-Vessel Release
Duration (hours)	0.36 (0.50)	4.4 (1.3)	6.5 (2.0)	16 (10)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr, Xe)	0.028 (0.050)	0.86 (0.95)	0.05 (0)	0.026 (0)
Halogens (Br, I)	0.028 (0.050)	0.48 (0.35)	0.06 (0.25)	0.055 (0.10)
Alkali Metals (Rb, Cs)	0.014 (0.050)	0.44 (0.25)	0.07 (0.35)	0.025 (0.10)
Alkaline Earths (Sr, Ba)	-	0.0015 (0.020)	0.008 (0.1)	9×10^{-5} (0)
Tellurium Group (Te, Se, Sb)	0.014 (0)	0.48 (0.05)	0.04 (0.25)	0.055 (0.005)
Molybdenum [§] (Mo, Tc, Nb)	-	0.27 (0.0025)	[0.0025]	0.024 (0)
Noble Metals (Ru, Pd, Rh, etc.)	-	0.005 (0.0025)	[0.0025]	3×10^{-4} (0)
Lanthanides (Y, La, Sm, Pr, etc.)	-	1.1×10^{-7} (0.0002)	3×10^{-5} (0.005)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	1.0×10^{-7} (0.0005)	5×10^{-4} (0.005)	-

[‡] Note that the NUREG–1465 report evaluated uranium dioxide fuel at 40 GWd/MTU and did not evaluate MOX fuel.

[§] The Molybdenum Group release fractions from NUREG–1465 are assumed to be identical to the release fractions of the Noble Metals Group in that report; hence, the Molybdenum and Noble Metal Groups listed here have identical values.

4. PEER REVIEW COMMITTEE FINDINGS AND OBSERVATIONS

The peer review committee considered the proposed source term and the supporting documentation. During the peer review process, two meetings were held to discuss several SNL reports [17,18,19,20,21,22] with the peer review committee. To support this process, the committee submitted a series of RAIs, to which SNL supplied detailed responses that committee members then reviewed and critiqued. This peer review process is documented in Appendix B at the end of this report.

Appendix C of this report includes individual observations and conclusions of the peer review committee members. The subsections that follow summarize these findings and observations.

4.1 General Approach

The accident analysis calculations underlying the study were conducted to specifically address the essential points in RG 1.183 [7] regarding "attributes of an acceptable alternative accident source term."

> *The accident source term must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. The accident source term must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.*
>
> *and*
>
> *The accident source term must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Insights from risk assessments may be used (i.e., not a single risk-significant accident, but a range of events to be considered). Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.*

The peer review committee endorses the approach taken in Reference [16]. The report documents an improved source term based on deterministic analyses of the likely severe accident sequences leading to core damage. The models used are available for public inspection and are more objective than the expert elicitation process used in NUREG–1465 [6] or in Reference [15]. The source terms are divided into four phases that match the NUREG–1465 format: gap release, in-vessel release, ex-vessel release, and late in-vessel release. The use of MELCOR 1.8.5 is well grounded and appropriate.

The peer review committee finds that Reference [16] and the supplemental documents (References [17,18,19,20,21,22]) provide a defendable technical basis for the proposed source term and the documentation supporting the source term is appropriate, thus satisfying this attribute of an acceptable alternative accident source term.

4.2 Selection of Plants and Core Radionuclide Inventories

The peer review committee finds that the nuclear power plants considered in Reference [16] reasonably represent the U.S. nuclear fleet, and the methods used to determine the radionuclide inventories are appropriate.

4.3 Accident Sequences

Accidents analyzed using the MELCOR 1.8.5 computer code included station blackout (SBO) accidents, transients involving the loss of injection or loss of decay heat removal, Anticipated Transient Without Scram (ATWS), and breaks in the reactor coolant system. When combined, the accidents consider 92 percent of the core damage frequency estimated for Peach Bottom, 98 percent of the core damage frequency estimated for Grand Gulf, 85 percent of the core damage frequency estimated for Surry, and 88 percent of the core damage frequency estimated for the Sequoyah plants based on the results of Individual Plant Examinations (IPEs) documented in NUREG–1560 [31]. No arrested accidents and operator interventions were included in the study.

The peer review committee finds that this spectrum of accidents is sufficient to satisfy the following stated attributes of an acceptable alternative accident source term:

> *The accident source term must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.*

Reference [16] provides the proposed source term in tabular form, thus replicating the format of NUREG–1465 [6]. The only change is the introduction of an additional radionuclide group—the Molybdenum group that includes Mo, Tc, and Nb. This is to address an important finding of the accident source term study, which is in response to (indirect) experimental findings, especially in the Phébus-FP program but also in response to analytic investigation findings that the cesium released from the fuel that is not in the form of cesium iodide (CsI) is present as a vapor or particulate cesium molybdate (Cs_2MoO_4), rather than the much the more volatile species cesium hydroxide (CsOH) as previously considered.

4.4 Representation of Source Terms

The peer review committee finds that the proposed source term (a) is expressed in an appropriate format, (b) represents the appropriate radionuclide groups, (c) reasonably allows the representation of chemical forms of released iodine to the best of our understanding today, and (d) is sufficient to satisfy this attribute of an acceptable alternative accident source term.

> *The accident source term must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.*

With regard to the need for a weighting process, SNL indicated that any defensible level of weighting would be about equal for each class of reactor, with the possible exception of ATWS. Although accident frequency weighting is possible, another approach was used that simplifies the analysis and is reasonable for both PWRs and BWRs.

The peer review committee discussed the use of the median versus the mean. Typically, when considering a large range of uncertainties, the mean value is more conservative than the median value. This finding was true for the original NUREG–1465 results, which included large epistemic uncertainties. The median value is generally considered more representative than the mean. For this reason, Reference [16] used the median value, which in instances such as an iodine release, led to more conservative results. It should also be pointed out that NUREG–1465 used the lesser of the mean and the 75th percentile.

The timing of releases in Reference [16] are of longer duration than in NUREG–1465 [6], and in many sequences there is a large early release followed by a much slower, later release to the containment. This has to do with improved models and methods in MELCOR regarding material release and transport. The peer review committee finds the information related to the rate and timing of releases to be appropriate.

In-vessel release fractions for volatile radionuclides such as iodine are larger than indicated in the NUREG–1465 source term. Consequently, there are larger inventories of the more volatile radionuclides such as iodine and tellurium deposited in the reactor coolant systems. Despite the longer duration of the ex-vessel release phases of accidents, release fractions for volatile radionuclides such as cesium, tellurium, and iodine are, on the whole, much lower than indicated by the NUREG–1465 source term. Phébus-FP experiments have shown substantial radionuclide releases from degrading reactor fuel in the form of aerosols. However, some of these tests (FPT-3) have also shown that some fraction of the radioactive iodine is released into the containment in gaseous as well as particulate form. Treatment of the release of iodine to containment as a mixture of gaseous species and metal iodide particulate has been adequately validated by the results of tests carried-out in the Phébus-FP program. It should be noted that the Phébus FPT-3 tests may not be prototypic of U.S. reactors, and analyses of the tests are still underway, but it appears that a large release of gaseous iodine arose from the degradation of the boron carbide control rod that is quite different from the degradation of BWR control blades seen in the other (U.S.) in-pile tests and modeled in the MELCOR code. In FPT-3, control rod degradation left very significant amounts of boron carbide available for steam oxidation to form boric acid, which in turn reacted with most metallic products (e.g., cesium), and reduced the formation of iodides with iodine. However, the large stainless steel to boron carbide ratio in control blades for BWRs is expected to alter this behavior. For instance, in the SNL DF-4 test, the boron carbide was completely dissolved in the melting stainless steel and subsequently, relocated out of the heated region of the reactor fuel. Furthermore, it is noted that the gaseous iodine concentration in the Phébus FPT-3 model containment declined even faster than the aerosol concentration declined. Thereafter, gaseous iodine chemistry in the containment model was rather similar to that observed in previous tests.

The precise ratio of gaseous and particulate forms of iodine released to the containment and the behavior of iodine within the containment are subjects of ongoing investigations. Simple descriptions of iodine dissolving and partitioning from aqueous solution are being questioned.

Investigations are focusing on the thermal and radiolytic chemistry of iodine in the gas phase and the aqueous phase as well as the interactions of gaseous and particulate iodine species with metallic and painted surfaces. Furthermore, considerable data are expected to emerge from the Fukushima accidents that should improve our understanding of core degradation, radionuclide release, and transport characteristics in BWRs.

Pending the outcome of these ongoing investigations, it is prudent to retain the prescription of iodine chemical form adopted in the NUREG–1465 source term. That is, 95 percent of the iodine released into the containment is assumed to be in the form of metal iodide particles that will agglomerate with other aerosol particles in the containment atmosphere. The remaining 5 percent of the iodine is released into the containment as a gaseous species.

Most of the iodine species released into containment will readily dissolve in water. The exception is insoluble silver iodide (AgI) that can be formed by a reaction of iodine with silver from silver-indium-cadmium control rods. Iodine dissolved in water at a pH of less than 7 can react both thermally and radiolytically to form molecular iodine and volatile organic iodides that will partition from the aqueous phase back into the containment atmosphere. Maintenance of basic pH in sump waters within the containment cannot rely on the availability of cesium hydroxide as the dominant chemical form of cesium released into the containment. Cesium molybdate is thought to be a more important chemical form. Cesium molybdate is water soluble, but its hydrolysis will not drive solutions to a basic pH.

The peer review committee finds that the differences between the new source terms and those in NUREG–1465 [6] are due to changes in state-of-the-art evaluation capabilities and not to differences in fuel burnup and/or fuel composition (i.e., UO_2 versus MOX).

4.4 Experimental Basis

The proposed source term is supported by a very active analytic program that calibrated MELCOR 1.8.5 to all historical and recent experimental programs, including those of other nations. This includes the most recent VERCORS, Phébus, and VEGA experiments and the relevant results of the ORNL experimental program. The source term synthesis report and supplementary material reviewed by the peer review committee during deliberations show remarkable improvement in the ability of MELCOR 1.8.5 to model test results. This was demonstrated for a full compliment of radionuclides that are of importance to the accident source term.

4.5 Releases

4.5.1 General

It is important to recognize that the in-vessel release phase is the only phase considered in the most typical regulatory application, the assessment of the site dose. Even though the MELCOR computer code is as good as any other available tool, the accurate modeling of severe accident processes is challenging. Some aspects of source term analysis can be analyzed with much greater accuracy than others. Because offsite doses tend to be dominated by the release of volatile radionuclides for which the fractional release from fuel tends to be high, the demand for

modeling accuracy is not as great as would be required for regulatory issues that are dominated by the less volatile radionuclides. Furthermore, there is substantially greater validation data available for the quantities of noble gases, iodine, cesium, and tellurium released under molten fuel conditions (and therefore more likely to be released in-vessel) than there is for more refractory radionuclides, such as cerium or plutonium. From a regulatory perspective, the upside uncertainty is bounded by a 100 percent release. For a 50 percent release fraction, the maximum uncertainty is only a factor of two (which is certainly sufficient for the regulatory application). For a nonvolatile radionuclide with a best-estimate release fraction of $1x10^{-7}$, the potential upside uncertainty is dramatically larger.

The ability to model in-vessel releases is better than the ability to model either ex-vessel or late-in-vessel releases. The ex-vessel release estimates for the proposed source terms are smaller than the NUREG–1465 values. These releases are also not considered in the assessment of site doses. The late in-vessel release is also very difficult to assess, in part because of our limited ability to predict the deposition of radionuclides in the vessel. Again, this is not a consideration for the principal regulatory applications of the source term.

4.5.2 Release from Fuel

MELCOR models for fission product release were adjusted for the calculations in Reference [12] using the results from the VERCORS RT6 experiment for HBU fuel and VERCORS RT2 experiment for MOX fuel. The adjustment modifies the cesium diffusion coefficient in the Booth model to reproduce the measured release kinetics as a function of temperature. For other radionuclide classes, the coefficient is scaled according to their relative volatility. This method/approach is standard procedure in modeling radionuclide releases during fuel damage in MELCOR .

Noble gas, iodine, and cesium releases are not substantially different in the proposed source terms than in NUREG–1465 [6]. The chemical forms of the fission products appear to be important only for iodine, cesium, molybdenum, ruthenium, and (perhaps) tellurium. Even though there is not yet enough evidence to change the chemical form of iodine specified in NUREG–1465, which is 95 percent CsI and 5 percent gaseous iodine, the committee notes that there is significant uncertainty in the magnitude of the release and the chemical form of the gaseous iodine released.

On the other hand, there is ample evidence that cesium does not behave as CsOH does, which is the assumption in past source term specifications. The chemical forms of cesium were modified in Reference [16] according to observations from the Phébus FP results that were attributed to the formation of Cs_2MoO_4 based on post-test calculations. Cs is split between CsI (stoichiometric value) and the less volatile Cs_2MoO_4. This new modeling was validated against the Phébus FPT-1 experiment. Note that putting part of molybdenum into Cs_2MoO_4 increases its release, which is in agreement with the experimental results. The effect is to slightly reduce the release of cesium and to increase the release of molybdenum. In the revised source term, molybdenum is now assigned its own group.

Similarly, experimental evidence indicates that tellurium will not be chemically bound to the Zircaloy cladding, as assumed in NUREG–1150 [6]. The peer review committee concludes that

the larger release fractions for the tellurium group in Reference [16] appear to be warranted. The ORNL-Booth models that were modified to better represent the results of recent data are sufficiently accurate.

The chemical form of iodine during a release is important because of the volatility and radiological importance of iodine. The chemical form of iodine is discussed in Reference [16] as this is the subject of continuing debate and research. The continuing evaluation of the Fukushima Dai-ichi event is expected to shed additional light on the chemical form of iodine. The outcome of future experiments and the conclusions drawn from the Fukushima Dai-ichi event are uncertain, but one possible outcome is a significantly higher gaseous iodine fraction in the containment, which would result in a greater release of iodine into the environment.

The predicted release from the non-volatile radionuclide groups is the most striking difference between the revised source terms in Reference [16] and the NUREG–1465 source terms. The difference is a reduction in more than three orders of magnitude. In particular, this affects the cerium group that includes plutonium, which has a very important radiological effect.

Although the peer review committee questioned the very small release value predicted by MELCOR for the Ce group, the committee did not find an adequate basis for recommending any changes. The peer review committee believes it is quite possible that the values calculated by MELCOR are low, perhaps even by orders of magnitude for non-volatile radionuclide groups, but the contribution of the low volatility radionuclides to site dose and other expected applications of the source term is expected to be very small, even when accounting for possible under-predictions.

4.5.3 Transport to Containment

The transport of radionuclides from the vessel to the containment, which is not normally analyzed when performing regulatory applications, is a complex physical and chemical process involving aerosols, gases, vapors, liquids, and solids at high temperatures and pressures. Fully realistic descriptions of such a complex process are presently not feasible, and certainly there is a need for more development in this area.

The fission product transport modeling in MELCOR considers deposition via condensation of vapors and mass transport of aerosols. The major difference involving the new proposed source term is in the new specification of the chemical form of cesium and molybdenum. This change tends to slightly reduce the re-vaporization of Cs and increases the re-vaporization of molybdenum. These changes do not markedly affect the source term other than for molybdenum and are considered to be acceptable.

4.6 Gap Release Phase

The concept of gap release phase is most useful when considering consequences relating to a situation such as a dropped fuel assembly. But accidents of the type under consideration in Reference [16] have a multitude of rods failing at different times and in different regions of the core. In such situations, the release from the fuel gaps is interspersed with non-gap releases.

The peer review committee believes that the gap release phase should be removed from the revised source term as a separate entity.

If there is a regulatory need to define a gap release, such as to determine the release if a fuel element is dropped, a gap release source term should be developed that is appropriate to that application.

4.7 Applicability

The proposed source terms are applicable to PWR and BWR designs that are representative of the existing and evolutionary LWRs for the purposes stated below. However, the peer review committee also recognizes that there is a need to consider what is the appropriate source term for Small Modular (light water) Reactors (SMRs), and there may be a need for changes to accident source terms for other technical regimes such as fuel handling accidents.

The proposed source terms are appropriate for their intended purposes which include LWRs with UO_2 or MOX fuel in a zirconium alloy cladding; the assessment of site doses and for equipment qualification; and for fuel with burnups up to 62 GWd/MTU for the peak rod.

In addition, the following are noted:

- Applications that could be heavily influenced by Late In-Vessel Releases will have substantially larger source term uncertainties.

- Source terms for light water SMRs require a type-specific analysis and/or justification for using any existing source term.

- The proposed source terms have not been evaluated for uses beyond DBA applications, such as in probabilistic risk assessments (PRAs) and emergency planning.

4.8 Uncertainty Analysis

Reference [16] does not consider epistemic uncertainties (i.e., due to incomplete knowledge of governing phenomena) from the overall distribution of source terms calculated for the various accidents. The peer review committee notes that the magnitude of the effort required to perform an epistemic uncertainty analysis would be unwarranted, given the objective of generating realistic or best-estimate severe accident source term values for regulatory applications.

An uncertainty analysis was performed in Reference [16] to address the aleatory (i.e., statistical) uncertainties. Therefore, the peer review committee believes that the uncertainties quantified as part of this study [16] are due to a variability in accident sequences and as such, they do not include uncertainties associated with incomplete knowledge of severe accident and radiological releases.

4.9 Other Issues

4.9.1 Multiple Source Terms

The Reference [16]-proposed source terms developed for PWR HBU UO_2 and MOX fuels do not differ markedly from one another when viewed within the context of the total variability known to exist in the calculational framework. Similarly, some peer review committee members believe that source terms developed by similar means for lower burnup fuel are also not markedly different when so viewed. The peer review committee discussed whether there is a need to maintain the current separation of source terms by reactor type and fuel type. With the exception of one committee member, the committee determined that it is appropriate to continue this differentiation.

4.9.2 Plutonium Release

After a review of the few available experimental data, the peer review committee generally believes that Reference [16] underestimates the release of plutonium. However, this deficiency does not significantly affect the results of the study given the fact that the objective is to characterize the in-containment source term and not the radiological consequences of releases into the environment. Therefore, it is considered acceptable to keep the plutonium releases (that are a part of the Ce Group) as they are, provided that the possible underestimation is explained in the final version of Reference [12].

4.9.3 Implications of the Fukushima Dai-ichi Accidents

The Fukushima Dai-ichi accidents occurred after the SNL work on the proposed source terms had been completed and the peer review work already underway. During the preparation of this report, NRC hosted a meeting to discuss Fukushima accident issues, as part of the Cooperative Severe Accident Research Program (CSARP) [25]. Some members of the committee attended the CSARP meeting, and all committee members received copies of the materials presented at the CSARP meeting for their consideration.

Before addressing the CSARP material, it is useful to recall that the purpose of the study being reviewed by this committee is limited to releases into containment, and it does not address release to the environment. The Fukushima Dai-ichi accidents are dominated by an extended SBO. A non-terminated SBO is considered one of the significant contributors to the source term in Reference [16], but not to the exclusion of other potential core damage events.

Ultimately, Fukushima Dai-ichi will provide valuable data once the sequence of events, release paths, and measurements are validated and analyzed. Until such a time, the radionuclide release information from the Fukushima Dai-ichi accidents is preliminary and incomplete. Since multiple units were involved and the damage to each unit is still unclear, it is difficult to attribute any detected releases as confirming, being inconclusive regarding, or disproving the proposed source terms in Reference [16].

The current belief is that releases are dominated by volatile radionuclides (e.g., isotopes of I, Cs, and Te). The actinides that have been measured in the vicinity of the reactor do not appear

to exceed the deposits that existed prior to the accident (primarily from fallout of the atmospheric nuclear weapons tests). Substantial attenuation of the release into the environment is attributed to the retention of fission products in the pressure suppression pools, which is consistent with predictions from Reference [16]. There is general agreement among the peer review committee members that the iodine releases in the proposed source term [16] are somewhat uncertain, and the preliminary Fukushima Dai-ichi information also supports this finding.

The recommendation is to continue to evaluate the implications of the Fukushima Dai-ichi accidents on the accident source term after more information and data become available. The committee recognizes that this will take time and considerable effort.

4.10 Documentation

The quality and transparency of the documentation of the source term work is crucial to its acceptance and credibility. Generally, the documentation reviewed by the peer review committee was adequate and complete. However, there are several areas where the committee recommends that the documentation should be improved, including:

- Inconsistencies between the plots in Section 6 and the tables in Section 5 of Reference [16] from which the plots were taken.

- Difficulties in reconciliation of data in Appendix C of Reference [21] with the tables in Section 5 of Reference [16].

- A clearer identification of sequences for the Surry plant analyses.

- The inclusion of plots in Appendix D of Reference [21] for both low-burnup and HBU calculations.

- The addition of Appendix D-type data for more sequences to Reference [22]. Similarly, including figures that correspond to Figures 20 and 21 in Reference [22] for all sequences.

- There should be revisions to correct the Reference [21] source term values, since SNL has indicated that they are incorrect and out of date.

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5. CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

The following summarize the findings of the peer review committee:

- The changes between the NUREG–1465 [6] source terms and those reported in Reference [12] result predominantly from advances in modeling and not from differences introduced by using MOX fuel or extensions to the burnup range.

- The code, models, and sequences used to generate the proposed source terms are appropriate.

- The proposed source terms are more realistic than those provided in NUREG–1465 in terms of the timing, chemical forms, and release fractions; plutonium release fractions may be underestimated, but do not significantly impact the final source term.

- The proposed source terms are representative or typical, rather than conservative or bounding.

- The representation of variabilities in release quantities and other characteristics associated with the proposed source terms are reasonable. However, there are significant uncertainties in release quantities and species for iodine, which are currently being investigated.

- The technical basis of Reference [16] is adequate.

- The gap release should be eliminated from the source term recommendation.

- Even though there is a slight increase in the release of I, Cs, Te, and Mo, there is a marked and important increase in the timing of the release in the proposed source terms compared to the previous estimates in NUREG–1465 [6].

- There is a sufficient increase in understanding source term behavior that it is appropriate to revise the source terms used by applicants and regulators.

- The revised source terms are adequate for determining containment equipment qualification parameters and for performing siting calculations to support siting decisions.
- The proposed source term satisfies the regulatory requirements of RG 1.183 [7] pertaining to attributes of an acceptable alternative accident source term.

Even though the accident source terms in the reference report provide challenging source terms for mitigation system performance and containment design for DBAs, given (1) the Fukushima accident, which exceeded the design bases of these plants, and where the collateral damage effects go beyond dose to the public (e.g., hydrogen explosion, worker dose, land contamination); and (2) the public's low tolerance for accidents that release radiation, the question arises as to whether existing safety goal and related guidance are specific enough with

respect to source term and other severe accident matters, such as clean-up and land contamination. Although outside the scope of the peer review committee's evaluation of accident source terms, this is an area that deserves consideration in the future, perhaps as part of the work on beyond DBAs.

5.2 Recommendations

The peer review committee recognizes the enormous effort that was required to produce the revised source terms [16]. Furthermore, the committee recognizes that in order to proceed forward with regulatory changes, certain activities are required. Mindful of that, the peer review committee offers the following recommendations:

1. Modify the revised source terms in Reference [16] to:

   (a) Eliminate the gap release phase.

   (b) Consider the peer review committee comments on under-prediction of the plutonium release, and if warranted modify such releases.

   (c) Consider the preliminary information available from Fukushima Dai-ichi and state whether there are any clear indications as to whether that data confirms or contradicts any assumptions or results in the proposed accident source terms or methodology [16].

2. Issue a separate report that provides:

   (a) An expanded discussion of iodine chemical forms to include a summary of recent experimental and analytical work,

   (b) A more complete explanation of the source term for applications where late releases are important,

   (c) Additional information provided during the meetings and during the technical exchanges (for example graphs of the different distribution functions),

   (d) Additional information about the progression of fuel degradation as calculated with MELCOR for the different accident sequences, and

   (e) Tables indicating the amount of molten materials/debris at the time that the reactor pressure vessel lower head fails.

   (f) Additional details regarding the adjustments made to the Booth models.

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    b. Randall Gauntt, " Overview of SNL Activities and Response to Fukushima Supported by USDOE," SNL.

    c. Masao Ogino, "Overview of Accident Analysis and Findings of Fukushima Dai-ichi Nuclear Power Station."

d. Martin Sonnenkalb, "Fukushima Accident – GRS Crisis Team, Lessons Learned and Future Activities," GRS.

e. Bernd Jäcke, "Spent Fuel Pool Behaviour under Severe Accident Conditions," Paul Sherrer Institute.

f. Vola, "IRSN response to the Fukushima situation and the consequences on IRSN future research activities," IRSN.

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APPENDIX A: IMPACT OF NUREG–1465 ON REGULATORY GUIDE 1.183

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On February 2 and 3 an expert panel met to discuss a revision to NUREG–1465. The revision will provide a source term for high burnup and mixed oxide fuels. One of the tasks of the committee members is to provide a recommendation on the suitability of the method to calculate the gap release. The panel stated that while the onset of the gap release would remain the same, the concept of a gap release was impacted by new modeling of the reactor core. With this modeling, the gap release phase and the in-vessel phases overlap and are not easily separable. The panel questioned which accidents would be impacted by not providing a gap release phase and combining it with the in-vessel release phase. Secondly, the committee members requested information on how NUREG–1465 was used to inform Regulatory Guide (RG) 1.183.

A.1 Use of the Gap Release Phase

The following discussion provides an assessment of not including the gap release information in NUREG–1465 for both the LOCA and non-LOCA accidents.

For LOCA accidents there is a potential impact for containment designs that have release pathways that isolate during the new gap/in-vessel release phase. The containment purge system might provide such a pathway. Because of these pathways, the timing of the onset of the gap release and the magnitude of the release during these phases impacts the dose analysis. The timing for isolation of these pathways currently does not extend beyond the current NUREG–1465 gap release duration and the release magnitude is a fraction of the total source term. It is therefore, expected that combining the gap and in-vessel phase would have a conservative, but small impact on the LOCA dose analyses
and, therefore, would be acceptable.

For non-LOCA design basis accidents (DBAs), where the cladding is postulated to be breached but no fuel melt is postulated, NUREG–1465 gap release fractions are not used unless the event would involve the entire core. Typically, the non-LOCA accidents do not involve the entire core (like a full core meltdown assumed for the LOCA) but involve only these fuel pins that exceed DNB. For non-LOCA DBAs, the timing of the gap release to the RCS and the containment is conservatively assumed to be instantaneous. Therefore, RG 1.183 does not rely upon NUREG–1465 gap release fractions or the timing of the releases for these accidents.

A.2 Other Uses of NUREG–1465 for Non-LOCA DBAs

Because of the caveats in NUREG–1465 regarding its applicability to existing plants and to fuel burnups in excess of 40 GWd/MTU, the staff only used selected information from NUREG–1465, the majority of which, was associated with the isotopic mix and release fractions for accidents involving fuel melt.

In Regulatory Guide 1.183, the NRC assumes the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident is consistent with the speciation assumed in NUREG–1465 (95% CsI as an aerosol, 4.85% elemental and 0.15% organic). If fuel melt is postulated, RG 1.183 uses RG 1.77 release fractions for the two accidents where fuel melt is typically considered. These accidents are the boiling water reactor (BWR) control rod drop accident (CRDA) and the pressurized water reactor (PWR) rod ejection accident (REA). Releases from fuel, due to melt, are assumed to be instantaneous and, therefore, are not dependent on NUREG–1465. Table A.1 below provides more details on the use of NUREG–1465 in Regulatory Guide 1.183 for non-LOCA DBAs. Table A.1 indicates that in practice the NUREG–1465 speciations assumed will only have an impact on the dose calculations for the BWR CRDA and the PWR REA.

Table A.1 Evaluation of the Use of NUREG–1465 in Regulatory Guide 1.183 Dose Analysis (non-LOCA DBAs only)

App.	Accident	Speciation of release to containment atmosphere same as NUREG–1465	Magnitude of fractional release to containment atmosphere same as NUREG–1465	Speciation of release to RCS same as NUREG–1465	Magnitude of fractional release to RCS same as NUREG–1465[5, 6]	Release pathway to Env. (other than containment)	Speciation of release pathway to Env. same as NUREG–1465[7]	Magnitude of fractional release pathway to Env. function of NUREG–1465	Do NUREG–1465 assumptions impact the dose results?
B	Fuel Handling	No	No	No	No	SFP/SFP Building	No	No	No[8]
C	BWR CRDA	N/A	N/A	Yes	No	Condenser	No	No	No
D	BWR MSLB	Yes[9]	No[10]	Yes, see note 5.	No. See note 6.	Steamline	Yes	No	Yes[11]
E	PWR MSLB	N/A	N/A	Yes	No	Steam Generator	No	No	No[12]
F	PWR SGTR	N/A	N/A	Yes	No	Steam Generator	No	No	No
G	PWR Locked Rotor	N/A	N/A	Yes	No	Steam Generator	No	No	No
H	PWR Rod Ejection	Yes	No	Yes	No	Steam Generator	No	No	Yes[7]

Key -
App. – Regulatory Guide 1.183 Appendix
SFP – spent fuel pool
SGTR – steam generator tube rupture
RCS – reactor coolant system
MSLB – main steam line break
CRDA – control rod drop accident
Env. – environment
BWR – boiling water reactor
RG – Regulatory Guide
PWR – pressurized water reactor

[5] For the BWR MSLB and PWR SGTR typically no fuel failure is postulated, but guidance is provided for situations where failure may occur.
[6] For the BWR CRDA and PWR Rod Ejection, RG 1.183 assumed the same release fractions as given in RG 1.77 as directed by Office of Nuclear Regulatory Research (RES). These release fractions are different than NUREG–1465.
[7] For releases from the steam generator and condenser the releases to the environment are assumed to be 97% elemental and 3% organic.
[8] Speciation from NUREG–1465 is given in RG 1.183 for the release from the fuel to the SFP, but there is no impact to the dose calculations.
[9] The release to the containment is via the RCS/MSL.
[10] Fuel melt (FM) is not considered for this accident.
[11] Yes. The impact is only due to the NUREG–1465 speciation assumed.
[12] No. Some plants (typically Combustion Engineering plants) assume FM, but the release fractions used are taken from RG 1.77.

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APPENDIX B: COMMITTEE QUESTIONS AND SNL RESPONSES

Following their review of the various documents and the presentations during the first meeting, the committee members submitted a series of questions, comments, and Requests for Additional Information (RAIs). Subsequently, SNL provided responses to these questions and supplied additional information to the committee. During the second meeting, the committee members presented their evaluation of these SNL responses and additional clarifications were made by SNL. Finally, following electronic communication and telephone conference, outstanding issues were discussed and the committee members submitted, in writing, their final evaluation of the SNL responses to various questions, comments and RAIs. This appendix documents the questions, comments, and related RAIs, the SNL responses, and the evaluation of SNL responses by each committee member.

B.1 Dr. Bernard Clément

Question 1 - The documentation is of good quality as well as the presentations made during the first meeting of the review committee and the technical exchanges that followed. Some additional information was provided during the meeting and it would be worthwhile that part of it be included in the final report (see other comments). Note also that the document on BWR calculations that was transmitted to the committee is a 2006 draft and not the SAND2007-7697 final version.

Sandia Response

Our objective is to keep the source term document terse – akin to NUREG–1465 – while being clear on how the source term was generated. It may be necessary to issue an additional support document that would contain additional support information. The final version will be sent.

Evaluation

OK.

Question 2 - Important differences among the accident source terms derived here and the NUREG–1465 source term are not attributable to either fuel burnup or use of MOX fuel. Rather, differences among the source terms are due predominantly to improved understanding of the physics of core meltdown accidents" (from the abstract). This statement is consistent with the conclusions of the expert panel convened in 2001-2002. The fact that this conclusion is now supported by state-of-the-art calculations and not only by expert judgment is an important progress.

Sandia Response

Agreed.

Evaluation

OK.

Question 3 - The selected accident sequences are intended to be representative for the class of accidents that involve substantial core meltdown. It would be interesting for the reader to have some figures showing how much the core meltdown is substantial for the selected sequences. This could be done by giving in a summary table the amount of molten fuel (or of debris in the lower head) at the time of vessel failure, and the amount of molten fuel at the end of the ex-vessel phase. Note that some information is given for Catawba calculations about the amount of debris relocated in the lower head of the pressure vessel before its failure.

Sandia Response

We would prefer to keep the document succinct – akin to NUREG–1465. Such information is available from MELCOR calculations. In general, eventually the entire core melts for these uninterrupted sequences. The amount on the lower pressure vessel head at the time of head penetration is variable. Such additional information would have to be reported in some, additional, support document.

Evaluation

OK.

Question 4 - It is said (p.22/23 of source term report) that together the accidents consider 92% of core damage frequency for Peach Bottom, 98% for Grand Gulf, 85% for Surry and 88% for Sequoyah. As the selection does not include mitigated or arrested sequences, contributing to CDF but leading to less substantial core meltdown, I suppose that the numbers correspond to families of accidents (SBO, ATWS…) and not to individual accident sequences as this can be inferred from slides #35 to 37 of the draft presentation on mechanics. Note also that the values given in the slides and the report are not exactly the same.

Sandia Response

The interpretation is essentially correct. Accidents were taken from major families including two types of transients, various sizes of loss of coolant accidents, station blackout accidents, and anticipated accidents without scram. Specific sequences were defined in each class – typically ones highlighted in the Individual Plant Examination (IPE) prepared by the licensee for the particular plants considered. The IPE only reports sequences that lead to core melting (typically >5% of the core). The analyses did not extend to core meltdown progression and the ability to arrest an accident. The coverage of possible accident sequences obtained in the accident selection procedure depends somewhat on how finely one categorizes sequences. The coverage reported in the presentation is the highest level of breakdown of the sequences as they were reported in the IPE insights document – NUREG–1560.

Evaluation

OK.

Question 5 - The graphs entitled “IPE insights” or the CDF tables of the draft presentation on mechanics could be usefully added to the main document.

Sandia Response

It would be preferred to not clutter the document with ancillary information. The document does not purport to be a discussion of risk and the information in the plots is readily available in the IPE insights document. Perhaps it is adequate to note the points in the text and refer the reader to the IPE insights report.

Evaluation

OK.

Question 6 - Non parametric order statistics are used, considering the results of calculations as samples from an unknown distribution. Instead of simply using the empirical distributions, cumulative distribution functions were developed and refined using a bootstrap re-sampling method. This would only be fully correct for random samples. The method is however considered as acceptable given the relative weight of sequences derived from IPEs (see also Clément comment # 5).

Sandia Response

Agreed.

Evaluation

OK.

Question 7 - The median value of the distribution has been selected as being the "representative" value for release phases duration and released amounts during each phase. It is true that using the median value provides more robust results as compared with using the mean value or high or low percentile values (e.g., 5% or 95%). This is understandable and acceptable on a statistical point of view. Whether or not this value is depends on which use will be done of it.

Sandia Response

Agreed.

Evaluation

OK.

Question 8 - The main question, and this is not a statistical issue, is to define what is a "representative" value for the class of accidents considered. It should be made clear that the choice of accident sequences involving a substantial core meltdown induces some "conservatism" (although not being bounding), whereas the state-of-the art nature of the calculations and the statistical treatment point to "best-estimate" analyses. The reviewer has no problem with such a mix provided it is more clearly explained in the report.

Sandia Response

Modifications to the text will be made to clarify. Selection of sequences involving substantial core meltdown is dictated by regulation. Selection of sequences without operator intervention assures that the source term from the selected sequence will not be exceeded by a credible sequence that would involve operator measures to mitigate release.

Evaluation

OK.

Question 9 - The standard deviation on the median location is also given as an aleatory uncertainty. It should better be said that this standard deviation gives an indication of the variability within the selected sequences. A special caution is needed when using this standard deviation for comparison with values quoted in NUREG–1465 which were not resulting from the same statistical treatment.

Sandia Response

The standard deviations reported in the document indicate the inability to accurately locate the true median of the distribution based on a finite set of results. The variability due to accident type is reflected by the full distribution of each result which is available for most results from the support document. It is clear that standard deviations reported as they are now are a source of confusion. They are needed for making critical comparisons such as comparison of results for LEU and MOX. They also serve to show where differences between current results and those obtained in NUREG–1465 are significant. It is, however, clear that a better method of exposition is needed. Simple modifications or additions to the text will be insufficient. The table themselves need to be modified or augmented. Furthermore, it is planned that the document will be amended to eliminate reference to aleatory uncertainty in favor of accident variability.

Evaluation

OK.

Question 10 - A number of graphs of the distribution functions have been presented at the first meeting of the review committee. It could be useful that such graphs are appended to the main report. Some readers might want to see how the distributions look like and examine what are the values for a given percentile (e.g., 80% that is easily readable on the graphs).

Sandia Response

There would be at least 72 graphs which would more than double the length of the document. Appending the distributions would be antithetical to the general aspiration that the document be succinct. A compromise might be to include the graphs and a discussion of the graphs in an additional support document available to those with an interest in the details. (There has not been a great deal of interest in the support documents for NUREG–1465.) It should be noted that recently plotting of the distributions has been upgraded so that more realistic confidence

intervals can be shown for the extremes of the distributions (>75% and < 25%). The upgraded confidence intervals are based on the Dvoretzky-Kiefer-Wolfwitz inequality.

Evaluation

OK.

Question 11 - As explained before, what is called “aleatory uncertainty” in the report is more a variability than an uncertainty (see Clément Question 9). It is also said that the treatment of epistemic uncertainties are out of the scope of this study. Is this because the amount of work required would be too large, or because it was not considered as being very important?

Sandia Response

Neither. What was sought by the regulatory process was how source terms might change given the current state of the art and the emerging practices within the industry to use fuel to higher burnups at all reactors and to use MOX at selected reactors. Just as NUREG–1465 was a ‘snapshot’ in time of the state of the art, this too is to be a ‘snapshot’ in time of the state of the art. Many of the epistemic uncertainties that afflicted results obtained in support of the NUREG–1465 source term have been resolved by research. Many uncertainties unknown at the time NUREG–1465 was developed have been identified and resolved or are being researched now. Clearly, resource commitments being made to these research efforts are indicative of the importance attached to epistemic uncertainties. Resolution of these and other issues now being researched will lead to an evolution of the state of the art that will have some impact on the projected source term in the future. For now, there is an immediate need by the regulatory process to have some confirmation given the current state of the art that decisions being made to evolving practices provide adequate protection of the public health and safety as prescribed by the Atomic Energy Act.

Evaluation

OK.

Question 12 - As said in the report, there is no phenomenological boundary to mark the end of the gap release phase. The criterion of 5% of the initial total core inventory is acceptable.

Sandia Response

Agreed.

Evaluation

OK.

Question 13 - During the discussions that took place during the review committee meeting, it appeared that the NUREG–1465 gap releases might be used by some people for a different use than originally foreseen. It should be made clear somewhere in the report that the values given

correspond to releases to the containment during the gap release phase of an accident involving a substantial core meltdown. They are not necessarily applicable to other purposes.

Sandia Response

The report will be so amended.

Evaluation

OK.

Question 14 - It is said on page 27 of the main report that most, though not all, of the plutonium dioxide is atomically dispersed in the uranium dioxide matrix. This does not seem to be consistent with the presence of plutonium-rich aggregates in MOX fuel.

Sandia Response

The French micronized process is used to achieve a very high, but not complete, level of atomic dispersion of plutonium in the uranium dioxide lattice. Details of the process and dispersion were extensively discussed in the preparation of the safety analysis report for the lead test assemblies for MOX in the Catawba reactor. Source term models currently available in MELCOR do not treat the release of radionuclides in a way that necessitates the differentiation between atomic dispersion and more concentrated islands enriched in plutonium. The models are keyed to empirical release data. It is unknown whether incomplete dispersion will affect core degradation. As noted in the report, data on the issue of degradation of MOX under severe accident conditions are not now available.

Evaluation

OK.

Question 15 - No modifications were made in MELCOR modeling to account for differences in the core degradation as compared with LEU fuel. Differences are expected probably an acceleration of fuel degradation. However it is true that there is a lack of experimental data to support modifications in modeling. In addition, as the sequences correspond to a substantial meltdown of the core, the final results of the calculations should not be much affected.

Sandia Response

Agreed. It is also to be noted that less than 40% of the fuel assemblies in a real core will have MOX. Presence of large amounts of LEU may mitigate core-wide propagation of any accelerated degradation experienced by a MOX fuel assembly.

Evaluation

OK.

Question 16 - Improvements have been made in MELCOR modeling to better take into account the outcomes of several experimental programs such as VEGA, VERCORS and Phébus PF. It is thought that this allows one to derive more reliable values for the releases of cesium, molybdenum and tellurium in particular.

Sandia Response

Agreed. It is noted, however, that modifications to account for Knudsen diffusion in the transport of vapors through the fuel pore network as identified in VEGA have not yet been made in MELCOR.

Evaluation

OK.

Question 17 - For MOX fuel, the experimental database is scarce. The parameters of the model have been fitted on VERCORS RT2 experiment rather than VERCORS RT7. As long as MELCOR models do not take into account the oxidizing character of the atmosphere surrounding the fuel, this is considered as acceptable (as RT7 experiments w[ere] performed in pure hydrogen).

Sandia Response

It is thought that the open lattice core of a PWR that allows for radial flows makes it quite difficult to sustain pure hydrogen conditions over volumes of the core as large as a MELCOR node size. RT2 test data were thought to be more realistic than data obtained under pure hydrogen conditions.

Evaluation

OK.

Question 18 - For high burnup LEU fuel, the same procedure has been used from the data of VERCORS RT6 experiment with the same limitation that MELCOR models do not take into account the oxidizing character of the atmosphere surrounding the fuel.

Sandia Response

Again, open lattice geometry of PWR fuel prevents extremes of oxygen potential because of radial cross flows. Vapor speciation in MELCOR is selected to be consistent with available release data and expectations based on thermo-chemical modeling. Much of the detail of possible chemical forms has been derived from analyses using the VICTORIA model as well as experimental findings.

Evaluation

OK.

Question 19 - Some insights are given about the increased release of ruthenium when fuel is exposed to air, including graphs of ruthenium oxides vapour pressure and also for cladding oxidation. The understanding of the reviewer is that the ruthenium release under air ingress conditions after pressure vessel failure is not accounted for in the present study. A complete treatment would indeed require (i) a calculation of air+steam+other gases flow entering the breached reactor vessel, (ii) an estimate of the impact on fuel degradation through the oxidation of claddings, (iii) modeling of fuel oxidation, (iv) calculation of the release of volatile ruthenium oxides from the fuel, (v) calculation of the transport of those species to the containment through the RCS. Are there some plans to address these issues in the future?

Sandia Response

The issues of air ingress are topics of ongoing research. NRC hopes to derive further insight into the issues from planned tests in France in the Verdon project. Analyses of the issue undertaken by NRC in support of the Phébus-FP project showed that details of core degradation and especially the separation between fuel and residual metallic cladding were critical. Unfortunately, uncertainty in large scale core degradation made it possible within the current state of understanding to adopt coding options that promoted or retarded ruthenium release. Clearly, more experimental data will be needed to resolve the issue. The resolution is beyond the current state-of-the-art and outside the scope of this study.

Evaluation

OK.

Question 20 - Keeping the NUREG–1465 prescription of 95% of iodine released as metal-iodide particulate and 5 % as gaseous species is reasonable at the time being.

Sandia Response

Agreed.

Evaluation

OK.

Question 21 - Some more details are given in NUREG–1465 about gaseous iodine speciation. In §4.5, it is stated that iodine entering containment from the reactor coolant system is composed of at least 95% cesium iodide, with no more than 5% iodine plus hydrogen iodide. Is this still considered as valid? Did the end-users of NUREG–1465 take into consideration the distinction between iodine and hydrogen iodide?

Sandia Response

Note that for regulatory purposes, NUREG–1465 is not used directly. Rather, Regulatory Guide 1.183 is used which has more specific prescriptions for the chemical form of iodine and its

behavior under accident conditions. In light of results from research since the time NUREG–1465 was developed, it is preferable to denote the particulate form of iodine as metal iodide rather than specifically cesium iodide. Similarly, it is preferable to denote gaseous iodide generically rather than identify specifically hydrogen iodide and atomic iodine. At modestly elevated temperatures there is interconversion among iodine, hydrogen iodide, and free iodine with relatively well known kinetics under laboratory conditions. Other processes are important under accident conditions where there can be reactions to form a variety of gaseous species such as hydrogen iodide, iodide oxide and even higher oxides. In addition, the atmosphere is expected to be contaminated with a variety of organic vapor species that could lead to the transient formation of volatile organic iodides. Substantial improvements in our understanding of iodine chemistry in reactor containments are expected to come from the BIP and STEM programs now planned or underway. In the interim, it is proposed that regulators continue to adopt the prescription they have used in the past.

Evaluation

OK.

Question 22 - There is also some consideration of organic iodides in NUREG–1465. Although this is not a prescription, one can read in §3.5 about the conversion from inorganic to organic forms of iodine that "in view of the results of Ref. 23 that a conversion of 3.2 percent is unduly conservative, a value of 3 percent is considered more realistic and will be used in this report. Where the pH is controlled at values of 7 or greater within the containment, elemental iodine can be taken as comprising no more than 5 percent of the total iodine released, and iodine in organic form may be taken as comprising no greater than 0.15 percent (3 percent of 5 percent) of the total iodine released." Are these figures still considered as valid and, if so, should not they be recalled in the new paper?

Sandia Response

The issue of the conversion of molecular iodine to volatile organic iodide is now undergoing wholesale revision in directions quite different than envisaged at the time NUREG–1465 was developed. This revision is not yet complete. It is still thought prudent to consider that both particulate and gaseous iodine will be present in containment atmospheres following accidents and that some fraction of the gaseous iodine will be present as a volatile organic species. There is not yet an adequate basis for modifying regulatory guidance in this area. It is hoped that such a basis will be possible as research efforts in the BIP, Phébus and STEM programs reach completion.

Evaluation

OK.

Question 23 - Are there any plans to revise the chemical form of iodine (amount and speciation) in the future according to progress in knowledge?

Sandia Response

Revision of the treatment of the chemical form of iodine is a priority research topic at NRC. A strategy for this revision has been developed and reviewed by the ACRS. NRC has invested in the Phébus–FP and –ISTP projects and is investing in both the BIP and STEM experimental research projects. NRC will revise its mechanistic modeling in a standalone computer model that will be validated against data from these and other programs. Analyses of results derived from the standalone model will be used to determine the depth and detail of modeling included in the MELCOR systems level computer code. Mechanistic modeling is thought necessary because of challenges associated with the extrapolation of results from tests to reactor accident conditions that are more complicated and diverse than experimental studies.

Evaluation

OK.

B.2 Dr. Richard S. Denning

There were five initial requests for additional information to support my assessment of the validity of the proposed modifications to the NUREG–1465 source terms. Two follow-up questions were subsequently provided to Sandia. In some cases, the Sandia replies were not fully responsive when it was perceived that the question was outside the established bounds for the peer review. The concept that the peer review should not question the validity of MELCOR models under the pretext that MELCOR is state of the art is untenable. It is inconsistent with the planned regulatory use of the revised source terms. How could one possibly perform a technical peer review of the revised source terms, which are directly based on MELCOR results, without making an assessment of how accurately MELCOR is able to predict source terms? Saying that a code is state of the art tells us nothing about how accurate it is. Consider the “state-of-the-art” capability to predict Emergency Core Cooling behavior in the early 1970s. It was woefully inadequate. To compensate, the NRC was forced to incorporate conservatism to account for poorly understood phenomena. Furthermore, it isn’t clear what state of the art means. Many of the models in MELCOR are far from state of the art (e.g. lack of CFD modeling). This is not a criticism of MELCOR. But to say that it is the best available code to perform integrated sequence analysis is not sufficient to conclude that the MELCOR results are sufficiently representative of reality to be used for regulatory purposes. Nevertheless, I feel that I was able to subsequently obtain adequate information to draw conclusions about whether the accuracy of the MELCOR source term analyses was sufficient to support the planned regulatory uses.

Question 1 - Although this is not a MELCOR validation exercise, we need to determine how well MELCOR is able to predict temperatures and release rates from the fuel.

Question 1a - Please provide a copy of the Strizov report NSI-SARR-149-03.

Sandia Response

Sure, once he who has the report gets back from Japan and can find it.

The report was subsequently provided showing the results of validation efforts for MELCOR performed by IBRAE.

Evaluation

The response was sufficient and helpful.

Question 1b - In the comparisons shown for the Phébus experiment, please show comparisons between the temperature predictions obtained in the MELCOR analyses with the measured values from the test. How well is MELCOR able to model the experiment? Often in comparisons of this type it is necessary to make assumptions about the boundary conditions to the experiment. Please indicate how these boundary conditions were applied. Were blind calculations performed for any experiments?

Sandia Response

This request is for work and results considerably outside the scope of the project being reviewed. There is a project looking at the ability of MELCOR to predict Phébus tests. It operates on its own schedule and with its own milestone. Results will be published in due time. For the work being reviewed by the experts it was assumed that MELCOR produced accident predictions that represented the current state of the art.

Evaluation

SNL's response was limited.

Question 1c - We need to consider the possibility that MELCOR could under-estimate time vs. temperature behavior. If you can, for some example scenarios, I would like to see time vs. temperature for fuel in the various phases of heatup, candling, slumping, and pool development. What fraction if any of the fuel achieves values in the range of melting of 3100K? What is the basis for selecting values of TMELT e.g., 2500 K vs. 2800 K?

Sandia Response

It is not evident why this information is needed. It can be provided for selected sequences, but would take some time and might involve re-running the cases to obtain the desired output.

Evaluation

SNL's response was limited. Apparently, SNL had not previously performed a detailed examination of time-temperature behavior of fuel degradation during the various phases of melt progression in order to determine the impact on radionuclide release. I agree that the additional analyses required would have required rerunning some cases. I was able to proceed without the information.

Question 1c.1 - Is the S/V correction term used in the release calculation? After pools have formed, does MELCOR continue to calculate the release of radionuclides?

Sandia Response

Yes. Much of this information is available in the RN section of the MELCOR user's manuals. It is assumed for the work under review that MELCOR provides state of the art predictions of radionuclide release and transport. This assumption is not considered a topic for review.

Evaluation

SNL's response was limited.

Question 1c.2 - Are there conditions of high zirconium oxidation in which some UO_2 isn't liquefied until around its melting point of 3100 K? Do experimental programs adequately address conditions of this type or are they terminated too early?

Sandia Response

Liquefaction of fuel is governed by phase relations in the Zr-U-O system. Phase boundaries in this system are all at temperatures less than 3100 K. Were it possible to completely oxidize the cladding, then liquefaction would be controlled by phase relations in the ZrO_2-UO_2 system which is a minimum melting point system. The phase boundaries are again at temperatures less than 3100 K. To achieve high superheats in core debris within the vessel, large masses with limited heat transfer to the boundaries must accumulate. This does happen in especially PWR accident analyses. But, the low surface to volume ratio that limits heat transfer also limits mass transfer and consequently the release of radionuclides. Experiments address realistic conditions of core degradation. High core debris temperatures can be generated during the Ex-vessel Release Phase of an accident when core debris containing large quantities of unoxidized zirconium interact with a concrete having a high melting point. Typically this is encountered in rapid BWR sequences. With prolonged in-vessel core degradation, there is more opportunity to oxidize the zirconium. Consequently, energetic core debris interactions with concrete are less commonly encountered even with BWRs than they were during the time NUREG–1465 was being developed.

Evaluation

SNL response was informative and sufficient.

Question 1c.3 - Have you done sensitivity studies for the release of involatile groups and cesium molybdate for cases in which the parameters that could lead to high fuel temperatures have been selected?

Sandia Response

No. Validation of MELCOR was not an objective of this work. It was assumed that MELCOR produced state-of-the-art predictions of radionuclide behavior and accident progression.

Evaluation

SNL was adequately responsive.

Question 1d - Please provide insights regarding validation of release from fuel during core concrete attack, particularly for the cerium group. In which cases has credit been given for an overlying pool of water in the reactor cavity? How substantial is the effect of pool scrubbing on the source term values?

Sandia Response

Validation of MELCOR and its various constituent codes is outside the scope of this work. Validation of the modeling of radionuclide release during core debris interactions with concrete was done quite some time ago. It is known that the VANESA module of MELCOR under predicts the releases of molybdenum and ruthenium during core debris interactions with concrete. This is because of an assumed and enforced partitioning of some radionuclides between the oxide and metallic phase of core debris. The assumption was based on older data and needs to be revised. Cerium releases tend to be very small during core debris concrete interactions except when there is a large mass of unoxidized zirconium in the core debris (Actually, because of phase relationships it is really a mass of metallic uranium and zirconium in the core debris.) During the transient period where gaseous and condensed products of concrete decomposition react with the electropositive metals in the core debris very high chemical heat generation rates can arise. A temperature excursion is usually predicted. Cerium is not especially volatile even at elevated temperatures because it is well solvated by the core debris. Mechanical generation of aerosols by bubble rupture can be an important mechanism of cerium releases. Release fractions are still quite small relative to other more volatile radionuclide.

Evaluation

SNL response was informative and helpful.

Question 2a - The accuracy with which we can predict radionuclide transport and deposition mechanisms from the core to containment is limited. How much credit is given to deposition in the pathway to containment for the different cases and how does that affect the values in the release tables?

Sandia Response

MELCOR models aerosol deposition and vapor condensation along transport pathways from the core to the containment. Details can be found in the MELCOR model documentation. Only radionuclides that successfully negotiate this pathway and reach containment are considered part of the release to the containment of interest for the work under review. It is deposition along the transport pathway to containment that makes release fractions for Alkali Metals so different than those of Halogens, as noted in the text of the document. Radionuclides that deposit in the pathway may subsequently revaporize and reach containment as part of the Late In-vessel Release. A detailed breakdown for deposition at various times along the pathway could be

developed but would require rerunning the calculational cases to get proper output. Again, the MELCOR treatment of accident progression including aerosol and vapor deposition along flow pathways through the reactor coolant system is treated for the purposes of this work as being state of the art.

Evaluation

There was additional discussion of this topic at the meeting in DC that convinced me that there was not inappropriate credit given to retention during transport within the RCS.

Question 2b - How significant is the effect of hygroscopicity for in-vessel transport on the deposition results?

Sandia Response

Hygroscopicity is not a consideration at the temperatures and pressures of the reactor coolant system during an accident in which coolant is boiling off. It is proving not to be a major issue even in the containment. Calculations of hygroscopic effects are difficult to do with great accuracy because aerosol particles are agglomerates of materials of varying hygroscopic character rather than pure materials with well known water solubilities and, consequently, hygroscopicity. Calculations done of the aerosol concentration in containment during Phébus tests (which typically had a relative humidity of about 60%) showed that it did not really matter whether the hygroscopicity model used in MELCOR was active or not. Results in good agreement with data were achieved until concentrations of aerosol fell below about 10^{-4} g/m^3.

Evaluation

The SNL response was informative and sufficient.

Question 2c - Re-evolution of radionuclides deposited within the vessel is also a difficult area to model. We should receive a more detailed presentation on the validity of MELCOR models for this phase of the accident."

Sandia Response

MELCOR modeling of revaporization is considered for this work to be state of the art. It is, however, a simple model. It is basically evaporation controlled by temperature and mass transport from heat structures. Revaporization of deposited radionuclides is an ongoing area of research and it is hoped that additional information will be obtained from the STEM project being planned now in France. Though presentations of MELCOR modeling are possible, MELCOR validity is the not the subject of the review. Documentation of MELCOR may provide a better vehicle for understanding the capabilities of the computer code.

Evaluation

More detailed discussion of this topic occurred at the meeting in Washington. It resolved questions that I had regarding the amount of credit given to deposition during transport to the containment.

Question 3a - The characterization of the containment source term does not address the size distribution of aerosols entering the containment

Sandia Response

The work under review followed the practice established by NUREG–1465 and did not provide estimates of the size distribution for aerosol entering the containment. There is growing confidence in the ability of MELCOR to predict size distributions based on comparisons to the results of the Phébus tests. Nevertheless, the size distributions is likely to be a function of plant details and is left to analyst to develop and defend according to the details of his plant and especially the availability of volatile, nonradioactive materials. Licensees have used both MELCOR and MAAP to predict the particle size distribution and the releases of nonradioactive aerosol mass.

Evaluation

We also discussed this issue at a meeting in Washington. Although the size distribution entering containment is an important aspect of the source term, historically it was not included in the NUREG–1465 document. Typically, the NRC requires some conservatism to the treatment of containment processes and assumptions regarding aerosol size distribution. The response is sufficient.

Question 3b - Is there evidence that the concentration of radioactive and non-radioactive aerosols is sufficiently high that the incoming size distribution becomes irrelevant or should the new source term also characterize size distribution?

Sandia Response

To the contrary, there is evidence that the size distribution of aerosol input to the containment can affect predictions of aerosol behavior in the containment. As implied by the question, the size distribution will depend on the nonradioactive as well as the radioactive aerosol mass. The nonradioactive aerosol will depend on plant design details. It will be, for example, quite different for plants using Zircaloy cladding than it is for plants using M5 cladding. The NUREG–1465 source term did not provide estimates for either the aerosol size distribution or the non-radioactive aerosol mass. The work under review followed this pattern as part of the admonishment not to deviate unnecessarily from the established NUREG–1465 source term. In fact, the predictions of aerosol size distribution and the ability to predict aerosol behavior in the containment as well as in the reactor coolant system are area of particularly intensive validation studies including comparisons to Phébus-FP test results.

Evaluation

The response is adequate.

Question 4 - The gap release is not well defined in the proposed new source term and is characterized by a time period of the accident rather than the conditions that lead to release from an individual fuel pin. I suggest that you consider the possibility of defining a gap release for the new source term that only addresses the needs of regulatory activities that address pin cladding failures but not gross core meltdown. Thus, the gap release could address only the operating condition inventory of the gap. This should depend on burnup. With regards to the gross meltdown applications (e.g., site dose calculations and EQ calculations), the gap inventory would be incorporated into the first phase of in-vessel release.

Sandia Response

No decision has been made to eliminate the Gap Release Phase. Opinions of experts on this topic are solicited. Should a decision be made to eliminate the Gap Release Phase, there would be, no doubt, additional deliberations on how to fulfill any information needs for the regulatory process. The gap release is more than the gap inventory. NRC has other models (FRAPTRAN and FRAP) better suited for calculation of gap inventories and more for normal operations.

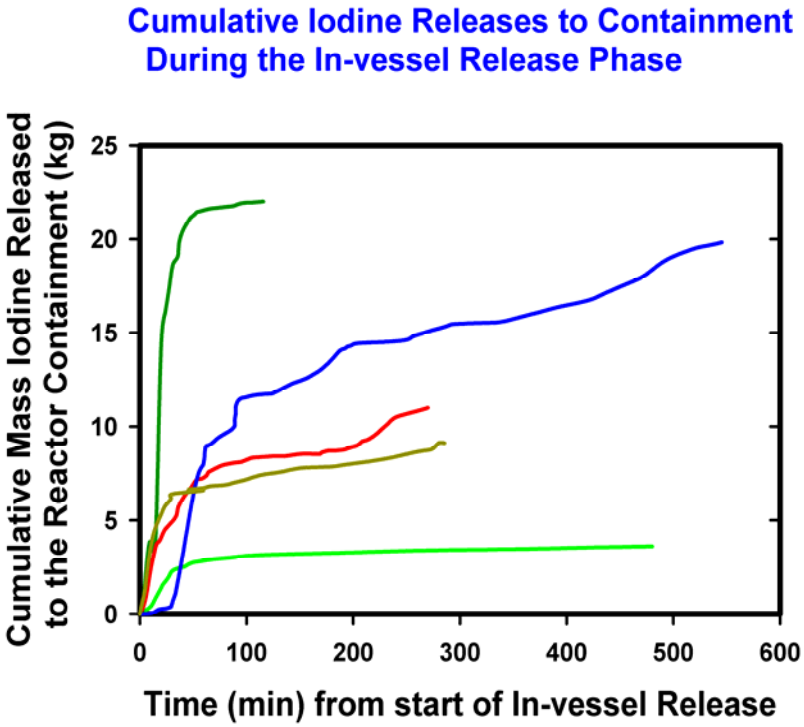
Evaluation

Response is sufficient.

Question 5 - There may be justification for dividing the initial in-vessel release phase into two phases. We should receive a presentation on a proposed division into two phases with a basis provided. Provide insights into how that could affect such calculations as the exclusion area and low population zone doses.

Sandia Response

No decision has been made to divide in-vessel release into two stages. Expert opinions on the wisdom of retaining the assumption of a single phase with a constant release rate are solicited. At this juncture, it appears that in-vessel release could be divided into two phases each with a different but still constant release rate. No decision has been made to develop this alternative. Calculations to assess the impact of such a division of the in-vessel release phase on consequences would be a considerable expansion of the scope of this work and might take some time to accomplish. A crucial issue that will arise in dividing in-vessel release into two stages will be the criterion for ending the first and starting the second phase. Quite a range of behaviors was observed in calculations (See following slide showing single and multiple plateau and varying rates of initial and later release.) A clear-cut phenomenological basis for division of in-vessel release into two stages has not been identified. Consequently, whatever is selected will be viewed, in all likelihood, as arbitrary by some. Experts may want to offer their suggestions on a criterion if they feel it essential to segment in-vessel release into two phases.



Evaluation

At this point, the peer review team doesn't have sufficient information to make a recommendation about the value or need for dividing the release into phases.

Following the second peer review meeting in Washington, some additional requests for information were submitted.

In reviewing the differences between the NUREG–1465 source terms and the proposed updated source terms, it is apparent for the in-vessel release that the largest differences are in the cerium and lanthanum groups where the proposed reduction is a factor of 1,000. Although I recognize that the appropriate focus of our review should be on the volatile groups, in particular iodine and cesium that tend to dominate the offsite dose effects, when considering potential differences associated with high burnup or mixed oxide fuels, the cerium group (which contains plutonium) and lanthanum group are also of high interest. Although it is true that the in-vessel releases of these groups is dominated by the ex-vessel release in a full core meltdown accident, the principal use of the revised source terms will only use the early in-vessel release component. In order to gain greater confidence that there really is justification for such a substantial change, I would like SNL to address some additional questions -

Question 1 - In the SAND2010-1635 document, based on Phébus FPT-1 results, a substantial scaling factor was applied to the Booth model release rate for UO_2. What scaling factor was used? In this document, it indicates that no modification was made to the Ce and La release parameters because the test did not measure releases for these groups. However, it then goes

on to say that "it could be reasoned that their releases ought to roughly follow UO_2 release if fuel matrix stripping occurs."

Sandia Response

The baseline Booth parameters were taken from the ORNL Booth parameters. The Ce and La parameters were taken from ORNL Booth. The U release was scaled up to conform to the observations from the integral FPT-1 test. The remark concerning matrix stripping should have been associated with highly oxidizing conditions such as with air oxidation. In this case the low volatiles are expected to follow the U release, which in this highly oxidizing regime, should be dominated by the more physical effects of grain stripping of the UO2.

Evaluation

The SNL response was informative and sufficient.

Question 2 - In Table 5 of the same document, for experiment VI-5, the release of Ce is indicated as 2%, far higher than the CORSOR-M and ORNL-Booth estimates for that experiment. Is there some aspect of the experiment that is far from prototypic?

Sandia Response

Regarding the Ce release cited for ORNL VI-5, I suspect that the reported release of 0.02 is a typographical error. The VI-5 test was done with input of hydrogen and no steam so it is not prototypical of reactor accidents.

Evaluation

The SNL response was informative. However, based on a review of NUREG/CR–6261, I believe that the value of 0.02 is not a typographical error.

Question 3 - In Dr. Hidaka's paper, "Outcome of VEGA Program on Radionuclide Release from Irradiated Fuel under Severe Accident Conditions," Figure 9 provides experimental values for Release Rate as a function of temperature. At the melting point of zirconium (2,128K) at which UO_2 solution could be initiated, the release rate of plutonium is 1E-4%/min and at the eutectic temperature (2,800 K) of 50 -50 ZrO_2/UO_2 the rate is greater than 1E-3%/min. Even over the first one-hour period of the accident this type of rate would result in releases much greater than 1E-7.

Sandia Response

The question is out of the scope of the review since it has been assumed for the purposes of the work under review that MELCOR yields state-of-the-art predictions of radionuclide behavior during severe reactor accidents. Indeed, MELCOR predictions of radionuclide release from the fuel are empirical and are based on the results of a variety of "separate effects" tests including the HI and VI tests done at ORNL and the VERCORS tests conducted in France. The prediction of radionuclide release and transport have been and are being compared to the more integral

core degradation tests done in the Phébus-FP program. MELCOR predictions have not been compared to the results of the VEGA tests done in Japan. Indeed, there is not sufficient data available now to conduct a comparison to the VEGA tests.

The question seems to confuse release from the fuel and release into the reactor containment. It is, of course, release into the reactor containment that is the focus of the work under review. We have no clear idea of what accident was simulated in the VEGA tests or the transport pathway followed by radionuclides once they were released from the fuel. It is dubious however that this transport pathway would be similar to those available to radionuclides in reactor coolant systems under accident conditions.

It is, however, true that, in general, MELCOR predicts low levels of release of lanthanides and actinides during the In-vessel Release Phase of a reactor accident. The lanthanide and actinide radionuclides are relatively refractory in comparison to volatile species such as iodine and cesium that are released extensive. Release of the lanthanides and actinides would not be expected until temperatures are quite high and degradation and liquefaction of the fuel is beginning. This tends to be the stage at which experimental investigations encounter difficulties controlling power and measuring temperature.

There is some understanding of how lanthanide and actinides are released from the fuel. The predominant mode of release is as the monoxide gas that is favored under relatively reducing conditions. For example in the case of cerium release (which is used in MELCOR to estimate plutonium release as well) -

$$[CeO_2]_{UO2} + H_2 \rightarrow CeO(g) + H_2O$$

where $[CeO_2]_{UO2}$ denotes condensed phase cerium dioxide dissolved in the degrading reactor fuel. Similarly, for the release of the lanthanides or actinides in the trivalent state -

$$[La_2O_3]_{UO2} + H_2 \rightarrow 2LaO(g) + H_2O$$

This release mechanism contrasts with the predominant mode of uranium release which is favored by oxidizing conditions that can lead to the formation of volatile hexavalent species -

$$UO_2(c) + H_2O \rightarrow UO_3(g) + H_2$$

It also contrasts with the behavior of ZrO2 which cannot oxidize to a volatile, higher valence state, and is less susceptible to reduction to a higher volatility state. Vaporization of zirconium is largely negligible under conditions predicted for reactor accidents.

The vapor of cerium over the degrading fuel is predominantly the monoxide and its partial pressure is given by -

$$P_{CeO} = K^o \exp\left(-\frac{E}{RT}\right)\left(\frac{P_{H_2}}{P_{H2O}}\right)\gamma_{UO_2}(CeO_2)\,x_{CeO_2}$$

where:

P_{CeO} = *partial pressure of* $CeO(g)$

P_{H_2} = *partial pressure of hydrogen*
P_{H_2O} = *partial pressure of steam*
T = *absolute temperature* (K)
$\gamma_{UO_2}(CeO_2)$ = *activity coefficient of* CeO_2 *dissolved in uranium dioxide*
x_{CeO_2} = *mole fraction* CeO_2 *dissolved in the degrading fuel*

The molar rate of cerium release is given approximately, then, by -

$$\frac{dm_{CeO_2}}{dt} = -\left(\frac{k_m}{RT}\right)\left(\frac{A}{V\rho_m}\right) K^o \exp\left(-E/RT\right)\left(\frac{P_{H_2}}{P_{H_2O}}\right) \gamma_{UO_2}(CeO_2) m_{CeO_2}$$

where -
k_m = *mass transport coefficient*
A = *surface area of degrading fuel available for vaporization*
V = *volume of degrading fuel*
ρ_m = *molar density of degrading fuel*
m_{CeO_2} = *molar amount of cerium in the degrading fuel*
t = *time*

The release of the radionuclide is a first order process. The rate of release depends on -

- Temperature
- Mass transport rate
- Concentration in the degrading fuel
- Hydrogen-to-steam partial pressure ratio
- Surface-to-volume ratio of the degrading fuel

Efforts are made in experiments to control the first three of these factors that affect release rates. At the very high temperatures where releases of lanthanides and actinides can be expected, temperatures are controlled typically by power input since it becomes quite difficult to measure temperature by either electromotive force or by pyrometry. Mass transport rate is usually controlled by the rate of gas (steam) input to the experimental device. The relative scale of experiments and reactor accidents can make it difficult to reproduce the Reynolds number and consequently the turbulence levels of reactor accidents. The concentration of radionuclide is controlled by the selection of the irradiated specimen used in the experiment.

The final two factors are very difficult to simulate in experiments. The hydrogen-to-steam partial pressure ratio in the vicinity of the fuel in a reactor accident is affected not only by what goes on along the fuel rod but also by mixing from other regions of the core. It is very difficult for experiments to simulate the mixing of steam from adjacent fuel channels. Experiments tend to produce high hydrogen to steam ratios if the test specimens are long enough. High hydrogen-to-steam partial pressure ratios locally will lead to higher than anticipated releases of lanthanides and actinides.

At high temperatures, fuel begins to sinter and creep and, eventually, melt. This all leads to a reduction in the surface to volume ratio. Experiments cannot produce as dramatic a reduction in the surface to volume ratio as will occur in reactor accidents. The experiments typically will change the surface to volume ratio a cylinder to a sphere of equivalent mass. In reactor accidents the change in surface area can be far more dramatic. The limited changes in the

surface to volume ratio in small scale experiments lead to higher release rates than will be predicted by MELCOR for reactor accidents where more dramatic changes in the ratio are usually predicted.

It is apparent, then, that it is not usually valid to blindly assume release rates measured in separate effects tests should be predicted by reactor accident calculations even when flow and temperature are well controlled to be those expected in a reactor accident. It takes quite a substantial effort to analyze experiments such as the VEGA experiments to understand why results of these tests differ from results seen in tests such as the Phébus-FP tests, the accident at TMI, and now accidents at Fukushima all of which yielded lanthanide and actinide releases entirely consistent with predictions made by MELCOR. Indeed, even releases of cerium observed in the Chernobyl accident were largely the results of mechanical comminution of the reactor fuel and not attributed to vaporization from overheated fuel even though this fuel did eventually attack the concrete biological shield below the core and flow into lower compartments of the reactor.

In summary, we have not seen situations of large releases of lanthanides and actinides in tests and accidents examined to date that cannot be explained with our current understanding of the release mechanisms for these radionuclides. In this sense, MELCOR produces state-of-the-art release predictions. Releases of the lanthanides and actinides are found to be sufficiently small in accident calculations that attentions have turned to radionuclides for which substantial releases are predicted including iodine, cesium, tellurium, and molybdenum.

Evaluation

The SNL response was quite detailed and informative but at that point I was trying to determine whether there was some aspect of the models actually used in the cases analyzed by MELCOR that might be under-estimating the release of low volatility species. The reply helped in understanding the state of understanding of the release of low volatiles but did not provide insights into the model actually employed in MELCOR.

Subsequently, the following follow-up questions were provided.

Question 1.a - In response to one of my previous questions an expression for cerium release was provided that is more detailed than the expression presented in the MELCOR 1.86 reference manual. Is there a good experimental data base for the activity coefficient of CeO_2 in UO_2 and for the activation energy and coefficient in the Arrhenius equation for the equilibrium pressure? If plutonium were used as its own class, is it reasonable to assume that the monoxide would again dominate the airborne species under reducing conditions and would the predicted release be larger or smaller than estimated for cerium?

Sandia Response

The Ce-U-O phase diagram is not a simple binary system. It like phase relations in the Pu-U-O system must be treated as a ternary system. This is because both plutonium dioxide and cerium dioxide have a propensity for reduction from the tetravalent state especially at elevated temperatures -

$$2\,[CeO_2]_{UO2} \leftrightarrow 2\,[CeO_{1.5}]_{UO2} + ½\,O_2(gas)$$

$$2\,[PuO_2]_{UO2} \leftrightarrow 2\,[PuO_{1.5}]_{UO2} + ½\,O_2(gas)$$

where $[\ ]_{UO2}$ designates the concentration of the bracketed species in the uranium dioxide lattice. This is better understood using Kröger-Vink notation for pure CeO_{2-x} -

$$2\,Ce_m^x + O_o^x \leftrightarrow 2\,Ce_m^\bullet + V_o^{"} + ½\,O_2$$

$$Ce_m^\bullet + V_o^{"} \leftrightarrow (CeV_o)'$$

$$O_o^x + V_o^{"} + (CeV_o)' = 2\text{-}x$$

$$Ce_m^x + Ce_m^\bullet + (CeV_o)' = 1$$

$$Ce_m^\bullet = (CeV_o)' + 2\,V_o^{"}$$

Defect associations are much more complicated than the binary pair shown above especially at lower temperatures where vaporization is quite modest. This is discussed in some detail in Toft Sorensen's book. There have been more recent refinements. Plutonium dioxide undergoes the same behavior with even rather similar quantitative features because of the actinide contraction. (A sadly incomplete writeup of features of the Pu-O system is attached here to illustrate the data base and the complexities of defect association.)

Within the uranium dioxide lattice there is the additional complexity of defect equilibria of the fluorite lattice – especially the formation of interstitial oxygen -

$$2U_m^x + ½\,O_2 \leftrightarrow 2\,U_m^{'} + O_i^{\bullet\bullet}$$

Complexities of the Wallis defects involving oxygen interstitials and electronically defected cations really become negligible at the elevated temperatures where vaporization is important. Near stoichiometry (where we think we always are even for fairly high burnup fuel) there is an significant equilibrium between the formation of oxygen interstitials and the formation of oxygen vacancies -

$$2U_m^x + O_o^x \leftrightarrow 2\,U_m^\bullet + V_o^{"} + ½\,O_2$$

Making the appropriate changes to the charge balance and oxygen balance equations and introducing the uranium balance for $(U_{1-y}Ce_y)O_{2-x}$ -

$$U_m^x + U_m^\bullet + U_m^{'} = 1\text{-}y$$

yields a set of equations that can be parameterized by comparison to the measured partial pressure of oxygen in equilibrium with the solid as a function of temperature. Then, using the Gibbs-Duhem relationship, the chemical activities of both uranium dioxide and cerium dioxide (or cerium sesquioxide depending on preference) can be deduced. These chemical activities

can be converted into empirical activities for the formulation adopted in the abbreviated response to a question completely outside the scope of the study under review. This has been done a lot in the pertinent literature when data for the $(U_{1-y}Pu_y)O_{2-x}$ system were scarce. The data base for the urania-plutonia system has become a lot more abundant in the last decade, so there has been less interest in the urania-ceria stimulant system, but occasional studies still appear.

The "activation energy" mentioned in the question is really just the enthalpy of vaporization that is modestly temperature dependent, but is usually approximated as a constant.

The cerium – plutonium analogy is one of the better analogies in the chemical grouping of elements done to economize on the source term description. The analogy is widely espoused in the solid state literature and would be expected to be even better in the gas phase. It is not as good as the Cs – Rb analogy, but it is probably better than the I-Br analogy and certainly better than Rh-Pd-Ru analogy. As far as a detailed comparison, I have never done one for Ce-Pu. The same is true for most of the groupings. Ba-Sr has been studied to some extent and this has led to episodic recommendations to split this grouping. Qualitatively, the vapor species for Ce-O and Pu-O are the same and the stabilities of these vapor species are very close as is the case for all the actinides.

Evaluation

The response was extremely detailed and very informative in understanding the current state of understanding of these issues. However, what I was truly looking for at that point was a clarification of the models actually used in the MELCOR analyses, a quantitative assessment of how substantially that might differ from the current state of understanding and whether the current MELCOR models might be substantially underestimating cerium (or specifically plutonium) release. A request was made for specific detail on the MELCOR 1.85 expression for pressure as a function of temperature.

Question 1b - A request was made for the specific expression used in MELCOR to determine equilibrium vapor pressure for the Cerium group as a function of temperature.

Sandia Response

The form used for vapor pressure in MELCOR is -

$$\log 10(P) = -\frac{A}{T} + B$$

For the Cerium group A = 21570 K and B = 8.74 with P in mm Hg, and T in Kelvin over the range of 1500 to 10,000 K.

Evaluation

The response was complete and helpful. The equilibrium pressure of a species enters into the assessment of release rate. The purpose for looking at the expression used for cerium was to determine whether there might be an apparent under-estimation of the volatility of the cerium

group, particularly as a representative of the plutonium isotopes. However, in comparing measured equilibrium vapor pressures for PuO2 as a function of stoichiometry, the MELCOR vapor pressures actually used for the cerium group were larger than the results of measured data. We found no data, however available in the range of 2500 – 2800 K, which is of particular interest.

Question 2 - The PWR and BWR write-ups indicate that a fuel rod collapse temperature was used of 2800 K rather than the default value of 2500 K. I assume that you have at some point done sensitivity studies on that parameter. Does the choice of that parameter have a major impact on the calculated releases of low volatile elemental groups?

Sandia Response

The response received from the MELCOR developers is that indeed 2800 K was used in version 1.8.5. They have since moved to 2600 K for versions 1.8.6 and 2.1. They feel the Zr (liquid) "drain out" temperature (2400 K) is more influential on the progression of core degradation. They think that the lower relocation temperature used now might lead to more modest releases of refractory species such as the actinides, but the releases are typically so small relative to the more important radionuclides that no detailed investigation of this issue has been conducted. They have been more focused on aligning MELCOR predictions of fuel relocation with observed relocation and temperatures in the Phébus-FP tests. MELCOR 1.8.5 was taken to be the state of the art for the purposes of the work under review. They don't feel they have an adequate data base to judge whether the releases of actinide other than uranium are in good agreement with results of the tests. Again, releases are observed to be very modest and measurements are likely to have substantial uncertainty ranges especially if repeatability is considered.

Evaluation

This response was very helpful. One possible means by which the release of low volatile radionuclides could be under-estimated is if a significant portion of the fuel melts at high temperature. The use of a 2800 K relocation temperature would tend to over-estimate rather than under-estimate the release.

Conclusion

In conclusion, I have received sufficient information to complete my peer review.

B.3 Dr. Akihide Hidaka

Question 1 - According to the puncture test of irradiated fuel at JAEA, the fractional gap release of volatile radionuclides such as Kr, Xe in BWR (= 5-10%) is clearly larger by about one order of magnitude than that in PWR (= less than 0.5%). This is because the bigger pellet diameter in BWR results in the higher temperature in the pellet center and the enhancement of transportation of volatile radionuclides to the open pore at the peripheral pellet [1].

Sandia Response

Interesting. It would be useful to obtain data for the puncture tests. We have access to data for PWR fuel shown in Figure B.3.1. It appears that variability in the gap inventory depends on both burnup and linear power rate. For the work under review, we attempted to control both at prototypic levels. It is also noteworthy that BWRs are going to fuel rods similar to those of PWRs as burnups are taken to higher levels. Currently, 10 x 10 BWR fuel assemblies are being used and there appears to be efforts to go to 14 x 14 arrays.

MELCOR uses a very simple, constant, 5%, for gap inventory of volatiles. The inventory does not control the material reaching the containment and classed as Gap Release. This is controlled largely by the flow velocity through the reactor coolant system. This was taken to be high in the development of the NUREG-1465 source term. It varies among the accidents considered in the work under review. Gap Releases for noble gas radionuclides vary over about an order of magnitude from about 0.3% to ~3%. Again, this is not gap inventory. It is material released following initial clad ballooning and rupture that successfully negotiates passage to the containment prior to the start of the In-vessel Release Phase which is declared to occur when 5% of the core inventory of xenon has been released from the fuel but not necessarily from the reactor coolant system. By the time this arbitrary criterion is reached, some fuel can be in stages of in-vessel release while other fuel nodes have not yet reached the point of clad ballooning and rupture.

Evaluation

The given explanation for the gap release is quite logical. However, I am slightly concerned that the end of present gap phase or the initiation of in-vessel release is determined at the time of increase in release up to 5%. Since the present phase name, “gap release”, is closely related to the phenomenon, some people easily imagine that the FP release due to the cladding burst is equal to 5%.

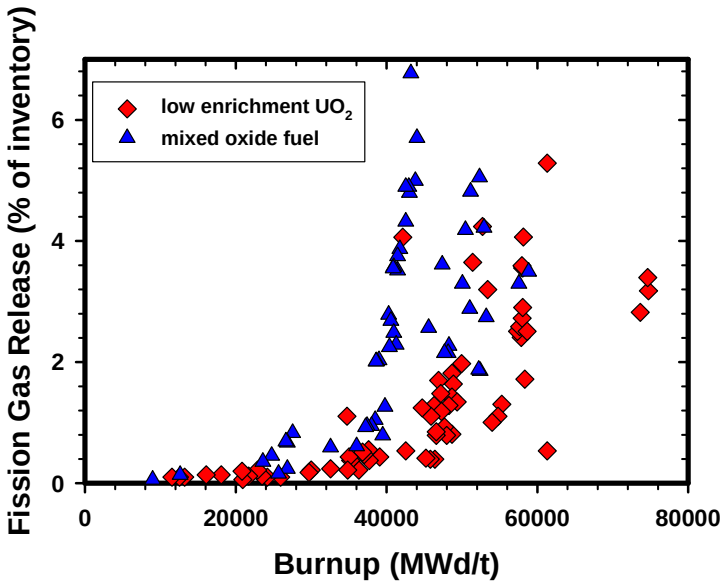


Figure B.3.1 Measured FP gas release to the gap and plena of fuel rods

Question 2 - A sensitivity analysis with the JAEA's source term analysis code, THALES-2 considering the pressure effect showed that the radionuclide release from PWR fuel before RCS failure could decreased compared with that from BWR fuel because the RCS pressure in PWR is higher than that in BWR and the release from PWR fuel has higher sensitivity to the pressure effect[1]. The pressure effect causes the decrease in release rate from fuel at elevated pressure by several factor of magnitude compared with that under atmospheric pressure, and does not mean that the fractional release of volatile FP would not increase more 60 or 70%. It would take more time to be release compared with the case under atmospheric pressure but the fractional release could react 100% for some time in future.

Sandia Response

MELCOR considers serial processes for release of radionuclides from the fuel. Radionuclides diffuse to the grain and then transport through the pore network to the fuel pellet surface and eventually into the ambient flow through the core. Currently, the gas phase transport through the pore network is simple gas diffusion and varies with the reciprocal of pressure. We agree, however, with the interpretation of the VEGA results that Knudsen diffusion in the pore network needs to be considered. Knudsen diffusion will depend on the reciprocal of the square root of pressure. This will ameliorate some the differences between BWR and PWR though such differences will remain since in most circumstances transport will be under mixed control. That is, the pressure dependence will vary from the reciprocal of pressure to the reciprocal of the square root of pressure.

Some accident sequences do not result in complete release of noble gas radionuclides during the In-vessel Release phase. This is because a fraction of the core melts quickly and penetrates the reactor vessel. Fuel that has not melted or liquefied remains in the reactor vessel and releases noble gas radionuclides during the Late In-vessel Release Phase.

Evaluation

I consider that the pressure effect does not only depend on the Knudsen diffusion but also on increase in the concentration gradient in the grain. That is, the rate-determining step exists at both of diffusion in grain and pores. Expected distribution of radionuclide concentration in UO_2 grain and open pores in pellet at elevated pressure are shown in Figure B.3.2.

The elevated pressure could cause increase in gas density in open pores and decrease in gaseous diffusion velocity. These result in increase in radionuclide concentration in small-volume pores and at grain surface although the diffusion velocity in grain is much slower than that in open pores. The concentration increase at grain surface may suppress radionuclide release from grain because it decreases the concentration gradient in grain that is a driving force of diffusion. Consequently, the pressure dependence will not vary from the reciprocal of pressure to the reciprocal of the square root of pressure but vary from the reciprocal of smaller pressure dependence, for example, the cubic or fourth root of pressure, to the reciprocal of the square root of pressure. In reality, the pressure dependence with the reciprocal of the square root of pressure will slightly overestimate the observed pressure effect as shown in Figure B.3.3.

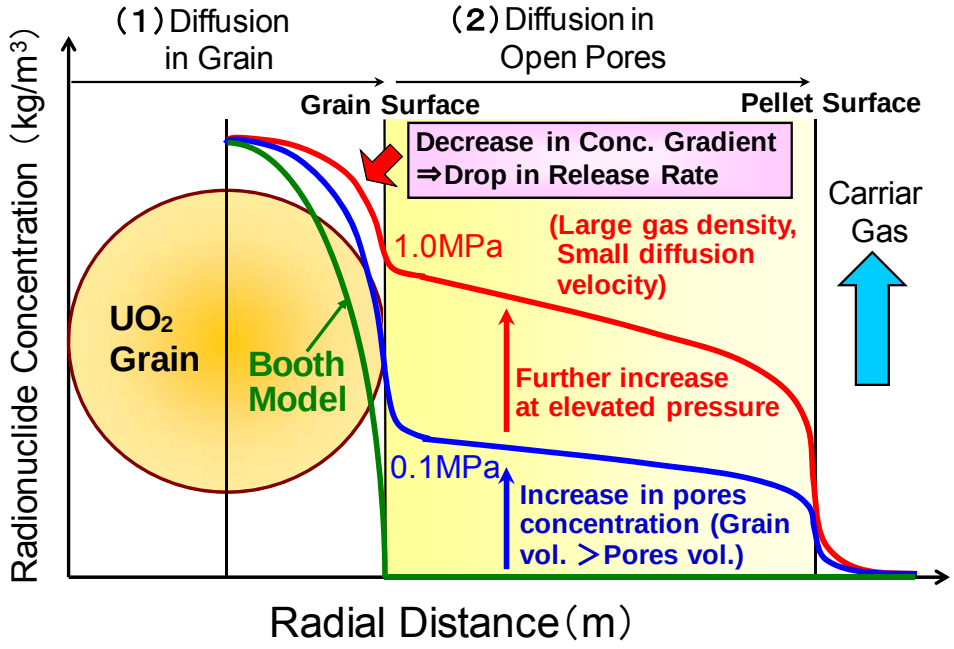


Figure B.3.2 Distribution of concentration in grain and pores in pellet at elevated pressure [1]

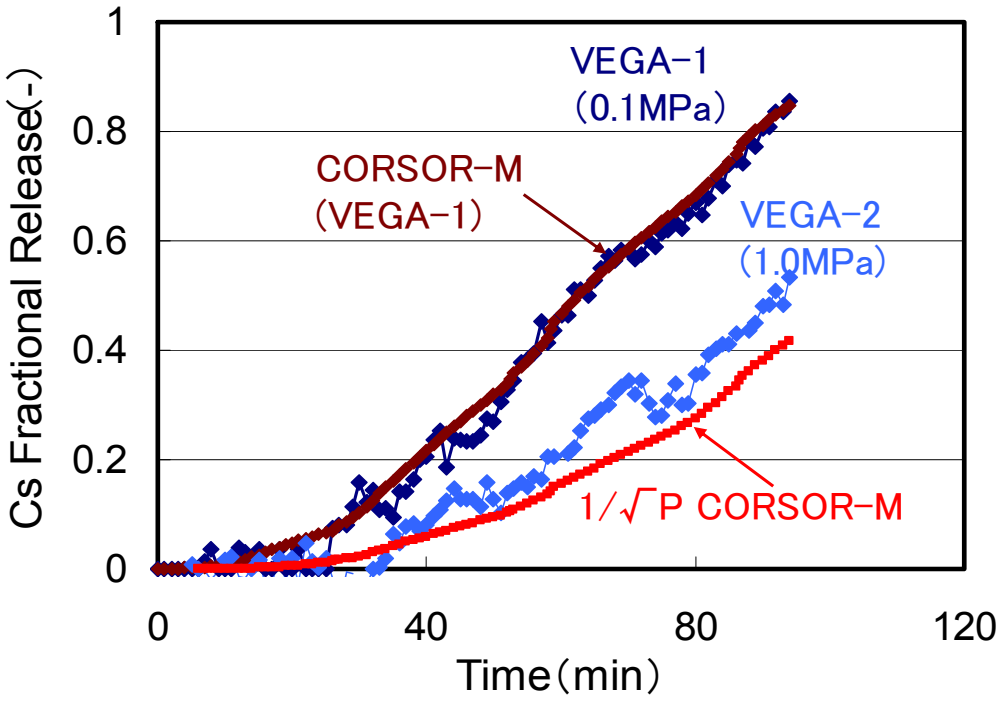


Figure B.3.3 Comparison of Cs release between measurements and 1/P0.5 CORSOR-M model [1]

Question 3 - The formation of Cs_2MoO_4 and enhancement of Mo release from MOX fuel were proposed in the SNL report based on the VERCORS results and the thermal equilibrium calculation. On the other hand, in the VEGA experiment with ICP-AES measurement for the nitric acid solution leached from the VEGA apparatus, the increase in Mo release from MOX was not observed, However, this VEGA result does not contradict the VERCORS result because Cs-134 and Cs-137 have high intensity of gamma spectrum and therefore in the VEGA experiment, cesium and its chemical compounds including Cs_2MoO_4 has to be removed from the solution before the ICP-AES analysis to avoid contamination of the ICP-AES system and to decrease exposure to gamma ray.

Sandia Response

The suggested form of cesium as cesium molybdate is still a topic of technical discussions that will refine, eventually, the state of the art. Note that MELCOR invokes cesium molybdate as the form of cesium in addition to CsI not just for MOX or HBU fuel. Cesium molybdate is invoked for all scenarios. By selecting cesium molybdate as the chemical form of cesium not bound to iodine, MELCOR can better predict results of Phebus tests for release and for transport. Still, there is no proof that cesium molybdate is the chemical form. We suspect that there are a variety of chemical forms. A systems level computer code does have to make some simplifications to achieve acceptable running times while still obtaining results of adequate validity.

It is clear from the results of the Phébus tests as well as the VERCORS tests that molybdenum releases for whatever reasons are higher than those of other transition metals such as Pd and Ru. Consequently, we have created a separate chemical release class for molybdenum.

Evaluation

SNL's estimation is considered to be reasonable. No objection for estimation of the chemical form of Cs2MoO4.

Question 4 - The oxygen potential in the MOX fuel generally becomes higher than that in UO_2 fuel. Therefore the releases of Te, Sb, Sn, Mo, Ru of which oxides have higher volatility could increase compared with UO_2 fuel while those of Ba, Sr, Eu, Ce, of which oxides have lower volatility, could decrease. According to the SNL report, Ru release from MOX fuel becomes slightly smaller than that from UO_2 fuel in contrast to the description above.

Sandia Response

To within the confidence interval of our ability to locate the medians of the distributions, the In-vessel Release fractions for ruthenium from MOX and from LEU fuels are identical:

MOX: 0.005±0.002
LEU: 0.006±0.002

Indeed, this is not only true for the median, but also true for the distribution of results for ruthenium release from MOX and LEU. Of course, MELCOR does not recognize air ingress as

an accident phenomenon that will affect releases from fuel in the Late In-vessel Release Phase. In particular, it could affect ruthenium release. The effects might well be similar for LEU and MOX fuels.

We have not been able to find experimental confirmation of the theoretical hypothesis that oxygen potential in irradiated MOX will be significantly higher than irradiated LEU. We believe that oxygen potentials in both fuels once they become hyperstoichiometric is buffered by the equilibrium:

$$[Mo]nodule + O_2 \leftrightarrow [MoO_2]lattice$$

where Mo is alloyed in nodules with Ru, Pd, Rh and other elements (see Figure B.3.4). The molybdenum dioxide product may or may not be soluble in the fuel lattice to an appreciable extent. The equilibrium is essentially the same for both types of fuel and so the oxygen potentials are about the same. Certainly, they are sufficiently similar that we are unable to find data that would provoke major changes in the modeling. It is hoped that further insights will be derived from the forthcoming VERDON experiments. We fully anticipate that results of these tests along with results from the VEGA tests will provoke improvements in the state of the art and, consequently, in the MELCOR modeling.

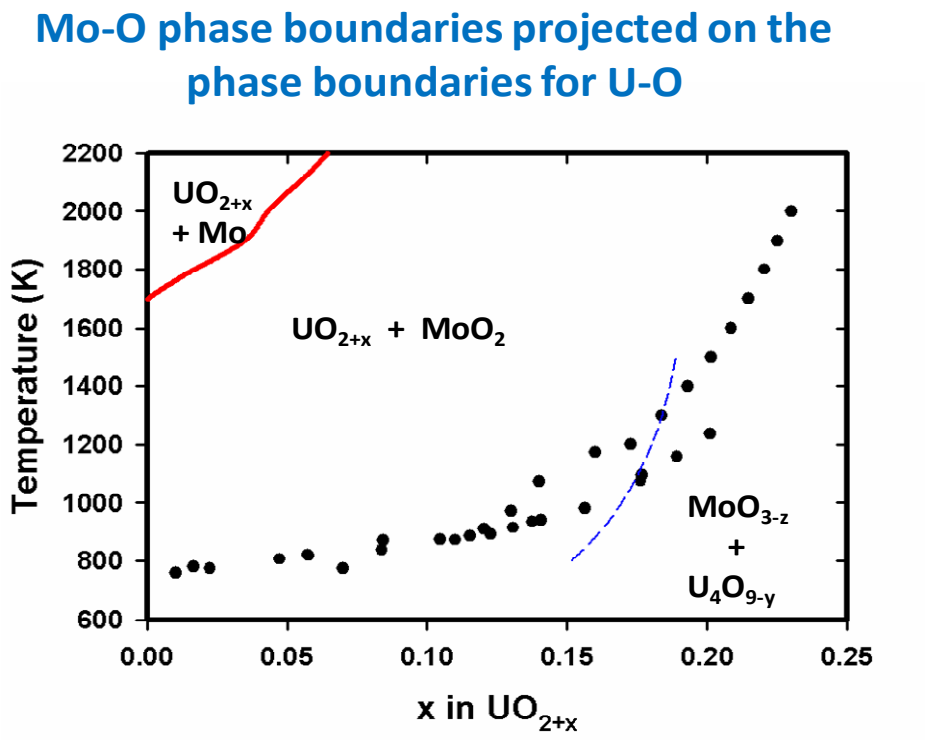


Figure B.3.4 Mo-O phase boundaries projected on the phase boundaries for U-O

Evaluation

I prepared the following comments in this May.

"As the technical basis for the higher oxidation potential in MOX fuel ($U_{0.8}Pu_{0.2}UO_{2.000}$) than that in UO_2 fuel, Figure B.3.5 [2] prepared by H. Kleykamp was referred in the VEGA test interpretation. In fact, the comparison between VEGA-M1 and VEGA-8 showed that the Ru release from MOX fuel became higher than that from UO2 fuel."

After that, I continued the discussion with several specialists and tried to find the experimental data which directly shows the higher oxidation potential in MOX fuel than that in UO_2 fuel. A lot of specialists agree with the idea that the oxygen potential in MOX could be larger due to the formation of precious metals but we could not find any data at all except for the release data in VEGA tests which support the higher oxidation potential in MOX fuel. Although I agree with the SNL's response that explains the buffering of stoichiometry by Molybdenum oxidation, further examination is considered to be needed for the oxidation potential of MOX and UO_2 fuels. Concerning the VEGA results, it is necessary that the reproducibility of the results be further examined.

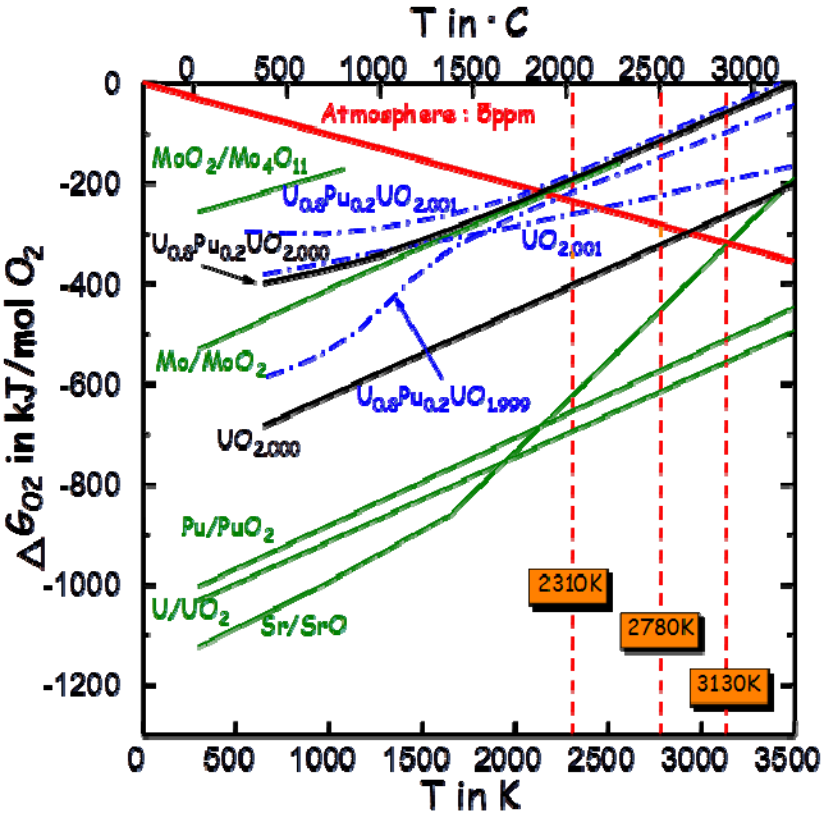


Figure B.3.5 Ellingham diagram of relevant elements [2]

Question 5 - The melting point of PuO_2 is about 2670 K which is lower by about 430 K than that of UO_2. In addition, the eutectic reaction between ZrO_2 cladding and fuel becomes active above 2600-2700 K. Therefore, Pu release from MOX fuel at high temperature before RCS failure

could increase, as observed in VEGA test, by one or two orders of magnitude in comparison with conventional CORSOR-M model prediction.

Sandia Response

We are assured that there will not be many instances of isolated PuO_2 phases within the MOX fuel to be used in US reactors. Elaborate efforts are expended to get atomic dispersion of plutonium in the uranium dioxide lattice. Still there will be some areas enriched in plutonium dioxide though this plutonium will be dissolved in urania. PuO_2 and UO_2 form a miscible system so the actual melting point of the fuel will be little different than that of irradiated LEU. We don't expect the melting point to be lower than that of LEU fuel by more than about 170 K. We expect that in MOX as in LEU, melting will not occur. Instead, there will be liquefaction as cladding attacks the fuel. We are unaware of a eutectic reaction between ZrO_2 and fuel. There is a minimum in the melting point in the system ZrO_2-UO_2 and there is a monotectic reaction in the system Zr-U-O. As noted in the text, for the purposes of the work under review modifications were not made to the core degradation modeling for MOX since we lack data to guide the modeling changes. The MELCOR modeling does not use directly phase relations. Rather, a "fuel relocation temperature" is specified. Values derived from PHÉBUS tests are lower than the melting points of both urania and plutonia.

Though MELCOR allows users to adopt CORSOR-M as the release model, this model was not used for the work under review. Finally, a one to two order of magnitude increase in the release of plutonium would still be a very small release.

Evaluation

It seems me that there are two issues on Pu release in the present study on revised NUREG-1465. One is the issue relating to the fitting of HBU-Booth and MOX-Booth model used in this study as described in chapter 2. The other is the issue that the release rate of Pu could increase rapidly above 2,800K and therefore use of the booth model with the fixed or constant pre-exponential factor for temperature above 2,800K may not be appropriate for the severe accident analyses.

The release of Pu from MOX below 2,800K was predicted well by the conventional models such as ORNL-Booth while the release above 2,800K was enhanced and became larger by about one order of magnitude than that of the conventional models[3]. The existing models including the ORNL-Booth model underestimated it by about two orders of magnitude due to lack of experimental data at such high temperature as shown in Figure B.3.6. The enhancement of Pu release above 2800K was also observed in the VEGA-8 test with UO_2 fuel (see Table B.3.1).

However, it should be pointed out that the eutectic reaction could proceed too much in the VEGA geometry with crucible that maintains liquefied materials compared with the reality without crucible. The eutectic reaction is expected to be further examined in other future tests that simulate better the actual conditions during severe accidents.

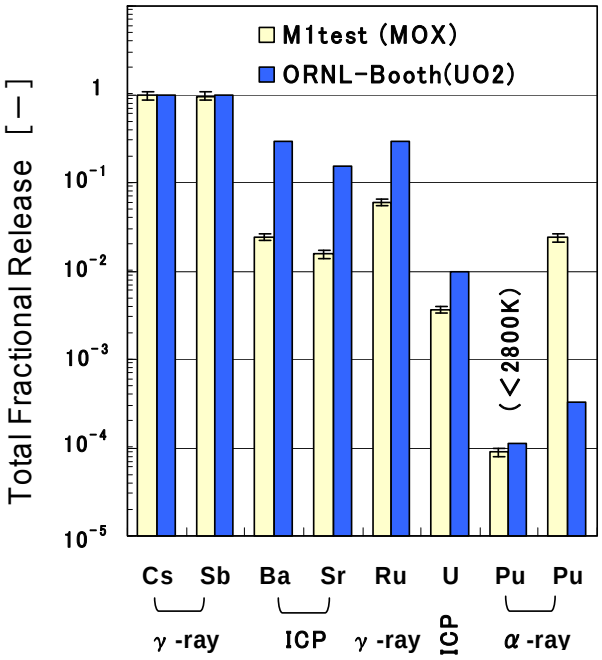


Figure B.3.6 Comparison of radionuclide release between VEGA-M1 and ORNL-Booth [3]

Table B.3.1 Release fraction of radionuclides at the end of VEGA tests [1]

Test No.	Fuel	Test conditions	Fractional release (%)											
			γ ray measurement (Half-life)								ICP-AES, α ray			
			^{137}Cs 30yr	^{125}Sb 3yr	^{131}I 8d	^{132}Te 3d	^{140}Ba 13d	^{106}Ru 1yr	^{103}Ru 39d	^{140}La 2d	Mo	Sr	U	Pu
1	PWR 47GWd /tU	2,773K, 0.1MPa, He	86	89	—	—	—	5	—	—	—	—	—	—
2		2,773K, 1.0MPa, He	61	68	—	—	—	0	—	—	—	—	—	—
3		3,123K, 0.1MPa, He	100	95	—	—	—	0	—	—	—	—	—	—
4		2,773K, 0.1MPa, Steam	93	—	—	—	—	22	—	—	—	—	—	—
5		8hr NSRR Re-irradiation He, 2,900K, 1.0MPa	84	67	—	—	7	0	0	—	—	—	—	—
M1	ATR/MOX 43GWd /tHM	3,123K, 0.1MPa, He	97	95	—	—	—	6	—	—	0.6a	2	0.4a	2.8
M2		3,123K, 1.0MPa, He	98	96	—	—	—	3	—	—	—	—	—	—
6	BWR 56GWd /tU	624hr JRR-3 Re-irradiation H_2O, 2,773K, 0.1MPa	93	—	97	98	49	14	16	3	—	—	—	—
7		624hr JRR-3 Re-irradiation H_2O, 2,773K, 1.0MPa	98	83	96	98	34	6	7	4	—	—	—	—
8		3,123K, 0.1MPa, He	97	—	—	—	—	0	—	—	0.3a	11	0.1a	2.1

a) Below detection limit

On the other hand, it is noted that during the Fukushima Dai-ichi accident, quite small but certain amount of Pu was released to the environment and was surely measured at the vicinity of the site. Japanese government's report describes as follows [4]:

"The concentrations of Pu-238, Pu-239 and Pu-240 detected on August 8 are almost equivalent to the level of fallouts observed in Japan in the past atmospheric nuclear tests. However, the ratio of radioactivity (Pu-238/Pu-239 + Pu-240) was greater than the ratio 0.026 which is the indicated value affected by the past atmospheric nuclear tests, so that we have determined that the detected values were derived from the accident at Fukushima."

References

1. A. Hidaka, "Outcome of VEGA Program on Radionuclide Release from Irradiated Fuel under Severe Accident Conditions" J. Nucl. Sci. Technol. 48, No.1, pp.85-102 (2011).

2. H. Kleykamp, "The Chemical State of the fission Products in Oxide Fuels," J. Nucl. Mater. 131 , pp.221-246 (1985).

3. A. Hidaka, et al., "Radionuclide Release from Mixed-Oxide Fuel under High Temperature at Elevated Pressure and Influence on Source Terms," J. Nucl. Sci. Technol. 42, No.5, pp.451-461 (2005).

4. Nuclear Emergency Response Headquarters, Government of Japan, "Report of the Japanese Government to the IAEA Ministerial Conference on Nuclear Safety - The accident at TEPCO's Fukushima Nuclear Power Stations -," June 2011. http://www.kantei.go.jp/foreign/kan/topics/201106/iaea_houkokusho_e.html

B.4 Dr. Thomas Kress

Question 1 - My overall assessment is that the design basis source term to containment as determined by Sandia using MELCOR 1.8.5 is an appropriate replacement for the NUREG–1465 alternative source term for LWRs with low enriched fuel as well as for both high-burnup and MOX fuels.

Although this new proposed design basis source term appears to be somewhat less stringent (less conservative) than is NUREG–1465, my opinion is that the lost conservative margin is still acceptable while providing the plants with a much more realistic source term that will permit optimizing the responses. This realism could very well result in improved safety. My one reservation is regarding the potential effects on siting (exclusion boundaries and distances to population centers) for any new plants that choose to adopt the new alternative source term specification. My reservations regarding this would be considerably alleviated if the new increased duration would no longer be given the artificial assumption of a constant rate of fission product release. In the interest of preserving margin and realism, this assumption must be discarded in favor of a realistic representation of the actual calculated timing.

Sandia Response

The constant release rate assumption is consistent with what was done for the NUREG–1465 source term and the objective of having a source term that is easily applied and reasonably conservative. A very detailed approach to timing would not be consistent with the ease of applicability requirement. It has been suggested that the in-vessel release rate could be divided into two stages – an early in-vessel release and an intermediate in-vessel release. The release

rate in each phase would be constant but rates would be different. Further breakdown of the in-vessel release phase into more numerous stages seems inconsistent with the breadth of the uncertainty distributions. Even breaking the In-vessel Release Phase into two stages poses a challenge – how to select the end of the first stage and the beginning of the second without being ridiculed by Kress as arbitrary.

At this stage, it is not possible to state definitively whether the two-stage in-vessel release is more conservative than the single in-vessel release. Calculations done with RADTRAD for just two cases show there not to be a radical change in dose, though the timing of the peak dose is clearly shifted to later times by the single-stage, constant rate, in-vessel release.

Evaluation

Acceptable.

Question 2 - This study utilized the adjusted Booth models for fission product release from the fuel. These basically empirical models provide the best comparison with all of the currently available experimental data. These can, therefore, be considered to be the state-of-the-art. The Booth model is applied to cesium releases. The release fractions for the remaining fission product groups are simply scaled to the cesium values based on experimental data.

This “relative volatility” scaling works well for the various groups and, thus, avoids the overwhelming necessity to develop individual Booth models for each group. However, more information is needed in the report on the scaling values that were used and the sources of these.

The chemical forms of the fission products appear to be important only for iodine, cesium, molybdenum, ruthenium, and (perhaps) tellurium. There is not yet enough evidence to change the NUREG–1465 specified chemical form of iodine as 95% cesium iodide and 5% free iodine . On the other hand, there is ample evidence that cesium does not behave as cesium hydroxide as has been the assumption in past source term specifications. Based on PHEBUS and VERCORS data, the behavior of cesium seems to be best represented as cesium molybdate, which also further clarifies the behavior of molybdenum. Making this specification change for the new proposed source term is amply justified and is an improvement. Another improvement involves the release of tellurium which, previously has been considered to “tie-up” with un-reacted zirconium from the clad. The range of experiments in PHEBUS and VERCORS has contradicted this and it now appears that tellurium gets released during the in vessel phase as a relatively volatile species. This change, too, is a good one that addresses the bulk of the experimental data. My overall assessment of the fission product release modeling in MELCOR 1.8.5 is that it represents the state of the art and is acceptable for use in determining the design basis source terms.

Sandia Response

Agreed.

The objective is to keep the report succinct. Information on the scaling factors is available in the documentation on the MELCOR model (see the next two [responses]). It does not appear useful for most readers to reproduce and explain this material in the document itself. It might be possible to include the information in some additional support document.

From the MELCOR RN Package Reference Manual

The diffusion release rate for species other than cesium is given by multiplying the cesium release rate by an appropriate scaling factor S_k for each RN class k -

$$DIFF_k = DIFF_{Cs} \times S_k \quad (2.18)$$

Nominal values for S_k are given in sensitivity coefficient array 7103. For certain conditions of cladding oxidation and temperature, the scaling factors must be modified for some classes. When the oxide mass fraction exceeds a critical value F_{k1} and the temperature exceeds a critical value T_{k1}, the class scaling factor is given by

$$S_k = S_{k1} \times \exp(C_k T) \quad (2.19)$$

where T is not allowed to exceed a maximum value T_{max}. When the oxide mass fraction is below a minimum value F_{k2}, the class scaling factor is given by

$$S_k = S_{k2} \quad (2.20)$$

Values for F_{k1}, T_{k1}, S_{k1}, C_k, T_{max}, F_{k2}, and S_{k2} are all contained within sensitivity coefficient array 7107.

RN class	CORSOR-Booth Class Scaling Factors: Nominal Values	cladding oxide mass fraction upper threshold	cladding temperature upper threshold	upper threshold scaling factor constant	upper threshold scaling factor exponential constant	upper threshold maximum cladding temperature	cladding oxide mass fraction lower threshold	lower threshold scaling factor
	S_k	F_{k1}	T_{k1}	S_{k1}	C_k	T_{max}	F_{k2}	S_{k2}
k	C7103(k)	C7107(1,k)	C7107(2,k)	C7107(3,k)	C7107(4,k)	C7107(5,k)	C7107(6,k)	C7107(7,k)
1	1.0	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
2	1.0	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
3	4.0E-04	1.1	0.0	0.0	0.0	0.0	0.05	0.002
4	0.64	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
5	0.64	1.1	0.0	0.0	0.0	0.0	0.70	0.64
6	0.0025	0.75	2300.0	0.0025	0.0	2700.0	-1.0	0.0
7	0.001	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
8	4.08E-08	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
9	4.08E-08	1.1	0.0	0.0	0.0	0.0	0.05	4.0E-08
10	0.00032	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
11	0.25	0.75	2000.0	0.25	0.0	2300.0	-1.0	0.0
12	0.16	0.75	2000.0	0.16	0.0	2300.0	-1.0	0.0
16	0.64	1.1	0.0	0.0	0.0	0.0	-1.0	0.0
17	1.0	1.1	0.0	0.0	0.0	0.0	-1.0	0.0

The scaling factors given by C7103(k) above must be modified under certain conditions of cladding oxidation.

When the oxide mass fraction exceeds C7107(1,k) and the temperature (TEMP) exceeds C7107(2,k), the class k scaling factor is given by:

SFACT(k) = C7107(3,k) x EXP(C7107(4,k) x MIN(TEMP , C7107(5,k)))

When the oxide mass fraction is below C7107(6,k) the class k scaling factor is given by:

SFACT(k) = C7107(7,k)

Evaluation

Acceptable.

Question 3 - The fission product transport modeling in MELCOR still considers deposition via condensation of vapors and mass transport of aerosols. The major current difference involving the new proposed source term is in the new specification of the chemical form of cesium and molybdenum. This change tends to slightly reduce the re-vaporization of cesium and increases the re-vaporization of molybdenum. These changes do not markedly affect the source term other than for molybdenum and are acceptable changes.

Sandia Response

Agreed. The effects on molybdenum did prompt the definition of an additional chemical class for the radionuclides.

Evaluation

Acceptable.

Question 4 - My first comment on this area is that, in my opinion, there is not a strong need for assessing uncertainties in specifying a design basis source term. The real uncertainties are in the MELCOR determinations themselves. These uncertainties, while substantial, have been previously assessed and we have fairly good knowledge of the extent of these.

Sandia Response

A design basis source term might consider accounting for these by selecting bounding sequences and/or some higher (than the mean or median) confidence level. However, there is essentially no technical basis for any choice other than how the specified source term affects those items it is used for (e.g. siting) and whether the resultant values are acceptable from a risk standpoint. Such a technical basis has never been developed other than that the current and past design basis source terms appear to be conservative enough that the resultant effects have been acceptable. The currently proposed source term is not sufficiently different from NUREG 1465 and one would expect any different result.

Recognize that these uncertainties are not being ignored in the overall NRC research program. There are active efforts both in modeling and in experiments to reduce the uncertainties. This revision of the source term was done under the requirement that it reflect the current state-of-the-art which was defined as MELCOR. As research reaches fruition and MELCOR is upgraded, there may well come a time when it is useful to again assess the accident source term used in the regulatory process.

A different and useful perspective is that the source term is simply a mechanism to facilitate the performance-based evaluation of capabilities to mitigate consequences of accidents. Conservatism is incorporated in the definition of accidents and the acceptance standard which

is about half the dose necessary for non-stochastic effects of radiation to appear in the affected person.

Evaluation

Acceptable.

Question 5 - The integration into MELCOR mostly involves the selection of appropriate Booth models for LEU, HBU, and MOX fuels. In addition, the finite cellular representation of the cores have utilized real specifications for radial and axial power distributions and locations of MOX fuel. The use of ORIGIN to develop inventories under these new conditions was well-advised and gives assurance that the initial inventories and their distributions are on a good technical basis. This integration is entirely appropriate and was well done.

Sandia Response

Thank you.

Evaluation

Acceptable.

Question 6 - I very much like the idea of basing a design basis source term on accident sequences selected to encompass most of the core damage frequencies (CDF) for the various reactor types. I prefer CDF as opposed to risk-dominant sequences because this tends to minimize the effects of siting and atmospheric transport on the selection. The sequences selected for the various PWR, BWR, HBU, and MOX conditions are appropriate and the specific plants for which the ST [source term] assessments were developed are good choices because they have had a good history of PRA assessment giving some assurance that the CDF values can be relied on.

Each of these sequences were then given a MELCOR assessment to determine sequence specific fission product releases into containment along with the timing thereof. At this point, Sandia chose to use a non-parametric statistical analysis of the releases and separately the timing to determine distributions and to extract from those the median values. One might ask what is the added value of this non-parametric treatment? There are basically two reasons for such treatment. The first is that the selected sequences from specific plants represent only a sample from the broader population of similar plants. If one had the entire set of sequences from all plants, the releases and the timing would likely resemble a continuum very much like that developed by a non-parametric statistical analysis treating the specific source terms as if they represented a random sample from this continuum. If that is indeed a good approximation, then this gives a way to actually select the median (or some other value) and have some assurance that it might be close to the actual median of the full population. The so-called “variation” developed for the distributions gives some notion of how well the median is known based on the number of samples and their locations. I don’t think these “variations” play any other role and indeed I don’t think they were actually used for anything. I would not have given these as much prominence in the report.

There does not seem to be any real technical basis for selecting the median values for the design basis source terms. Such a technical basis would have to involve an evaluation of the effects of the selection on the things that the ST is used for such as equipment qualification (failure rates of SSCs), containment isolation times, extent of control room operator protection, design leak rate specifications, and associated effects on siting (exclusion boundaries and distances to population centers) and whether or not these effects are acceptable. Developing any such technical basis is an overwhelming task and really cannot be done. So one is left with judgment and the fact that previous source term specifications seem to have provided sufficient risk margin and defense-in-depth. All-in-all, the use of the median at this time appears to give a ST that is not radically different from NUREG–1465 which has been in use for some time now and does seem to provide sufficient defense-in-depth. The various plants that have utilized NUREG–1465 have relatively good risk statuses and generally meet the Safety Goals. This gives some assurance that use of the new proposed ST will not erode the margins unacceptably.

Sandia Response

This is a good point and will be included in amending the document.

Agreed. CDF values for the particular plants used in the analysis were used only qualitatively since source terms are defined at a high level of aggregation. That is, aggregation is done at the level of BWR and PWR rather than at the level of BWR-3/4, BWR-5/6 or PWR-CE, PWR-B&W, and PWR-W.

The interpretation of the 'variations' is correct. They were devised so that meaningful comparisons could be made between LBU and HBU fuel source terms and between LEU and MOX fuel source terms. Other than that, they serve no important function. Perhaps, they show when the differences between the source terms calculated in the report are significantly different than the NUREG–1465 source term. It is clear that they have been overemphasized in the current version of the report and some other way will have to be found to present the comparisons between LBU and HBU and between LEU and MOX.

A different and useful perspective. The median was selected based on statistical robustness considerations and an alternative view of the use of the source term. It is good that the median is found acceptable by those with alternative views.

Evaluation

Acceptable.

Question 7 - The proposed release quantities make a good representation of accidents with considerable core melting and are based on the best available experimental data and analysis of sequences. The somewhat significant changes with respect to cesium, molybdenum, and tellurium have good experimental bases and provide a more realistic specification of the source terms. The one attribute that bothers me is related to the timing of the source terms. If the past practice of specifying the release as a constant rate is continued in the new specification, my judgment is that this would provide too much erosion of the margins and would not provide the

desired realism for optimizing responses. The new source term ought to propose abandoning the constant rate specification and recommend something more realistic and representative of the calculated values.

Sandia Response

The release rate during the in-vessel release phase of accidents is the only rate that deviates very significantly from a constant. The approximation of constant release rate seems adequate for Gap Release, Ex-vessel Release and Late In-vessel Release. A more refined approximation of the In-vessel Release Rate is to break this phase of the accident into two stages - Early In-vessel Release and Intermediate In-vessel Release. Separate, but still constant, release rates would be specified for each of these stages. More refined treatment of the in-vessel release rate would largely obviate the utility of the accident source term. Even the modest change to two stages of in-vessel release would prompt significant changes to computer codes used by both the regulator and the licensees. The judgment that the constant release rate will erode margin cannot be contested now. Two calculations that have been done suggest that dose timing is changed but peak doses are not greatly altered when using two stages rather than one stage.

Evaluation

Acceptable.

Question 8 - The report - “Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel” by Powers et al does not stand alone without the associated supporting documents. It is good that the document provides complete references to all these supporting documents.

With respect to the determination of the median values, it is not clear to me whether or not non-parametric statistics were used for each nuclide group for the three conditions or was done only for the Halides and the Alkali Metal groups with some sort of scaling for the other groups. Scaling would appear to work only for the in-vessel release portion of the source term. How the medians were obtained for all the groups should be made clearer in the report. In addition, the report should provide more information of the scaling factors for the in-vessel releases of the various nuclide groups (e.g. what were the scaling factors and how were they obtained.)

Sandia Response

The non-parametric analyses were done individually for each group release fraction and for each release phase for MOX, LEU, BWR LBU, BWR HBU, PWR LBU and PWR HBU. No scaling was done. Results were not considered significant and were not reported for Ex-vessel Release of refractory metals because of a known deficiency in the VANESA module of MELCOR that calculates molybdenum and ruthenium releases during core debris interactions with concrete. For these cases, the NUREG–1465 source term was adopted. For some elemental classes in some phases of the accidents such as gap release of more refractory elements, release fractions were so small reporting values was thought not meaningful. The report will be amended to make clear that distinct analyses were done and that there was no ‘scaling’.

Evaluation

Acceptable.

B.5 Dr. David E. Leaver

Overall Observations

The high B/U, MOX project is well-conceived and provides a much improved basis for a proposed accident source term, incorporating what has been learned over the last 20 years on severe accident core damage progression and fission product release and transport. In addition, the work is well-documented.

I have two additional overall observations. The first concerns the Fukushima accident. Above and beyond its effect on Japan, this accident is significant in two ways: (1) its potential impact on the U.S. industry as well as globally, and (2) the visibility of the accident and the public attention that it has drawn and continues to draw. Because of this, not mentioning the accident in the high B/U, MOX documentation strains the credibility of the NRC effort. I believe that there should be some discussion of the accident somewhere in the documents, even with the recognition that there is still a great deal we don't know about what happened. I will have some specific suggestions on how to proceed on this in my individual peer review report in July.
The second concerns the Ce group in-vessel release (1.3E-7) which applies to Pu and the fact that this estimate is several orders of magnitude lower than what is in NUREG–1465. This matter was discussed by the peer review group in a conference call on June 27, 2011. I will address this further in my individual peer review report in July. My current thought is that the MELCOR value for the Ce group should be used for the revised accident source term, but that there should be further explanation in SAND2011-XXXX of the reasons for the difference from the NUREG–1465 value, and a summary of what is known today about Pu release from Phébus, the ORNL VI tests, VERCOUR, VEGA, and Fukushima. I have also attached a table of data from the TMI-2 accident and the SFD tests on Pu release that I presented in the 2002 peer review meetings.

Evaluation of Comment Responses

The evaluation of SNL comment responses consists of the original comment, the SNL response to the comment, and my final evaluation of the response. Comments 1 to 9 are general comments which affect more than one report and/or address technical issues or matters affecting objectives and methodology, and comments 10 to 34 are more specific comments that refer to a particular page and/or subsection of one report.

Question 1 - The high B/U, MOX source term documents state that epistemic uncertainties from the overall distribution of source terms calculated for the various accidents are not considered. For example, page 21 of SAND2011-XXXX, "Accident Source Terms for LWR Nuclear Power Plants Using High B/U or MOX Fuel," states that… "No attempt has been made to systematically quantify the state of knowledge uncertainty (epistemic uncertainty) in the results." An uncertainty analysis was performed in SAND2011-XXXX to address the aleatory (statistical) uncertainties.

There has been much discussion within the peer review group regarding the need for doing something to address epistemic uncertainty and the merit of including the aleatory uncertainty analysis. I have three comments on this -

Question 1a - My view on this is that it is not worth expending significant additional effort to address uncertainties since the main source of uncertainty (given that containment failure is not being considered as part of a proposed accident source term) is in the range of possible accident sequences as opposed to phenomenological uncertainties in a given sequence.

Sandia Response

Agree for the purposes of this exercise. Recognize that phenomenological uncertainties are not being ignored in the overall source term research undertaken by NRC. Additional insight on release fractions and effects of air ingress are being sought from the VERDON experiments. Studies of aerosol transport, bounce and breakup are being researched in the ARTIST program. Transport and deposition of radionuclides in the reactor coolant system are being studied in the Phébus project. Additional studies of chemical form and behavior of radionuclides will be conducted in the STEM project.

Evaluation

See evaluation of comment b response.

Question 1b - A relatively complete range of accident sequences has been considered in the high B/U, MOX source term study. Having said this, it is noted that the SOARCA project, which is due to be released around mid-2011, is addressing epistemic uncertainties (at considerable effort and expense). It is urged that the high B/U, MOX source term project determine where information from the SOARCA uncertainty analysis is applicable to and could be used to strengthen the high B/U, MOX source term work.

Sandia Response

This could be done. The two projects are, however, on different schedules for completion. It may not be possible to incorporate completely all the latest findings from SOARCA. Recognize that what is developed here is not a definitive source term for severe reactor accidents. It is a representative source term. The adaption into the regulatory process of the analogous source term described in NUREG–1465 is used in more of a “screening” sense. If licensees or applicants find the source term acceptable, they can be confident that NRC staff will accept it. If licensees or applicants do not find the source term acceptable, they are encouraged to develop and justify a source term suitable for the expected accident phenomena at their particular plant.

Evaluation

It is understood that the two projects are on different schedules and that it may not be possible to incorporate completely all the latest findings from SOARCA. However, SOARCA results to date can be used, at least to some extent, to inform the high B/U, MOX source term study. Doing this would provide support for saying that while a systematic quantification of uncertainty

was not performed, much work, including SOARCA and source term research such as that cited in the response to comment a, has been done over the years to address epistemic uncertainty such that no surprises are expected. It is suggested that the documentation (for example, SAND2011-XXXX, Section 2.3, "No attempt has been made to systematically quantify the state of knowledge uncertainty…") be amended along these lines.

Question 1c - A final thought with regard to the aleatory uncertainty analysis is that including it is OK but the report should not emphasize it too much. It should be clearly stated why this analysis was done and how the uncertainty results are used, e.g., the uncertainty results provide confidence that the process for locating the medians is reasonably robust. But there is more detail in the report than is necessary on the analysis itself, or perhaps this detail could be located in an appendix since it is not really central to the results.

Sandia Response

A consistent comment from peer reviewers is that the uncertainty in the locations of the medians is overemphasized in the document. The uncertainties in the locations of the medians had to be evaluated to make meaningful comparisons between LBU and HBU fuels and between MOX and LEU cores. It is clear that the report has to be amended to make the comparisons without overemphasis of the uncertainty in median location or misleading interpretations of the uncertainty.

Evaluation

Agree.

Question 2 - One of the challenges of SAND2011-0128 is to defend the sequences selected and to defend the manner in which uncertainties are addressed. In this regard, it would be appropriate to explicitly discuss in some detail in the various reports the expected effect of operator actions (severe accident management and 50.54 (hh) mitigation actions) on release timing, release magnitude, and termination of core damage with the core in-vessel. This is not a suggestion to credit operator actions in the sequences, but rather to discuss the notion that operator actions would compensate for uncertainties in the accident sequence selection and calculation results and would tend to make the results reported in the draft documents conservative.

Sandia Response

The draft document will be amended to make clear that operator actions were not credited in the calculations. It would be incredible that operators would not take measures that would mitigate releases in some way and consequently credible accidents would have source terms that did not exceed those calculated. More detailed discussions are probably beyond the scope of work. Operator actions during accidents that go beyond the design basis are peculiar to each plant and, in many cases, the procedures are proprietary. It is difficult, then, to see how much specific information could be included in a public document.

Evaluation

It is agreed that more detailed discussions are not appropriate. The information on operator actions to be provided in the documents can be at a relatively high level, can be generic, i.e., the existence and expected effect of symptom-based EOPs for prevention, SAMGs , and B.5.b equipment and procedures as mandated by the regulations, and can be used to compensate for uncertainties and to defend the overall conservatism of source term results. The need for information on and some discussion of operator actions (on a generic basis) is even more important in the aftermath of the Fukushima accident.

Question 3 - A number of places in SAND2008-6664 indicates that for the MELCOR calculation matrix, sequences were selected based on risk-significance (see pages 35, 36, 39, and 53). NUREG–1465, however, states that it emphasizes likelihood (see NUREG–1465, page 4). Since SAND2011-0128 is looking at release into containment (i.e., containment is assumed intact for purposes of the DBA analysis), it would make more sense to select core damage sequences based on likelihood as opposed to risk which includes both likelihood and consequences. SAND2011-XXXX appears to indicate that sequence likelihood is used as the basis for selection (see SAND2011-XXXX, page 22, first paragraph, and page 23, first paragraph).

Sandia Response

Accidents were selected based on an evaluation of the core damage frequency – a metric for likelihood. Comments to the effect that risk-significance was the basis are purely sloppy nomenclature and reflect the common and incorrect identification of CDF as a measure of risk, which it is not.

Evaluation

Agree with response. It is assumed that the documentation will be globally corrected to eliminate statements that risk-significance was the basis for sequence selection.

Question 4 - The discussion regarding what burnup is being addressed and why is confusing and inconsistent, and needs to be tightened up. I have three comments on this:

Question 4a - SAND2011-XXXX states that the primary objective of the report is to develop accident source terms (i.e., core melt source terms) for reactors with burnup in excess of 40 GWD/t. But NUREG–1465 indicates that the concern regarding high burnup is for RIA accidents and gap activity. Further explanation as to the objective is necessary here.

Sandia Response

Motivation for the study was prompted not by limitations identified in NUREG–1465 but by the need to confirm adequacy of decisions that regulatory staff had made concerning the use of fuel to burnups in excess of those considered in NUREG–1465. It is also true substantial progress has been made in the understanding and modeling of severe reactor accidents over the two decades that have elapsed since NUREG–1465 was drafted. Regulatory authorities are

obligated by both policy and the Atomic Energy Act to assure the technical adequacy and defensibility of regulatory guidance. Regulatory authorities were interested in knowing if progress in understanding severe accidents would lead to changes, qualitative or quantitative, in accident source terms. The document will be amended to more carefully state the objectives.

Evaluation

Agree with response to amend document to more carefully state the objectives.

Question 4b - RG 1.183 states that the release fractions in Tables 1 and 2 are acceptable for peak burnup up to 62 GWD/t. So why must the high B/U, MOX project justify burnups up to 62 GWD/t when the NRC's main accident source term guidance already says that the licensee can go up to 62 GWD/t?

Sandia Response

NRC made the decision to allow burnups up to 62 GWD/t based on engineering judgment. The project was undertaken, in part, as confirmatory research for this judgment.

Evaluation

It is suggested that the response be incorporated as part of the restatement of the objectives.

Question 4c - SAND2008-6664 adds to the confusion with the statements on pages 16 and 31 that the regulatory limit is 62 GWD/t average for the maximum assembly (whereas the regulations say 62 GWD/t for the peak rod).

Sandia Response

The statement in SAND2008-6664 is in error.

Evaluation

OK

Question 5 - There has been much discussion in the peer review group on use of the median vs. the mean. Generally, I agree with use of the median as explained on page 18 of SAND2011-XXXX, and use of the median appears to be slightly conservative relative to the mean, at least for iodine (which is the most important contributor to dose) for both release magnitude (median > mean) and release timing (median < mean). It should also be noted that NUREG–1465 used the lesser of mean and 75% and the high B/U, MOX study is using median which is greater than the mean.

Sandia Response

The document will be amended to include the description of what was done for NUREG–1465 along with an indication that distributions of results considered in NUREG–1465 differed in bases than those considered here.

Evaluation

Agree with response with the additional suggestion that the response indicate that while the treatment of distributions was different in NUREG–1465 vs. the high B/U, MOX source term, the two treatments both used a central estimate process which in the end was not significantly different.

Question 6 - The matter of release rates is worth some discussion. NUREG–1465 assumed constant release rate. In addressing release rate, Section 4.4 of SAND2011-XXXX includes two figures that show calculated in-vessel release rate with higher short term aerosol mass in containment compared to constant in-vessel release rate.

Question 6a - The accident sequence type that was the basis for these figures was not stated, though it appears to be rather fast, typical of a larger-sized LOCA.

Sandia Response

The sequence was in fact a relatively slow station blackout sequence with core melting spread over 8 hours. The sequence was chosen simply to illustrate the gross effects that constant release rates and more realistic release rates for the in-vessel release phase would have on concentrations in containment. No decision has been made to abandon constant release rates. This is one of the questions that have arisen that would benefit from the judgments of the experts.

Evaluation

The SAND2011-XXXX, Figure 4 and 5 plots look more like large LOCA to me as opposed to slow SBO (i.e., based on the red-dashed, calculated in-vessel release rate, the release starts at ~0.5 hr and ends at ~2 hr which is not a slow SBO). The document should state what sequence is the basis for these figures and perhaps have similar plots for a slower, more probable sequence. My current thought is that constant release rates may be more representative of the spectrum of higher probability accidents, though NRC may want to retain the option of plant-specific release rates, esp. SMRs as discussed in next comment. Another reason for using constant release rates is that specifying a front loaded release could drive the design toward a containment fission product mitigation system that is limited in duration and mitigates the front portion of the release but is not functioning later when a significant portion of the release may actually occur. (Note that this was the case for the AP600 passive, accumulator-driven spray system which had capacity for 30 minutes and with the TID-14844 source term (instantaneous release into containment at time zero) would have been designed to actuate at time zero, thus missing the real accident release.)

Generally, it appears to me that for the range of accident sequences being considered, there is no definitive trend toward high release rates early in the accident. The basis for this is that the fuel heatup and oxidation process is incoherent, and other than medium to large sized LOCAs, which are relatively low in probability, the release to containment is relatively constant, possibly with a spike late due to lower head failure. Figure 20 of SAND2008-6664 (which is for a Sequoyah SBO with no AFW), illustrates this behavior.

Question 6b - There would be merit in allowing flexibility in release rate for SMRs. For example, at the bottom of page 50 of SAND2011-XXXX, a sentence could be added indicating that for non-LWRs and SMRs the licensee could propose design-specific release rates if not too complicated (e.g., no more than two constant release rate segments).

Sandia Response

The document will be amended to foreclose application of the proposed source terms to SMRs. To date, no SMR has been submitted for licensing, so definitive design data are not available. Indeed, it appears that there is considerable evolution in SMR designs taking place now. Accident source term guidance that encourages applicants to develop their own source term appropriate for new designs will still be in place.

Evaluation

Agree with response although I would say it a little differently, i.e., not to “foreclose” application of the proposed source term to SMRs, but rather to justify use of the proposed source terms or develop their own.

Question 7 - With regard to gap fraction, there appears to be justification based on the high B/U, MOX work, to eliminate a separate Gap Release Phase. However, if the separate gap phase is eliminated, I remain concerned as to what the basis for Table 3 of RG 1.183 will be. While it is not explicitly stated in RG 1.183, NUREG 1465 gap releases support Table 3. Particularly for FHA, RG 1.183, Table 3 or equivalent is needed.

Sandia Response

No decision has been made to eliminate the Gap Release Phase. It is hoped that the experts will comment on this question. A decision to eliminate the Gap Release Phase imposes no requirements on RG 1.183. There would be, no doubt, some considerable discussion of developing bases for tables in the regulatory guide.

Evaluation

It is agreed that eliminating gap release imposes no requirements on RG 1.183, although precedent suggests that RG 1.183 will follow what is in the high B/U, MOX documents. My thoughts are that the high B/U, MOX work provides a basis for not having a separate gap release phase, and that this makes sense since it simplifies application of the proposed accident source term by the licensee and simplifies NRC review. A separate effort may be necessary to

address release from reactivity induced accidents (neither NUREG–1465 nor the high B/U, MOX work address reactivity induced accidents).

Question 8 - While the high B/U, MOX work is generally well-documented, the documentation can be improved in certain places. See, for example, comments 15, 16, 17, 18, 20, 21, and 25 below. This is important to assure transparency and completeness.

Sandia Response

The suggestions for improved documentation are appreciated very much. See responses for the cited comments. Transparency and completeness are important for this work.

Evaluation

Agree with response.

Question 9 - The MOX core will be less than half MOX fuel assemblies, and high burnup cores are only about 1/3 high burnup assemblies. This may have been discussed but I couldn't find it. Does MELCOR automatically select the applicable fission product release model or allow the user to specify the applicable model on an assembly-by-assembly basis, e.g., specify HBU Booth for high burnup assemblies and modified ORNL Booth for LEU low burnup assemblies?

Sandia Response

Yes. The version of MELCOR used for the analysis allows the user to specify separate HBU, LBU, MOX and LEU Booth model fission produce release parameters. The mass of each fuel inventory (e.g., LBU and MOX) is specified for each core cell in the model. The release models are applied by the code to their respective fuel inventories.

Evaluation

OK, thank-you for the information.

Questions 10 to 34 are more specific, addressing a particular page or subsection of one of the reports as noted.

Specific Comments - SAND2008-6664, "Accident Source Terms for PWRs with High B/U Cores Calculated with MELCOR 1.8.5"

Question 10 - Page 15 – The report states that the authors of NUREG–1465 "…restricted application of the source term to hypothesized accidents at currently operating light water reactors. They encouraged designers of advanced reactors to use similar methods to develop source terms applicable to their novel designs." This is not quite correct. NUREG–1465 (page 4) states that because of general similarities in plant and core design parameters, NUREG–1465 results are also considered to be applicable to evolutionary LWR designs, and that despite lack of specific accident sequence information, the in-containment source terms may be considered generally applicable to the passive LWR designs. NUREG–1465 did restrict application for

designs “very different from LWRs, such as high-temperature gas-cooled reactors or liquid-metal reactors.”

Sandia Response

Agreed. The Preface of NUREG–1465 states, “Source terms for future reactors may differ from those presented in this report, which are based upon insights derived from current generation light-water reactors. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in this report, based upon and justified by design specific features.” Also under Limitations (Section 2.3) the source term is considered applicable to “…evolutionary LWR designs such as General Electric's Advanced Boiling Water Reactor (ABWR) and Combustion Engineering's (CE) System 80+” and “smaller LWRs employing some passive features for core cooling and containment heat removal”.

Evaluation

Agree with response, and I assume that page 15 of SAND2008-6664 will be amended accordingly.

Question 11 - Page 16 – What is the basis for the statement that, “It is anticipated that with changes in some plant features, including steam generators, PWRs may in the future be able to operate at substantially higher powers”?

Sandia Response

The statement is a reflection of widespread conventional wisdom and indeed preparatory work is underway at NRC to receive power uprate applications for PWRs. Many PWRs are limited in operating power by steam generator capacity. Others are limited by peak clad temperature. Best estimate analysis methods are allowed by regulation to permit power uprates for PWRs. Replacement steam generators with higher capacity may well permit other PWRs to power uprate.

Evaluation

OK, thank-you for the information.

Question 12 - Page 27 – “In this process, Eqs. 4 and 5 are…” Should that be equations 2 and 3?

Sandia Response

Yes.

Evaluation

OK.

Question 13 - Page 32 – The report states that nonradioactive aerosols can be important because of the effect of agglomeration and settling onto RCS structures. It should be added that this affects settling rates in containment as well.

Sandia Response

The comment is accurate. But, the focus of the discussion is on transport of radionuclides through the RCS to the containment. Behavior of radionuclides in the containment is outside the scope of the work being reviewed though it is calculated by MELCOR in the supported accident analyses.

Evaluation

Agree.

Question 14 - Page 35 – For clarification, in lines 3 and 4, and the first line below Table 9, change the phrase “revised source term” to “AST”.

Sandia Response

Agreed. AST should have been used instead of “revised source term”..

Evaluation

OK.

Question 15 - The first paragraph of Section 6.0 is confusing, e.g., second sentence – why “equally probable” as opposed to just random samples from a distribution of values; third sentence – why risk-significant as opposed to more likely; next to last sentence – does “populated’ refer to the sequences selected or to the development of data on a given sequence? Maybe this paragraph could be simplified to just state that the display of results is an approximation of the cumulative distribution that would otherwise have been generated if there were a large set of sequences considered.

Sandia Response

Yes. If the observations were not (assumed to be) equally probable, then the observations would have to be weighted by their respective probabilities in order for them to “…approximate the cumulative distribution function for the population of accidents.”

Accidents were selected based on an evaluation of the core damage frequency – a metric for likelihood. Comments to the effect that risk-significance was the basis are purely sloppy nomenclature and reflect the common and incorrect identification of CDF as a measure of risk, which it is not. “Populated” refers to sequences selected.

Evaluation

Agree with response. The paragraph needs to be amended to clean up the nomenclature on CDF vs. risk.

Question 16 - Perhaps I'm missing something, but I had trouble going from the tables in Section 5 to the plots in Section 6. For example, the points in Figure 18 (PWR in-vessel iodine release fraction) do not appear to have come from the halogen column of Table 22

Sandia Response

The plots in Section 6 are cumulative distributions done using an earlier version of the nonparametric methods adopted for the work under review. Current methods refine these plots quite a bit.

Evaluation

See Question 25 and associated Evaluation of Response.

Question 17 - Some examples would help with regard to going from tables in Section 5 to plots in Section 6.

Sandia Response

It might be possible to provide in a supplemental support document a pedagogic elucidation of the nonparametric methods.

Evaluation

This would be useful, perhaps in an appendix.

Question 18 - I also had trouble reconciling the data in Appendix C with the tables in Section 5. For example, in Sequoyah 4A HBU, Appendix C says coolant release begins at 5.6 hr whereas Table 13 says 0.8667 hr. Same comment for 4B, 4C, and 4D. 4E is closer, i.e., 1.8 hr in Table 13 vs. 2.0 hr in Appendix C. Same comment applies to other Section 5 tables. The correspondence between Section 5 and Appendix C is better for the Surry sequences although some of the Section 5 early in-vessel durations for Surry don't match up with Appendix C, e.g., C.8, Surry, Case 1A HBU, begin early release phase at 2.9 hr, vessel failure at 6.1 hr, thus duration of in-vessel release is 6.1 – 2.9 = 3.2 hr; this does not agree with Table 16, Case 1A HBU which has duration of in-vessel release phase of 3.9 hr.

Sandia Response

The reviewer has correctly identified an error. Tables in Section 5 are correct. They were amended because of incorrect edits in MELCOR output to obtain the needed source term characteristics. These incorrect determinations were used to make the Appendix C tables which

were not subsequently corrected. Errata for appendix C will either be included in an additional support document or a revision made to SAND2008-6664.

Evaluation

OK.

Question 19 - What is the breakdown of Cs release between CsI, Cs_2MoO_4, and CsOH for the different sequences?

Sandia Response

The process used to allocate cesium among the chemical classes is the same for all accident sequences. Table A-11 documents the breakdown -

- Stoichiometrically combine all iodine with cesium and place in Class 16 [CsI].
- Place 5% of CsI in the fuel gap. This represents 5% of the Halogen inventory, but a much smaller fraction of the Alkali Metal inventory.
- Determine the quantity of Cs required in addition to that represented by CsI in the gap (above) to reach a total or 5% of the core inventory of Alkali Metal. Place this additional mass in Class 2 [CsOH] and position the entire Class 2 inventory in the fuel gap.
- The quantity of remaining Cs (95% of the core inventory) should be of a sufficient quantity to completely react with a fraction of the core inventory of Mo to form Cs2MoO4. Place this mass and the stoichiometric fraction of Mo inventory in Class 17.
- Place all remaining (excess) Mo in Class 7.

Evaluation

OK, thank-you for the information.

Question 20 - Appendix D [in SAND2008-6664] should indicate which Surry sequence is being presented (1A or 1D). Also, several of the figures only have one set of curves (LBU or HBU?). Why not both?

Sandia Response

The plots in Appendix D are for the HBU Surry SBO, not an AFW sequence (Case 1A). The text should have stated this.

Evaluation

I am still confused about this. The Appendix D sequence appears to be neither 1A nor 1D. For example, the Appendix D sequence shows accumulator injection beginning at 12 hours whereas

1A and 1D have accumulator injection starting in the range of 3 to 5 hours. Also, the opening paragraph of Appendix D states that there are plots for both low burnup and high burnup calculations, but the graphs only have one plot.

Question 21 - Appendix D-type data [See Appendix D in SAND2008-6664] for all sequences would be useful. Similarly, including figures corresponding to Figures 20 and 21[in SAND2008-6664] for all sequences would be useful and would more fully document the work done to support the source term.

Sandia Response

It may be useful to include such information in a supplemental support document. The logistics of doing this does become outsized quickly since there are a total of 88 accident analyses. A few plots for each accident analysis would quickly total into a monumental effort. It is of some importance then to understand why the results are needed.

Evaluation

I understand the point on outsized logistics for the large number of sequences. However, there are two reasons I am suggesting that at least some of these results are needed. First is that these results will provide confidence in the proposed accident source term release parameters since it allows the user to see where the results come from (note the difficulty in reconciling Appendix D with Appendix C, and in reconciling the Appendix C and D data with the tables and figures of Sections 5 and 6). Second is the fact that these results may be useful in the future, e.g., for applicants who desire to propose alternatives to aspects of the regulatory guidance or for SMR designers who may have to justify their own source term. Note that this NRC effort to upgrade accident source terms is probably a once every 10 to 20 year exercise and if we don't capture this type of information now it could be lost forever to users.

Additional comment - Section 4, next to last paragraph, says that sequence 1C "… is screened from consideration," yet this sequence still appears in the set of sequences analyzed.

Specific Comments - SAND2011-XXXX, "Accident Source Terms for LWR Nuclear Power Plants Using High B/U or MOX Fuel"

Question 22 - Instead of weighting each of the accident analysis results equally (see page 18), would there be value in recognizing the relative probability of the various sequences being considered? For example, some type of weighting could be assigned to a sequence according to its fraction of the sum of the probabilities of the matrix of sequences.

Sandia Response

There is, of course, a distribution of probabilities for various classes of accident sequences (SBO, LOCA, ATWS, transient, etc.) for various classes of reactors. The level of aggregation adopted for this work is, however, BWR and PWR. Aggregation at this level based on results reported in the IPE Insights document [NUREG–1560] shows that any defensible level of

weighting would be about equal for each class with the possible exception of ATWS. (See slides that follow.)

Evaluation

Agree.

Question 23 - It would be clearer if the number of data points in Figure 1 (15) and Figure 2 (also 15) was the same as the number of sequences in the MELCOR calculation matrix (16 per Table 4-56 of SAND2008-6665). This comment applies to many of the figures in SAND2011-0128 and SAND2008-6664.

Sandia Response

All sequences were used in the analysis. Not all points plotted in Figures 1 and 2 show because source terms for two of the accident sequences are identical through the in-vessel release phase of the accident. These sequences do not diverge until the onset of the Ex-vessel Release Phase and the Late In-vessel Release Phase. The captions of Figures 1 and 2 can be augmented to make this point clear.

Evaluation

Agree with response.

Question 24 - Page 33, second full paragraph – reword to say, "There is a distinct phenomenological and temporal difference…"

Sandia Response

The suggested change has been made.

Evaluation

OK.

Question 25 - I had trouble confirming the Table 11 values (i.e., median PWR HBU source term parameters) [in SAND2006-6665] with SAND2008-6664 values. For example, for median duration of in-vessel release for PWRs, HBU, SAND2008-6664, Table 16 shows about 4.0, SAND2008-6664, Figure 15 shows about 2.9, and Table 11 shows 4.5 hr

Sandia Response

The values in SAND2008-6664 do not directly correspond to the values in Table 11. Rather, the values in Table 11 are the medians of cumulative distribution functions derived from the values in SAND2008-6664.

For example, for HBU, the median of the duration of the In-Vessel Release Phase (Table 16 in SAND2008-6664) is 3.9 hr, while the corresponding value in Table 11 (4.5 hr) is the median of cumulative distribution function derived from the Table 16 values**.**

Evaluation

There was a further interchange with Sandia on this via email. This email exchange documented the fact that the Table 11 values do not match up with the SAND2008-6664 Section 6 figures which are the cumulative distributions for the tables in Section 5. For example, SAND2008-6664, Figure 15 has the HBU median PWR duration of in-vessel release as about 2.9 hours whereas Table 11 shows 4.5 hours. The 4.5 hour value looks more like a 75th percentile value on SAND2008-6664 Figure 15. Sandia indicated that the SAND2008-6664 values are incorrect (and out of date), that these values had not been used in SAND2011-0128, and that SAND2008-6664 would be withdrawn and reworked with new numbers that are correct and were actually used.

Question 26 - Page 45 – What is the form of Te that does not interact with unoxidized cladding?

Sandia Response

Chemical forms adopted by tellurium under accident conditions are not well known. This is in part due to the relatively unstable thermochemical data base for tellurium and its compounds which has not been critically and comprehensively reviewed since the publication by Mills in the early 1970's. For some time it was thought that the interaction with cladding was really with the tin alloying agent. The volatility of the species SnTe seemed to be consistent with release observations made in the PBFA experiments. Experiments by Elrick et al. showed however that SnTe was not stable to dissociation over nickel-rich alloys. Such dissociation was not observed in the Phébus tests using heated Inconel lines for transport pathways. Thermochemical analyses have focused on tellurium oxides and even cesium tellurates. It is hoped that further insights may be gained from tests in the forthcoming STEM tests. It would also be of use to re-examine the thermochemical data base for tellurium. The remarkable affinity of tellurium for palladium and the rich nodules of palladium alloys in fuels with elevated burnup needs to be resolved.

Evaluation

OK, thank-you for the information.

Specific Comments - SAND2010-1633, "Accident Source Terms for LWR Nuclear Power Plants Using High B/U or MOX Fuel"

Question 27 - Some explanation of the basis for the inverse of the sum of the reciprocals in equation 8 would be helpful.

Sandia Response

The mass transport rates are determined by transport through the fuel grains followed by transport through pore structure of the fuel. These processes are serially connected. The respective rates are analogous to electrical conductances. Given this analog, the total mass transport conductance is equal to the diffusion and gas-phase mass transport conductances in series, which is equal to the sum of the reciprocals of the diffusion and gas-phase mass transport conductances.

Therefore, the diffusion mass transport rate is expressed as the inverse of the difference of the total mass transport rate and the gas-phase mass transport rate.

Evaluation

OK, thank-you for the information. This would be helpful if it were included in the report.

Question 28 - The first paragraph in Section 3.3 states that for VERCORS 4 data on release of fission products in addition to Cs were available. But Tables 3 to 5 show experimental values for fission products other than Cs for tests ORNL VI-2, VI-3, and VI-5.

Sandia Response

The reviewer is correct; the table with the VERCORS 4 data was inadvertently omitted from the document.

Evaluation

OK. Need to correct text in addition to adding VERCORS 4 table.

Question 29 - Page 31, bottom paragraph, refers to VERCORS test 5 in two places. Should this not be test 4?

Sandia Response

Yes, it should be “test 4”.

Evaluation

OK.

Question 30 - Page 43 cites Lorenz and Osborne [15]. Should this not be [11]?

Sandia Response

Yes. It should be Lorenz and Osborne [11].

Evaluation

OK.

Question 31 - Some additional detail would be helpful regarding the process of going from the experimental data for f (and thus D) to D_0 and Q.

Sandia Response

The additional detail requested is to be found in "A Summary of ORNL Fission Product Release Tests with Recommended Release Rates and Diffusion Coefficients", NUREG/CR–6261; ORNL/TM-12801, 1995, as well as in the documentation of the MELCOR computer code.

Evaluation

OK.

Question 32 - Table 7 grain radius should be 6 μm, not 6 m.

Sandia Response

Agree.

Evaluation

OK.

Question 33 - Figure 37 should say HBU-Booth for the red line, not HEU.

Sandia Response

Agree.

Evaluation

OK.

Question 34 - It would be helpful if the report stated what the burnup was for RT-6.

Sandia Response

The burnup for RT-6 was 70 GWd/t.

Evaluation

OK. This should be stated in the report.

Compilation of PWR Low Volatile Release Fractions[13] (from 2002 AST peer review)

Element[14]	VERCORS HT 1 (from fuel)	VERCORS HT 1 (from furnace)	VERCORS HT 1 (from TGT)	VERCORS HT 1 (from loop)	Phébus FPT-1[15]	TMI-1 (to contain.)	SFD 1-4 (from fuel)	Existing NUREG–1465[16]	Proposed Update to NUREG–1465[4]
Ru	0.08	0.002	0.002	0.000	0.005	0.005	0.00007	0.0025	0.0025
Rh								0.0025	**0.01**
Mo	0.49	0.35	0.26	0.09	0.23			0.0025	**0.2**
Tc					0.21			0.0025	**0.2**
Ce	0.05	0.0	0.0	0.0		0.0001	0.00014	0.0005	0.0005
Pu					0.00023	0.00003[17]	0.000002	0.0005	0.0005
Np	0.07	0.0	0.0	0.0	0.008			0.0005	**0.01**
La	0.08	0.001	0.0	0.0				0.0002	0.0002
Zr	0.0	0.0	0.0	0.0	0.00015			0.0002	0.0002
Nd					0.00004			0.0002	0.0002
Nb	0.09	0.003		0.0				0.0002	**0.002**
Pr								0.0002	0.0002
Y								0.0002	0.0002
Cm							0.000001	0.0002	0.0002
Am							0.000004	0.0002	0.0002

[13] Releases are fraction of core inventory.
[14] The elements listed are those which are included in the licensing dose calculation.
[15] Phebus releases are to containment (i.e., % transported through SG tube).
[16] Gap plus early in-vessel release which is intended to be representative or typical
[17] The TMI-1 Pu release to containment is based on scaling the TMI-1 Reactor Building (RB) Pu measurement (6E-6) by the ratio of the containment to RB measurement for Ce-144 (1E-4/2E-5 = 5)

B.6 Dr. Sudarshan Loyalka

Question 1 - In the expert panel report, ERI/NRC 02-202, "Accident Source Terms for Light – Water Nuclear Power Plants - High Burnup and Mixed Oxide Fuels," several areas were identified as High (7), Medium (4), and Low Priority (2). It appears that the present work has focused mainly on the two highest priority areas (use of some new release data, and use and some validation of the integrated code MELCOR against PHEBUS-FP). It will be useful and informative if the Sandia team could document the extent or level to which other areas have been addressed in this work, and what effect a lack of progress there and perhaps other areas may have on the reported estimations. A table or two summarizing Sandia observations would be very helpful.

Sandia Response

Some information is provided in the chart[s] that follow.

Progress Made in Research on Degradation of HBU and MOX Fuels

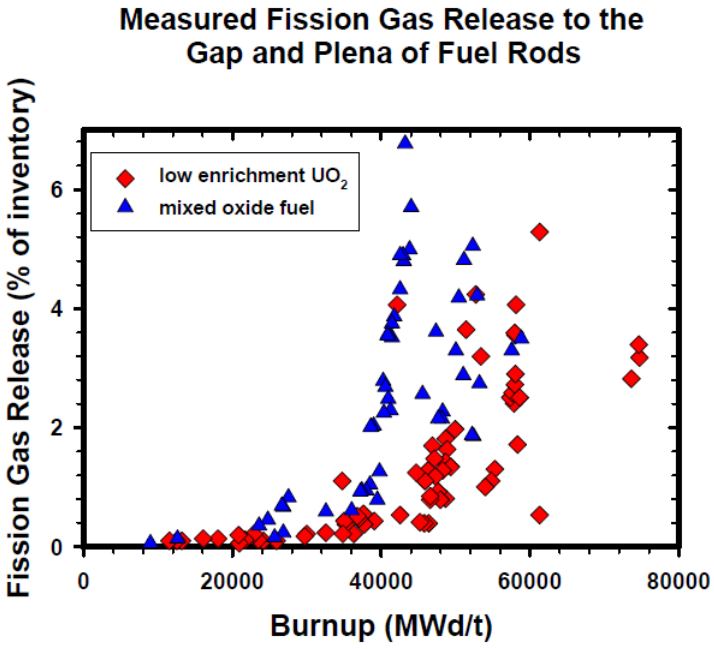
High Priority Recommendations - 1	Progress
Acquire available data on fission product releases from HBU & MOX	VERCORS data have been obtained and have been used. Further data are to be obtained in the VERDONE program.
Compare code predictions to data	VERCORS data have been used to modify MELCOR; further comparisons being made to PHÉBUS-FPT 1, 2, and 3; VERDONE tests will be used to refine release modeling.
Test air ingression effect on core degradation	Planned FPT-5 test to do this has been cancelled. Testing of air effects on release planned for VERDONE project. Much testing of air oxidation of cladding has been done.

Progress Made in Research on Degradation of HBU and MOX Fuels

High Priority Recommendations - 2	Progress
Tests of Core degradation with MOX	No tests planned; Matzke has measured oxygen potential; appears Mo/MoO_2 buffer keeps oxygen potential within range for LEU
Applicability of tests with reactor grade versus weapons grade MOX	No tests planned. No strong effect is expected. More important is the method of manufacture and clad type. LTAs showed unexpected growth.
Effects of Nb in clad and fuel foaming	No tests have been done or are planned.
Revaporization testing	Work completed in PHÉBUS-ISTP and planned for STEM

Progress Made in Research on Degradation of HBU and MOX Fuels

Medium Priority Recommendations - 1	Progress
Perform separate effects tests to resolve issue of Te release	PHÉBUS-FP clearly show much Te release despite Inconel piping. Empirically modeled now. Mechanistic understanding will require review of thermochemistry of Te. Some work will be done in STEM.
Use accidents other than LOCA to revise ST	Work done. Under review now.
Analyze risk significant sequences for MOX	Examined CDF significant sequences for MOX in Catawba plants. Other plants may use MOX in the future.
Tests of BWR fuel degradation	One PHÉBUS-FP test done with B_4C control rod. – results thought not applicable to US BWRs. Japanese have done 3 very large scale tests. Raw data will come available in 1-5 yrs.



Evaluation

Sandia has fully responded to this request, and the information provided has been most helpful.

Question 2 - With respect to the modeling of the release data, I had commented previously (01/27/2011) regarding some mathematical issues. I would like to see some greater documentation on exactly how the models are in fact being used in computations.

Sandia Response

Details of the modeling and how they are used in the codes can be found in the MELCOR documentation. Documents for version 1.8.5 are available. Revised documentation is being prepared for version 2.1.

Evaluation

Sandia has referred to MELCOR documentation. I was sufficiently interested in the issue so I decided to compute the release results independently. These calculations indicate that cumulative release rates (from the fuel) are being computed correctly (as per the model used) in MELCOR (judging from the figures in the Sandia reports/presentations).

Question 3 - I have some additional questions here with respect to extraction of diffusion coefficients from the VERCORS data. The equations (2)–(4), pages 33-34 in SAND2008-6665 (and similar equations in other related Sandia documents) are based on assumptions of a diffusion coefficient that is constant during the release (as the fuel is heated). But it seems to me that the fit is being made to data where the diffusion coefficient varied with temperature (and implicitly as a function of time as the fuel was heated). If I have interpreted the experiment correctly (and I may not have), then it would seem to me that the only proper way of extracting

the diffusion coefficient is by solving the diffusion equation with a temperature dependent diffusion coefficient, and then comparing the computed release fraction with the data (since the temperature profiles were measured, this can be accomplished without too much difficulty, I believe, and I am looking in to it a bit more).

Sandia Response

Gauntt extracted an effective diffusion coefficient using the methods that had been used in the past to analyze results from the hydrogen iodide and VI tests done at ORNL. The extraction is done by assuming temperature increases stepwise. The effective diffusion coefficient extracted in this method is not usefully interpreted as a physical quantity. The approximation done in the data reduction method and model use is one of the compromises made for systems level modeling – one that seems to work pretty well for analysis of PHÉBUS-FP tests.

Evaluation

Sandia has clarified how the diffusion coefficients were computed, and also the associated interpretations. I carried out independent numerical calculations in the manner I have indicated in the question and the attachment, and my results agree closely with those reported by Sandia. Thus, I have no issues with the response.

Question 4 - With respect to the validation of MELCOR against some PHEBUS data, it seems some adjustments in release models regarding the chemical forms (Cesium and Molybdenum) have been result informed. Thus it is essential to know what the overall parametric assumptions in MELCOR were, and to what extent these might have been result informed. I think it will be useful to know the values (used) of the parameters, and sensitivities of MELCOR results to variations in these values. It may be even more helpful if an input file for MELCOR concerning this validation could be provided for others to test (if no issues of any confidential or proprietary nature are involved).

Sandia response

There is some proprietary plant data in decks so a non-disclosure agreement would probably have to be executed to examine input files. MELCOR is not the subject of this review so sensitivities and the like have not been studied for this work. For this work, MELCOR was taken to be indicative of the current state of the art by definition.

Sandia has indicated that “There is some proprietary plant data in decks so a non-disclosure agreement would probably have to be executed to examine input files. MELCOR is not the subject of this review so sensitivities and the like have not been studied for this work. For this work, MELCOR was taken to be indicative of the current state of the art by definition. “

Evaluation

While I would have clearly liked to learn more, this response by Sandia is in accordance with the proprietary nature of data and constraints (parameters) for the review.

Question 5 - I recognize that the focus of this review is not MELCOR as such, but since MELCOR is the central tool for the analysis, it is important to have a good understanding of its strengths and limits. In particular, given the immense complexity of the problem, how is mass and heat balance being assured? How are phenomena at different time and spatial scales being modeled and numerics and results being reconciled? What models are the state of the art, and what have been frozen over the years and can be upgraded in view of the progress of last many years? Perhaps MELCOR developers already have a recent report(s) addressing such issues, and if so, it will be good for the review group to have a copy (or inclusion of a presentation at our next meeting in May could be given consideration).

Sandia Response

Mass and energy balance and code numerics are discussed in the MELCOR manuals. Models are discussed in these manuals. Models as they are now are considered for the purposes of this work to be the state-of-the-art for reactor accident analysis.

Sandia has indicated that, “Mass and energy balance and code numerics are discussed in the MELCOR manuals. Models are discussed in these manuals. Models as they are now are considered for the purposes of this work to be the state of the art for reactor accident analysis.”

Evaluation

I appreciate Sandia’s response given the parameters of the review. I think it is fair to state though that in certain areas some rate process models (for example, aerosol coagulation) have appeared in the literature that are improvements on models currently being used in MELCOR. These models have not necessarily undergone extensive testing, and I do not know though as to what impact they have would have on the overall results, and certainly any changes in the models/code structures are expensive undertakings. Yet such factors are part of all computations and will always remain so as our understandings and capabilities improve. Identification of such issues should be a continuing effort.

Question 6 - Regardless of the above, I wish to emphasize that in my view use of MELCOR is a major step forward beyond the use of STCP (NUREG–1465). Not only does MELCOR have better modeling in several areas that I am very familiar with, but its integrated nature makes it possible to both review and improve the modeling and estimations in a more organized and convenient fashion.

Sandia Response

Agreed.

Evaluation

Sandia agrees, and I again wish to emphasize that this remains my view.

APPENDIX C: INDIVIDUAL PEER REVIEW COMMITTEE MEMBER REPORTS

C.1 Dr. Bernard Clément

The present individual assessment is based on:

- The documentation provided to peer review committee members;
- The presentations and discussions during the two peer review committee meetings and the specific conference call meeting on the plutonium issue;
- The answers to questions and comments made by the reviewer.

A this point it is worth noticing that the technical exchanges and discussions with NRC and SNL staff have been very open and favoured a smooth review process, also greatly helped by Mohsen Khatib-Rahbar and other ERI people who were involved.

The present assessment report is organized in several sections. After the introduction, Section C.1.1 addresses the general approach including the selection of accident scenarios. Section C.1.2 deals with fission product release and transport including the experimental bases and their use for improvement and validation of the used simulation tool, i.e. MELCOR. Section C.1.3 provides some specific comments on the statistical treatment of the results. Sections C.1.4 to C.1.6 address the main outcomes of the draft report together with comments on the documentation.

C.1.1 Assessment of the General Approach

The proposed revision of the NUREG–1465 revised accident source term was motivated by the use of high burnup (HBU) LEU and MOX fuels in US Nuclear Power Plants. The proposed accident source term is intended to be representative of the source term to the containment for a class of Light Water Reactors accidents that involve substantial core meltdown. A selection of typical BWRs and PWRs was considered for HBU fuel and the Catawba reactor for MOX fuel. The accident sequences were selected from the Individual Plant Examinations (IPE) prepared by the licensees. sequences involving a substantial core meltdown. The selected sequences represent quite a large fraction of un-arrested sequences leading to core meltdown for different accident families (SBO, ATWS…). The selection procedure is considered to be consistent. More details were given in the presentations during the meetings than in the draft report;. The final version of the report should include some clear sentences about the approach that was used and make due reference to the IPE insights report NUREG–1560.

Whilst the selection of sequences is somehow conservative (sequences involving substantial core meltdown), the analysis of each individual sequence is best-estimate using a state-of-the-art calculation tool (MELCOR). The use of MELCOR makes a significant difference with previous studies based on NUREG–1150 using at that time the Source Term Code Package (STCP). Indeed, significant progress has been made in the knowledge of the progression of severe accidents through a number of R&D programmes and this progress is reflected in MELCOR. Again, the approach is well adapted to the purpose of the study. Although reviewing

MELCOR was not part of the mandate of the review committee, some specific points have been noted that deserve a special attention and are discussed in Section C.1.2.

The final objective of the report is to provide tables giving representative values of source term for different classes of radio-nuclides for different reactor types (PWRs and BWRs) and for different periods of time of the accident sequences (gap release, in-vessel, ex-vessel and late in-vessel). This is done through a statistical treatment of the distributions of the results for the different sequences. A single final value is given that is the median of the distribution. Considering the median as a representative value is considered as correct. More detailed comments are given in Section C.1.2. The main question, and this is not a statistical issue, is to define what is a "representative" value for the class of accidents considered. It should be made clear that the choice of accident sequences involving a substantial core meltdown induces some "conservatism" (although not being bounding), whereas the state-of-the art nature of the calculations and the statistical treatment point to "best-estimate" analyses. The reviewer has no problem with such a mix. It could be more clearly explained in the report.

Generally speaking, the general approach that is used, including the selection of accident scenarios, is considered as correct and fulfilling the objectives of the study.

C.1.2 Fission Product Release and Transport

The MELCOR models for the fission product release from the fuel have been adjusted using the results from VERCORS RT6 experiment for high burnup fuel and VERCORS RT2 for MOX fuel. The adjustment is obtained by fitting the cesium diffusion coefficient used in the Booth model to reproduce the measured release kinetics as a function of temperature. For other radionuclide classes the coefficient is scaled according to the total release. This is a standard procedure in MELCOR modeling. The choice of experiments for the fit is correct. In particular the VERCORS RT7 experiment on MOX fuel was not selected because it was performed under pure reducing conditions and so is less representative; as MELCOR does not take into account the influence of oxygen partial pressure on release, this is considered to be correct.

The chemical forms of cesium have been changed according to the Phébus FP results. Cs is split between CsI (stoichiometric value) and the less volatile Cs_2MoO_4. This new modeling was validated against the Phébus FPT-1 experiment. Note that putting part of Mo as Cs_2MoO_4 increases its release that is in agreement with the experimental results. These modifications, not related to the nature of the fuel (HBU or MOX) improved the modeling used for the present study.

The case of the cerium group calls for some remarks, not because of cerium itself but because of plutonium that is included in the cerium group of radionuclides. During the review process, a reviewer mentioned that the releases of Pu could be largely underestimated. After a review of the few available experimental data, it was concluded that this is the case, even if we take into account that most of the released Pu will be deposited near its emission point. However, it was also remarked that this deficiency does not affect too much the results of the study given the fact that the objective is to characterize the in-containment source term and not, for instance, radiological consequences of releases in the environment. It is therefore considered as

acceptable to keep the plutonium releases as they are, provided that the probable underestimation is clearly explained in the report.

The transport of fission product in the Reactor Cooling System (RCS) is mainly affected by the changes of repartition between different classes for Cs and Mo. The results of the changes were positively validated against the Phébus FPT-1 experiment. For the specific case of transport of iodine under a gaseous form, there is currently no model available and experimental programmes are still ongoing to acquire data on the issue. It was therefore decided to keep what was originally in the NUREG–1465 i.e., 5% of the released iodine is injected into the containment under a gaseous form. No distinction is made between organic and inorganic forms. This is considered to be acceptable.

We can conclude from this part that the experimental bases, for fission product release and transport, are well chosen and that MELCOR was well validated against the experimental results. This is not however the case for plutonium for which the release is underestimated. As mentioned above this would not impair the conclusions of the study.

C.1.3 Specific Comments on the Statistical Treatment

Non parametric order statistics are used, considering the results of calculations as samples from an unknown distribution. Instead of simply using the empirical distributions, cumulative distribution functions were developed and refined using a bootstrap re-sampling method. This would only be fully correct for random samples but can however be considered as acceptable given the relative weight of sequences derived from IPEs. Anyway it is an important progress as compared with the previous study.

The median value of the distribution has been selected as being the “representative” value for release phase durations and released amounts during each phase. It is true that using the median value provides more robust results as compared with using the mean value or high or low percentile values (e.g., 5% or 95%). This is understandable and acceptable on a statistical point of view.

The standard deviations on the median location of the different distributions are also given. From the discussions, it is understood that they are needed for making critical comparisons such as between results for LEU and MOX and to show where differences with results obtained in NUREG–1465 are significant. A better explanation in the final version of the report as compared to the draft is expected. Also, and as discussed, It would be very valuable to eliminate reference to aleatory uncertainty in favour of accident variability.

It was also mentioned in the answers to the questions that the plots of the distributions have been upgraded since the distribution of the draft report so that more realistic confidence intervals can be shown for the extremes of the distributions. This was done based on the Dvoretsky-Kiefer-Wolfwitz inequality and is fully supported.

C.1.4 Proposed PWR and BWR Release Quantities and Associated Characteristics

There is no phenomenological boundary to mark the end of the gap release phase and its definition is based on a criterion of 5% of the initial total core inventory. This is acceptable.

Apart from the specific case of plutonium, the released quantities for BWRs and PWRs (LEU and MOX) correspond to the state-of-the-art in source term evaluations and this fulfills the objectives of the study. The same stands for the durations of each release phase.

The statement made in the abstract *"Important differences among the accident source terms derived here and the NUREG–1465 Source Term are not attributable to either fuel burnup or use of MOX fuel. Rather, differences among the source terms are due predominantly to improved understanding of the physics of core meltdown accidents"* is consistent with the conclusions of the expert panel convened in 2001-2002. The fact that this conclusion is now supported by state-of-the-art calculations and not only by expert judgment demonstrates important progress.

C.1.5 Documentation of the Proposed Source Terms

The documentation is of good quality as well as the presentations made during the meetings of the review committee and the technical exchanges that followed. Some additional information was provided during the meetings and during the technical exchanges it was suggested to include part of it as an appendix to the report. This the case for selected graphs of the distribution functions. It was answered that *"this would be antithetical to the general aspiration that the document be succinct"*. It was proposed to issue an additional support document available to those with an interest in the details. This proposal is supported as it answers the original concern and as a support document can be useful in the future to find more easily where what is in the synthesis document is coming from.

The additional document could also include additional information about the progression of fuel degradation as calculated with MELCOR for the different accident sequences. Indeed it is quite difficult to evaluate how much the core meltdown is "substantial", as required, in the selected and calculated sequences. Synthetic tables indicating for instance the amount of molten materials/debris at the time of failure of the Reactor Pressure Vessel lower head would be helpful.

C.1.6 Features and Limitations of the Proposed Source Terms

The proposed source terms correspond to releases to the containment for accident sequences with substantial core meltdown. They are to be used for evaluation of the containment in the frame of a defense-in-depth approach. This must be clearly said in the introduction as some people might be tempted to use them for other purposes for which they are not necessarily well suited. This is for instance the case for the use of values of gap release for accidents without core meltdown.

The proposed source terms can be used for reactors using high burnup or MOX fuel. They also constitute an improvement of NUREG–1465 source terms for reactors using moderate burnup fuels.

C.1.7 Conclusions and Recommendations

Two main conclusion points can be made:

- The proposed revised source terms can be used confidently for high burnup and MOX fuels as the analyses that were made correspond to the state-of-the-art;
- The proposed revised source terms can also be used for moderate burnup fuels as they take into account the progress in knowledge since the issuing of NUREG–1465.

In seeking to improve the documentation, some recommendations were made during the review process and discussed. Most important ones are summarized hereafter, the rationale of which being explained in the main text:

- The final version of the report should include a more clear description on how the selection of accident scenarios was made with due reference to Individual Plant Evaluation (IPE) documentation;
- The likely underestimation of plutonium release should be pointed out in the final version of the report though it does not impair the general conclusions;
- A better explanation of the meaning of the standard deviations should be given in the report together with the suppression of the reference to aleatory uncertainties; and
- An additional support document including in particular more information about the fuel degradation in the selected accident sequences and graphs of the different distribution functions for fission product releases would be welcome.

C.1.8 Some Additional Comments Regarding Fukushima

It should be first understood that only limited data are available concerning the status of the reactors, especially concerning the location of corium (in and/or ex-vessel) and that the assessments done so far rely on the results of calculation codes the validation degree of which is probably limited for such situations.

Concerning the release of fission product and other radionuclides, the only reliable measurements come from samplings of the atmosphere, of water and ground deposits. These measurements can be compared with a priori and a posteriori predictions. A priori means that the releases to the environment can be evaluated assuming a degradation scenario, calculating release from fuel, deposition in the containment and retention in the wet-well during venting operations. A posteriori means using an atmospheric dispersion model to match the measurements by sampling. IRSN made the exercise and arrived to a figure of 3 10 16 Bq of cesium released from reactors 1 to 3 to the environment.

(see www.irsn.fr\EN\news\Documents\IRSN_fukushima-radioactivity-released-assessment-EN.pdf).

It must then be understood that assessments of Fukushima accident provide at this stage only indirect information for what is concerned by the proposed accident source term that is release to the containment and do not contradict the results of the study.

There were also some arguments concerning the release of gaseous iodine. One must be very cautious about the measurements by samplings as some results can hardly be explained (for instance similar radioactivity on the second charcoal then on the first one, indicating a lack of efficiency and/or pollution by aerosol particles and large amount of cesium quoted as gaseous).

One must be also very cautious when using the results of Phébus FPT-3 experiment for BWRs. Phébus FPT-3 used a B_4C control rod (B_4C pellets within a stainless steel cladding as in Russian VVERS and most recent French PWRs). The amount of stainless steel is much lower than in a BWR and, as a result, the amount of B_4C liquefied by molten stainless steel is limited and the remnant solid B_4C is oxidized by steam producing in particular a large amount of boric acid that is likely to react with cesium limiting thus the formation of cesium iodide and promoting the formation of gaseous iodine. This might have been one of the reasons of the measured fraction of gaseous iodine entering the containment (around 80%). On the contrary, in a BWR, the large amount of steel will dissolve most of the B_4C that will be drained in the cold lower parts of the reactors (see e.g. DF4 experiment at SNL). Very little B_4C oxidation products will then be able to react with cesium and to promote the formation of gaseous iodine.

Overall, neither observations at Fukushima nor Phébus FPT-3 results call for a change concerning the 5% fraction of gaseous iodine used in the proposed accident source term. Of course this could be reconsidered in the future when new information will become available.

C.2 Dr. Richard S. Denning

This report summarizes the results of a review of a planned revision to NUREG–1465, to consider changes that might be required to account for higher levels of burnup of fuel and for the use of mixed oxide fuel. Since the release of NUREG–1465 there has been additional experimental research that has provided changes in our understanding of source term phenomena that also make a revision timely. In addition to reviewing materials provided by ERI, the peer review committee met twice with representatives of Sandia National Laboratory and the U.S. Nuclear Regulatory Commission. We were also provided the opportunity to submit questions to obtain additional information. On January 27, 2011, I provided preliminary comments on the project and some questions. Following the second peer review meeting, I submitted additional questions, particularly related to the justification for the extremely small release of low volatility radionuclides in the in-vessel phase. On July 11, 2011, I provided an "Evaluation of Responses to Request for Additional Information." (See Appendix B, Section B.2) The preliminary comments and insights obtained from the requests for additional information are contained within this report (Section B.2).

I will first summarize two high level conclusions:

- There has been sufficient increased understanding in source term behavior that it is appropriate to make some changes in the NUREG–1465 source terms. Furthermore, the use of the MELCOR computer code for the analysis of a variety of characteristic severe accident scenarios provides a sounder technical basis than the expert elicitation based approach performed at the time NUREG–1465 was produced.

- On the basis of fractional inventory released, there is not sufficient difference between the release of low burnup, high burnup and mixed oxide fuel to warrant separate source terms.

C.2.1 General Approach

I strongly support the general approach taken in the development of an improved source term based on MELCOR computer code analyses of severe accident scenarios typical of those analyzed in risk assessments. The original NUREG–1465 severe accident source terms were based on the expert elicitations obtained in the NUREG–1150 risk assessment. Although the description of how the uncertainties were treated in NUREG–1465 is somewhat ambiguous, I traced the derivation back through the BNL support documentation. The principal source of uncertainty considered was the epistemic uncertainty implicit in the NUREG–1150 expert elicitations, from which the mean or 75th percentile was selected. The treatment of uncertainty in the revised source term is substantially different. Only the aleatory uncertainties associated with different severe accident scenarios are considered. To avoid confusion, it would be preferable to refer to the range of source term parameters obtained as variability rather than uncertainty. Because the regulatory objective is to use realistic or best-estimate severe accident source term values, I agree that the magnitude of the effort required to perform an epistemic uncertainty analysis was unwarranted. When NUREG–1465 was issued there was no alternative to reliance on expert judgment. The development and validation of MELCOR has provided a mechanistic basis for the revised source terms. Of course this places the burden of the peer review on assessing the ability of MELCOR to realistically estimate source terms, which in turn requires an evaluation of the state of validation of MELCOR within the scope of expected regulatory application.

The source terms have been divided into four phases - gap release, in-vessel release, ex-vessel release and late in-vessel release. Although there is historical basis for the definition of a gap release, the period of gap release is not clearly defined. Fuel pins fail at different times as melt progression occurs. In addition, the initial release that follows pin failure also depends on the time history of the fuel to that point. An arbitrary definition of gap release would lead to the possibility of misuse. I believe that the “gap release” should be removed from the revised source term. If there is a regulatory need to define a gap release for other reasons, such as to determine the release if a fuel element is dropped, a gap release should be developed that is appropriate to that application. It has been observed, however, that the in-vessel release typically occurs at a much higher rate early in the release period than at the end of the release period. An incomplete assessment of how the in-vessel release period could be divided into two time phases was presented to the peer review team. Subdividing the in-vessel release into two time periods appears to be a practical way to account for this variation of release rate. This approach should be pursued further.

C.2.2 Selection and Treatment of Scenarios

For the development of revised source terms, a number of scenarios were analyzed in different plant designs. The scenarios were recognized to cover a large fraction of the risk dominant scenarios for characteristic PWR and BWR plant designs. The different scenarios were effectively given the same weight in the analysis rather than weighting the different scenarios by their relative likelihood of occurrence or to their contribution to risk. This is appropriate, since the relative frequencies or contributions to risk are very design dependent. It was most important to provide a spectrum of conditions, which was accomplished.

For the revised source term, the median value of parameters was used to determine a central estimate from the range of values obtained for the different scenarios considered. The use of mean or median in circumstances of this type is always a topic of debate. The argument supporting the choice of the median value is appropriate for the current application.

The statistical approach taken to develop median values and ranges for the parameters that characterize the source term, e.g., a release time period or release fraction, appears to be more than adequate.

C.2.3 Proposed PWR and BWR Release Quantities and Associated Characteristics

It is important to recognize that the in-vessel release phase is the only phase considered in the most typical regulatory application, assessment of the site dose. Although the MELCOR computer code is “state of the art (as good as any other available tool)”, the accurate modeling of severe accident processes is a challenging problem. Some aspects of source term analysis can be analyzed with much greater accuracy than others. Because offsite doses tend to be dominated by the release of volatile radionuclides for which the fractional release from fuel tends to be high, the demand for modeling accuracy is not as great as would be required for regulatory issues that are dominated by the less volatile radionuclides. Furthermore, there is substantially greater validation data available for the amount of noble gases, iodine, cesium and tellurium released under molten fuel conditions than there is for involatiles, such as cerium or plutonium. From a regulatory perspective the upside uncertainty is bounded by 100% release. For a 0.5 release fraction that uncertainty is only a factor of two (which is certainly sufficient for the regulatory application). For an involatile radionuclide with a best-estimate release fraction of 1E-7, the upside uncertainty is dramatically larger.

Ability to model in-vessel release is better than the ability to model either ex-vessel release or late-in-vessel release. The ex-vessel release estimates for the revised source term are reasonably similar to the NUREG–1465 values. These releases are also not considered in the assessment of site doses. The late in-vessel release is also very difficult to assess in part because our ability to predict the deposition of radionuclides in the vessel is limited. Again this is not a consideration for the principal regulatory applications of the source term. For these reasons, in my assessment, I focused primarily on the in-vessel release fractions. In particular, I focused on aspects of the revised source terms that are different from NUREG–1465 and those aspects of the MELCOR analysis that might lead to an under-prediction of release.

Noble gas, iodine and cesium releases are not substantially different in the new source terms. The release of cesium is somewhat reduced by the new conclusion that the principal chemical form of cesium released (for the excess cesium that is not in the form CsI) is cesium molybdate rather than cesium hydroxide as assumed at the time of NUREG–1150. The effect is to slightly reduce the release of cesium but to increase the release of molybdenum. In the revised source term, molybdenum is now assigned its own group. Similarly, experimental evidence indicates that tellurium will not be chemically bound to the Zircaloy cladding, as assumed in NUREG–1150. The larger release of the tellurium group appears to be warranted. The ORNL-Booth models as modified to better represent the results of recent data have sufficient accuracy to support the regulatory use of the revised source terms.

The very small release of the involatile radionuclide groups, in particular the cerium group which includes plutonium, is the most striking difference between the revised source terms and the NUREG–1465 source terms. The difference is more than three orders of magnitude.

Table C.2.1 illustrates the changes in the new source terms versus the current NUREG–1465 source terms.

Table C.2.1 Comparison of Existing vs. Revised In-Vessel Cerium Release

	Table 12 BWR	Table 13 PWR	Table 16 MOX PWR
Proposed	1.4E-7	1.5E-7	1.0E-7
Existing	2E-4	5E-4	5E-4

In part, the difference is because in the original NUREG–1465 epistemic uncertainties were considered and the results represented either a mean or 75th percentile value. The proposed new source terms capture the variability that exists between different scenarios but are based on best estimate values rather than an explicit treatment of epistemic uncertainty.

I looked at the experimental database to determine whether there was evidence that the values predicted by MELCOR might be too low. There are some measured releases of cerium or plutonium in the various small scale experiments that yield fairly large releases under reducing conditions. The strongest evidence is probably in the VEGA experiments as described by Dr. Hidaka. In Figure C.1 (Figure 9 from the VEGA report), the release rates measure at 2128K (melting point of Zr; lower bound on fuel melting) is 1E-4 %/min and at 2800K (50-50 ZrO2/UO2 eutectic temperature, which was the fuel collapse temperature used in MELCOR calculations) is 1E-3 %/min. At either of these rates, it would take less than a minute to exceed 1E-7 release fraction. I realize that these tests are always distorted in some respect and a direct comparison is questionable but it is evidence of a potentially larger release mechanism than modeled in MELCOR. From NUREG/CR–6261, an ORNL summary of release experiments, in the VI series releases from fuel were observed as shown in Table C.2.2. Of these the VI-5 release is the most surprising. Nevertheless, the other releases shown are also substantially larger than 1E-7.

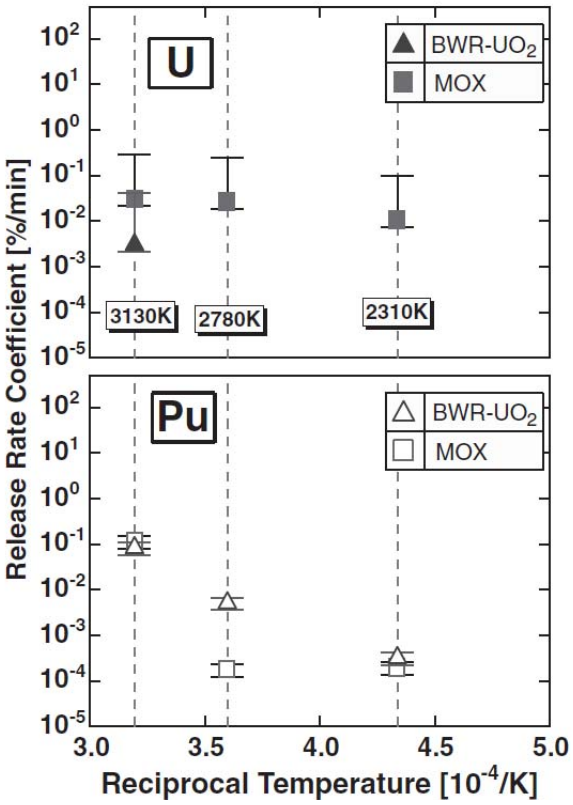


Figure C.1 Release rate coefficients U and Pu in VEGA-M1 and VEGA-8 tests

Table C.2.2 Release Fractions of Low Volatile Radionuclides in VI Test Series

Test	Ce	Pu
VI-2	-	3.1E-5
VI-3	<0.002	5E-5
VI-4	-	9E-4
VI-5	0.02	-

Dr. Clement provided evidence from the Phébus tests, as indicated in Table C.2.3. For VERCORS he relates that the Pu release from fuel is generally far below 1%. He also indicated that at IRSN it is generally assumed that only 10% of the amount released from the fuel would actually reach the containment. In the VI-5 test with the large cerium release a large fraction of the release was deposited in the near vicinity of the fuel.

Table C.2.3 Release Fractions of Low Volatile Radionuclides in Phébus Tests

Test	U	Pu	Zr
FPT1	1E-3	2E-4	1.5E-4
FPT2	3E-3	-	-

None of this is definitive evidence that under realistic core melting conditions, the releases of the low volatility groups would be as high as some of the values shown in Tables C.2-2 and C.2-3, but there are some conditions for which it is clear that releases substantially larger than 1E-7 are possible. I explored various aspects of the MELCOR modeling of fuel temperature and radionuclide release to determine if there were some apparent deficiency that might lead to an under-prediction of the release including the treatment of fuel liquefaction and slumping, temperature at which residual fuel would collapse, surface-to-mass ratio, and the partial pressure of cerium (or plutonium) as a function of temperature, which could affect the gas phase resistance in the release model. The value of the temperature used for the collapse of residual fuel rods is 2800K, which is higher than the current best estimate values based on VERCORS results. That would be expected to over-predict rather than under-predict the release. Similarly, the partial pressure of cerium as a function of temperature used in the model is somewhat higher than data available for plutonium oxide (over the range for which these data are available). A more detailed treatment of release of low volatility radionuclides could be developed but would require additional experimentation. In summary, it is quite possible that the values calculated by MELCOR are low, perhaps even by orders of magnitude, but the bulk of integral test results indicate that the contribution to release can be bounded. The contribution of the low volatility radionuclides to site dose and other expected applications of the source term is expected to be very small even accounting for the uncertainty. Nevertheless, this does imply that some care must be exercised in the use of the revised source terms.

C.2.4 Documentation of Proposed Source Terms

Consistent with the conclusions of the revised source term document, there is insufficient difference between the low burnup and high burnup UO_2 fuel source terms to warrant different tables. The differences between the PWR and BWR source terms are also small relative to the scenario variability and epistemic uncertainty. However, there is historic precedence to the use of different source terms.

The revised source term document also has different tables of source terms for MOX fuel and UO_2 fuel. However, for the same set of scenarios, the results for MOX and UO_2 fuel are virtually identical indicating that, at least within the capabilities of MELCOR, there is no discernible difference. Thus, the only reason for the difference is that a different set of scenarios, which is somewhat arbitrary, was selected to account for the source term variability. Only a single table of PWR source terms is warranted.

C.2.5 Limitations on the Use of the Proposed Source Terms

The applicability of the proposed source terms is clearly limited to light water reactors with UO_2 or MOX fuel in a zirconium alloy cladding. Whether the source terms could also be used for some Small Modular Reactors (SMR) that are LWRs is not totally clear. However, there is no reason to believe that MELCOR could not be applied in the same manner as used to generate the proposed revision to NUREG–1465 source terms to develop source terms that would be appropriate to the LWR SMRs.

Care must be exercised in applying the proposed revised source terms to regulatory issues outside the range of expected applications, such as for the assessment of site doses or for

equipment qualification. Applications that could be heavily influenced by Late In-Vessel Releases, Ex-vessel Release or by low-volatility radionuclide releases should recognize the substantially larger uncertainties associated with those aspects of the source term.

C.2.6 Observations Concerning Fukushima

Referring to the CSARP presentations concerning Fukushima, there is substantial potential for more gaseous iodine release that the 5% value in the proposed source term. I am not suggesting that the proposed source term move away from the 5% value but it might be appropriate to recognize this as an area in which there is still considerable uncertainty. I saw nothing else at the CSARP meeting that I thought would impact our understanding of source terms.

C.2.7 Conclusions and Recommendations

I conclude that the technical basis for the work presented for peer review is sufficient that the existing source terms in NUREG–1465 should be replaced by the revised source terms. The following list summarizes recommendations for the presentation of revised source terms:

- Clarify the difference in treatment of uncertainties in NUREG–1465 and the revised source terms (I think that the justification is the intent to use “best-estimate” results not the claim that the variability among scenarios is greater than the epistemic uncertainty).

- Refer to ranges of source term parameters as variability rather than uncertainty to avoid confusion.

- Do not include gap release as a phase of release.

- Pursue the potential to divide the in-vessel release into two phases.

- Qualify the release values obtained for low volatility groups (particularly the cerium group which contains plutonium) by indicating that these releases are sensitive to the modeling of severe accident progression, the uncertainties in the release models are large, and that the extent of model validation is limited. Nevertheless, indicate that the expected contribution of these groups in historic regulatory applications is expected to be small.

- Restrict the applicability to LWR fuel (UO_2 or mixed oxide) with zirconium alloy cladding and a maximum specified burnup.

C.3 Dr. Akihide Hidaka

C.3.1 Overall Evaluation

This study is considered to be excellent, useful and the challenging work under the condition of limited availability of experimental data on radionuclide release from HBU and MOX fuels during severe accident conditions. It is, however, recommended that the revised source terms be used as the state-of-the-art because the number of the experiments referred technically in this study was quite limited. It would be preferable that the representativeness and reproducibility of the test results be further examined in future.

C.3.2 Review of Radionuclide Release Model Used in this Study

(1) Positioning or role of FP release model in this study

In the present SNL's study [1], the revised source terms (numerical values) for HBU and MOX fuel in containment were proposed as many tables based on a lot of calculations with the MELCOR 1.8.5 code [2] which incorporated the new FP release models [3] (HBU-Booth and MOX-Booth) adjusted based on the IRSN's VERCORS RT-2 and RT-6 test results [4]. Therefore, it may be no exaggeration to say that the accuracy of the revised source terms depends mostly on the adequacy of the adjusted Booth models although the MELCOR 1.8.5 itself includes some uncertainties in FP transport calculations in the reactor coolant system. It should be noted that a part of the VERCORS test results were provided as the proprietary information.

(2) Fitting of the Booth model and its application

The ORNL-Booth model used in this study is basically empirical one which provides the best comparison with all of the currently available experimental data. The original Booth model [5] is the functions of two parameters, that is, diffusion coefficient (pre-exponential factor) and activation energy. For application of the Booth model to a source term code such as MELCOR, the class scale factors as shown in Table 1 on page 18 of the report SAND 2010-1633 [3] are also necessary to be defined in addition to the two parameters described above.

In the VEGA test at JAEA [6], the diffusion coefficient and activation energy were determined by using the least square method against a lot of diffusion coefficients (function of temperature) calculated from the measured increase in Cs release for a short time. After that, the class scale factors were also determined so that the fractional release of various FP groups at the end of experiment calculated by the Booth model using the diffusion coefficient and activation energy obtained from the Cs release data might agree well with the measured final ones, respectively. The reason for taking this method is because it is generally difficult to measure continuously the weak gamma rays emitted from FPs other than Cs which emits strong gamma rays during the temperature evolution.

This method is based on the following way of thinking. The transportation of FP molecule in the crystal lattice of the fuel grain depends mostly on the activation energy and there would be no large difference in the activation energy between volatile and low-volatile FPs. The difference in

volatility of various FP could be roughly expressed by the pre-exponential factor (diffusion coefficient) or the class scale factor.

(3) Necessity for adjustment of the class scale factor for the ORNL-Booth model

As the mechanisms for radionuclide release from MOX or BWR fuel under severe accidents, the following ones were proposed from the VEGA program.

(a) Vaporization from the peripheral region of the pellet at relatively low temperature (T<1500K - 1800K).

(b) Diffusion in UO_2 gain followed by diffusion in open pore (1500K - 1800K <T<2800K).

(c) Direct vaporization from surface of the pellet at high temperature (2800K<T).

In case of MOX fuel, the temperature near the pellet center during normal operation could increase compared with that of UO_2 fuel because lower thermal conductivity in MOX fuel. As a result, volatile radionuclide such as Cs could move from the pellet center to the peripheral region during normal operation. This could cause the increase in Cs release rate due to vaporization at relatively low temperature below 1500K - 1800K during severe accident conditions. However, semi- or low volatile radionuclide would remain at the original place in the pellet and their release rates at relatively low temperature would not be enhanced.

In case of the UO_2 fuel for PWR of which diameter is smaller than that of BWR, the fuel temperature at pellet center during normal operation could be roughly equal to 1000K and Cs could remain at the original place. As a result, the radionuclide release including Cs during severe accident conditions could be described well by the Booth model which assumes that the rate-determining step exists in the diffusion in grain because the release rates of radionuclide other than Cs are also determined by the amount, in other word, the volatility of radionuclide which reached the grain surface by diffusion in grain.

For the high temperature above 2800K, radionuclide such as Pu could be vaporized directly from the surface of the pellet in addition to the vaporization from the grain surface. This could also enhance the release rate compared with the Booth model prediction.

On the other hand, most of the source term studies including the present SNL study use the Booth model for simplification by ignoring the effects of (a) and (c). This expedient is accepted by a lot of researchers in the world including myself.

Going back to the class scale factor used in the ORNL-Booth model, the present peer review document describes as follows (First paragraph of 4.5.2 Release from fuel):

“The adjustment modifies the cesium diffusion coefficient in the Booth model to reproduce the measured release kinetics as a function of temperature. For other radionuclide classes, the coefficient is scaled according to their relative volatility.”

The word “their relative volatility” seems to be better than “total release” in such technical document but it is considered that the volatility used in the present ORNL-Booth model was determined actually from the total release measured in a lot of experiments in United States. Generally speaking, it is difficult to quantify the volatility because the radionuclide release is usually measured as a summation of (a), (b) and (c) mechanisms described above and therefore the class scale factor is not always proportional to the volatility which may be described by the vapor pressure and so on.

For example, in case of release from MOX fuel observed in VERCORS RT-6, it is considered that the Cs release was caused by and measured as summation of (a) and (b) mechanisms because the RT-6 results showed the increase in release rate at relative low temperature below 1500 K – 1800K compared with the previous U. S. experimental data in which the mechanisms (a) and (c) could be ignored. On the other hand, it is considered that the release rate of semi- or low volatile radionuclide in VERCORS RT-6 did not increase at relatively low temperature because the mechanism (a) can be ignored.

Accordingly, I believe that the class scale factor has to be redefined from the total releases measured in the VERCORS RT-2 and RT-6 tests because the ORNL-Booth model is just an expedient for simplification.

Since I have no right to access to the raw cesium release data of the VERCORS tests, I cannot confirm the accuracy of the newly revised models proposed by SNL described in SAND 2010-1633. I guess that the present SNL’s model cannot reproduce well the total release of radionuclide measured in VERCORS RT-2 and RT-6 tests except for Cesium release.

I would like to point out one more important thing. Since the class scale factor of the ORNL-Booth model was used as it is in the present SNL’s study, it is considered that the revised source term for HBU or MOX fuel calculated by MELCOR were not changed so much compared with those of the existing NUREG-1465. If the new calculations with the redefined class scale factor are performed, our conclusions may be changed slightly. At least, the features of source term with HBU or MOX fuel will be able to be identified.

(4) Present SNL method

A. It seems to me that the diffusion coefficient and activated energy used in this study were determined mostly by using the method described above. Although these two parameters are generally obtained by using the least square method, the present SNL’s method does not correspond necessarily to the least square method from what I can see Figure 37 on page 46 of the report SAND 2010-1633 [3].

B. In addition to A., there is a possibility that the class scale factors for FP groups other than Cs were not newly prepared based on the VERCORS test results in this study and the factors of the adjusted ORNL-Booth as shown in Table 1 on page 18 [3] were used as they are since there is no description in chapter 5 of the report SAND 2101-1633 [3].

(5) Effect of present SNL's method on the results

Concerning the issue A., there is a possibility that the FP release from HBU before in-vessel release phase 2 be slightly underestimated but the values in each 4 phase will not change so much. In other word, this issue may not affect the volatile FP release because almost 100% could have been released from the fuel during phase 2 but may affect surely the release results of semi- or low volatile FP release.

Concerning the issue B., there is a possibility that all of the FP release results other than Cs could be affected and the source term values in each phase could be changed. The tendency or trend of the proposed source terms in many Tables may be almost the same as the present ones but most of the proposed values could be forced to be changed slightly.

(6) Comments and/or opinion

If my understanding described above is correct, the present HBU-Booth and MOX-Booth models would have been prepared taking into account only the Cs releases but not taking into account the other radionuclide release measured in the VERCORS RT-2 and RT-6 tests. Since the VERCORS test of IRSN is one of the most reliable experiments in the world which provide the data on fission product release from fuel under severe accident conditions, it is recommended in this study that the method described in (2) (pre-exponential factor and activated energy obtained using the least square method, adjustment of the class scale factors) be used to make efficient use of the VERCORS RT-2 and RT-6 test results.

C.3.3 MELCOR Models on Radionuclide Transport in RCS

The RN package of the MELCOR code treats the radionuclide behavior including aerosol dynamics by using the MAEROS solution. The package considers condensation/evaporation, aerosol agglomeration and deposition such as gravitational settling, thermophoresis, diffusiophoresis and resuspention. These models have been well validated by many experiments and verified through the code comparisons [7].

C.3.4 Additional Comments

1) Fraction of organic or gaseous iodine in BWR containment atmosphere:

The FPT-3 test[8] of the PHEBUS program of IRSN[9] showed that use of B4C in control rods could result in increase in the fractions of organic and gaseous iodine in containment during severe accidents. Although all the data of the FPT-3 test have not yet been open the public, rough estimation assuming 100% release of iodine from fuel and no deposition onto the containment floor showed that the fractions of organic and gaseous iodine in BWR containment during severe accident could reach approximately 8% and 27%, respectively. The reason for increase in organic or gaseous iodine with the B4C control rods has not yet been clarified systematically but it is considered that the oxidation of boron could result in liberation of carbon and finally formation of organic iodine such as CH3I through the formation of CH4.

In the Fukushima Dai-ichi accident with B4C control rods, the ratio of gaseous iodine in the containment atmosphere was not included in the Japanese government reports [10-11] but the measured gaseous and particulate iodine concentration near the R/B of unit 2 was press released in June, three months after the accident, by the Nuclear and Industrial Safety Agency (NISA) for press release as shown in Table 2 [12]. The measurement showed that the ratio of gaseous I-131 is approximately 15% (=3.5E-05+1.8E-05)/3.5E-04 and those of gaseous Cs-134 and Cs-137 are 34% (=3.2E-04+2.0E-04)/1.6E-03 and 38% (=3.6E-04+2.1E-04)/1.5E-03, respectively.

In addition to that, the monitoring data at Tokai which is located at approximately 120km south of the Fukushima Dai-ich nuclear power plants, showed that the concentration of gaseous I-131 are almost the same as that of the particulate[13]. It seems to me that the assumption of gaseous iodine of 5% for BWR containment in the present SNL's report may be slightly underestimated according to the facts described above, although that for PWR containment is considered to be appropriate as it is.

The measurement also showed that the ratio of gaseous cesium was larger than that of iodine. The study on the gaseous cesium has not yet been sufficiently performed so far as long as I know. It is considered for me that the high concentration of gaseous cesium may become one of future issues.

Table C.3.1 Dust concentration at mountain side of R/B at unit 2 of Fukushima Dai-ichi site (Literal translation of Page 17 of reference [12]

Measured point	Fukushima Dai-ichi Unit 2, Mountain side, Outside of R/B				
Date	12:30 - 14:00 on May 26, 2011				
Nuclides (half life)	Sample concentration (Bq/cm^3)				
Filter type	1) Particulate	2) Wet filter paper	3) 1st Gaseous (charcoal)	4) 2nd Gaseous (charcoal)	1) +2) +3) +4)
I-131 (8 day)	6.4E-05	2.3E-04	3.5E-05	1.8E-05	3.5E-04
Cs-134 (2 yr)	4.9E-04	5.2E-04	3.4E-04	2.0E-04	1.6E-03
Cs-137 (30 yr)	4.9E-04	4.8E-04	3.6E-04	2.1E-04	1.5E-03

2) Fukushima Daiichi Accident

The Fukushima Dai-ichi accident took place on March 11, 2011 by long-term station blackout and loss of ultimate heat sink as the initiated events of earthquake and tsunami. The accident corresponds mostly to the long-term station blackout of BWR Mark-1 in Table 3 of chapter 3 in SNL's report. In Unit 1, the isolation condenser (IC) was available for about 1 hour just before the tsunami. In Unit 2, the RCIC was available for about 70 hours. In Unit 3, the RCIC in addition to HPI was available for about 40 hours. However, after the malfunction of IC, RCIC or HPI, all the three Units caused the severe accidents due to loss of the core cooling functions. Although the accident has not yet been terminated completely as of October and the detailed evaluation for the accident has just begun, I deeply hope that the various lessons learnt from this unfortunate accident could contribute to enhancement of accuracy of this alternate source term and eventually ensuring of the nuclear safety in the world.

C.3.5 Others

Dr. Rich Denning's comment "We have no clear idea of what accident was simulated in the VEGA tests or the transport pathway followed by radionuclides once they were released from the fuel."

Answer

In order to facilitate the comparison with the ORNL's test results[14], the similar temperature increase rates and plateau duration in the ORNL's were used in the VEGA tests. Concerning the transport pathway of Pu in the JAEA's THALES-2 calculation[15], it was calculated that most of Pu is transported through the broken hole of the RPV lower head to the containment or is directly released during the molten-core concrete interaction.

C.3.6 References

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C.4 Thomas Kress

C.4.1 Overall Assessment

My overall assessment is that the design basis source term to containment as determined by Sandia using MELCOR 1.8.5 is an appropriate replacement for the NUREG–1465 alternative source term for LWRs with low enriched fuel as well as for both high-burnup and MOX fuels.

This new proposed alternative design basis source term is much more realistic than that of NUREG–1465 in the timing, chemical forms, and release fractions. This realism can permit plants to optimize their responses and could very well result in improved safety at less cost. In order to preserve this increase in realism, I recommend that the timing of the fission product release no longer be given the artificial assumption of a constant rate. I recommend that the source term timing be specified in two constant release rate phases – an early rapid one to be followed by a later slower rate.

C.4.2 Fission Product Release

This study utilized the adjusted Booth models for fission product release from the fuel. These basically empirical models provide the best comparison with all of the currently available experimental data. These can, therefore, be considered to be the state-of-the-art. The Booth model is applied to cesium releases. The release fractions for the remaining fission product

groups are simply scaled to the cesium values based on experimental data. This relative volatility scaling works well for the various groups and, thus, avoids the overwhelming necessity to develop individual Booth models for each group.

The chemical forms of the fission products appear to be important only for iodine, cesium, molybdenum, ruthenium, and (perhaps) tellurium. There is not yet enough evidence to change the NUREG–1465 specified chemical form of iodine as 95% CsI and 5% I_2. On the other hand, there is ample evidence that cesium does not behave as CsOH as has been the assumption in past source term specifications.

Based on PHEBUS and VERCORS data, the behavior of Cs seems to be best represented as cesium molybdate which also further clarifies the behavior of molybdenum. Making this specification change for the new proposed source term is amply justified and is an improvement.

Another improvement involves the release of Tellurium which, previously has been considered to tie-up with un-reacted Zr clad. The range of experiments in PHEBUS and VERCORS has contradicted this and it now appears that Tellurium gets released during the in vessel phase as a relatively volatile species. This change, too, is a good one that addresses the bulk of the experimental data.

My overall assessment of the fission product release modeling in MELCOR 1.8.5 is that it represents the state of the art and is acceptable for use in determining the design basis source terms.

C.4.3 Fission Product Transport

The fission product transport modeling in MELCOR considers deposition via condensation of vapors and mass transport of aerosols. The major current difference involving the new proposed source term is in the new specification of the chemical form of cesium and molybdenum. This change tends to slightly reduce the re-vaporization of Cs and increases the re-vaporization of Moly. These changes do not markedly affect the source term other than for Moly and are acceptable changes.

C.4.4 Experimental Basis and Uncertainties

My first comment on this area is that, in my opinion, there is not a strong need for assessing uncertainties in specifying a design basis source term. The real uncertainties are in the MELCOR determinations themselves. These uncertainties, while substantial, have been previously assessed and we have fairly good knowledge of the extent of these. This proposed design basis source term in a sense addresses such uncertainties by selecting high CDF sequences to include in the source term options and selecting the median values of the calculated releases for these.

There does not seem to be any real technical basis for selecting the median values for the design basis source terms. Such a technical basis would have to involve an evaluation of the effects of the selection on the things that the ST is used for such as equipment qualification

(failure rates of SSCs), containment isolation times, extent of control room operator protection, design leak rate specifications, and associated effects on siting (exclusion boundaries and distances to population centers) and whether or not these effects are acceptable. Developing any such technical basis appears to be outside the scope of this work. It was appropriate, thus, that judgment was made that selecting the median provides sufficient risk margin and defense in depth for the various uses of the source term.

All-in-all, the use of the median at this time appears to give a ST that is not radically different from NUREG–1465 which has been in use for some time now and does seem to provide sufficient defense in depth. The various plants that have utilized NUREG–1465 have relatively good risk statuses and generally meet the Safety Goals. This gives some assurance that use of the new proposed ST will not erode the margins unacceptably.

C.4.5 Integration into MELCOR & Model Assessment Studies

The integration into MELCOR mostly involves the selection of appropriate Booth models for LEU, HBU, and MOX fuels. In addition, the finite cellular representation of the cores have utilized real specifications for radial and axial power distributions and locations of MOX fuel. The use of ORIGIN to develop inventories under these new conditions was well-advised and gives assurance that the initial inventories and their distributions are on a good technical basis. This integration is entirely appropriate and was well done.

C.4.6 Selection of Accident Sequences

I very much like the idea of basing a design basis source term on accident sequences selected to encompass most of the core damage frequencies (CDF) for the various reactor types. I prefer CDF as opposed to risk-dominant sequences because this tends to minimize the effects of siting and atmospheric transport on the selection.

The sequences selected for the various PWR, BWR, HBU, and MOX conditions are appropriate and the specific plants for which the ST assessments were developed are good choices because they have had a good history of PRA assessment giving some assurance that the CDF values can be relied on.

Each of these sequences were given a MELCOR assessment to determine sequence specific fission product releases into containment along with the timing thereof. At this point, Sandia chose to use a non-parametric statistical analysis for the releases and separately the timing to determine distributions and to extract from those the median values. One might ask what is the added value of this non-parametric treatment? There are basically two reasons for such treatment. The first is that the selected sequences from specific plants represent only a sample from the broader population of similar plants. If one had the entire set of sequences from all plants, the releases and the timing would likely resemble a continuum very much like that developed by a non-parametric statistical analysis treating the specific source terms as if they represented a random sample from this continuum. If that is indeed a good approximation, then this gives a way to actually select the median (or some other value) and have some assurance that it might be close to the actual median of the full population.

The so-called variation developed for the distributions gives some notion of how well the median is known based on the number of samples and their locations.

C.4.7 Proposed Release Quantities and Associated Characteristics

The proposed release quantities make a good representation of accidents with considerable core melting and are based on the best available experimental data and analysis of sequences. The somewhat significant changes with respect to Cs, Mo, and Te have good experimental bases and provide a more realistic specification of the source terms.

C.4.8 Documentation of the Proposed Source Term

The report - Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel by Powers et al does not stand alone without the associated supporting documents. It is good that the document provides complete references to all these supporting documents.

C.4.9 Observations Concerning Fukushima

Referring to the CSARP presentations concerning Fukushima, the choice of a gaseous iodine fission product source of 5% is highly uncertain because it is based on speculated and not actual measured values. It was arrived at by noting in the various fission product release experiments that the amount of molecular cesium release was far more than sufficient enough to completely “take up” the amount of released molecular iodine as CsI. As this take-up was considered a gas phase chemical reaction, it depended on the molecular concentrations of Cs and I, their reaction rates at the temperatures encountered after release from the fuel, and their residence times in the various decreasing temperature regions before the reaction essentially ceases leaving some unreacted iodine. The fission product tests referred to here, however have not been entirely prototypic from this chemical standpoint. If the real chemical situation (in the design basis accidents) also includes substantial amounts of other species competing for the Cs (such as Molybdenum and Boron), then there could be less opportunity for the Iodine to be almost completely reacted and compounded with the Cesium – meaning there could result in much more than 5% gaseous Iodine in the source term. As significantly increased amounts of gaseous Iodine in the source term could have implications with respect to the aspects of containment design to cope with the design basis source term (such as allowed leakage rate, containment atmosphere removal concepts, water chemistry, etc.), we should flag this as a substantial uncertainty and make some effort to establish the uncertainty level.

Our design basis source terms are to represent those accidents that are not associated with containment failure because those severe accidents are supposed to meet acceptable risk safety goals. Unfortunately, the risk acceptance safety goals are based on individual risk measures. The Fukushima accident clearly shows that societal risks (total deaths, total injuries, land contamination, clean-up, loss of the nuclear stations, etc.) are the dominant risks. In order to meet the currently non-existent societal risk acceptance criteria, I believe there would have to be more stringent requirements on CDF [e.g., 10^{-6}/yr] and LRF [e.g., 10^{-7}/yr]. Such stringent design basis might not be accompanied with a significantly different fission product source term (other than possibly the related timing) but there would be need to include specifications on

other sources (e.g. hydrogen and core concrete releases) as well as there would be a need for more reliable containment mitigation measures and strength. In other words, there would need to be a much different source term specification. This probably is not relevant to our current source term specification but is something to think about for a future need.

Other than the above, I found no additional implications for our current source term specifications.

C.5 Dr. David E. Leaver

C.5.1 Introduction

This note is my individual peer review report on the NRC effort to update the technical basis for the accident source term [i.e., SAND2011-0128] to address high burnup (B/U), mixed oxide (MOX) fuel. During the peer review process, the peer review group had the benefit of a complete set of documents describing the accident sequence selection, the MELCOR models and validation, special measures taken to make the calculations applicable to high B/U and MOX, and source term parameter results. In addition there were two 2-day peer review meetings which included presentations by Sandia experts as well as a comment resolution period in which Sandia responded to all peer reviewer comments, and the peer reviewers had the opportunity to evaluate the Sandia responses.

A separate document with my comments on the proposed accident source term documentation was provided on June 27, 2011 (see Appendix B, Section B.5). The present note provides a more integrated discussion of key issues on the accident source term update.

C.5.2 General Approach

Use of MELCOR vs. Expert Judgment The proposed accident source term update was motivated by the need to address high B/U, MOX fuel. However, general insights on accident source terms, rather than differences in the source term due to high B/U, MOX, are turning out to be the most interesting and useful aspect of the work. The use of MELCOR is a major improvement in the basis for the proposed accident source term vs. the expert judgment approach that was used in NUREG–1465. This is the case for several reasons:

- Significant MELCOR model development and validation has been performed over the last several years (see high B/U, MOX documentation and documentation on the NRC State-of-the-Art Reactor Consequence Analysis (SOARCA) now available in ADAMS) making MELCOR the state-of-the-art integrated severe accident analysis code.

- Use of MELCOR provides a consistent and transparent basis for specifying source terms that avoids the problems which are inherent in the expert elicitation process. These problems result from differences in background or interpretation of data from one expert to another or biases that are not fully explained. (The mean of a given source term parameter can be essentially completely controlled by the tail of the distribution which in turn can be unduly influenced by one outlier expert elicitation value.)

- While there are still uncertainties in MELCOR results due to use of simplified models to limit run time, limited experimental data in some areas, and selection of accident sequences to be calculated, the body of severe accident work performed by the global nuclear community since the early 1990s and the recent model development and validation of MELCOR provides confidence in the results such that no big surprises are expected downstream. See paragraph below on uncertainties for further discussion.

A final point regarding use of MELCOR is that it will have been roughly two decades since the NRC issued the original basis for accident source terms. It is reasonable that this can be revisited again 10 to 20 years in the future as more is learned from operating experience, severe accident research, and understanding of the Fukushima accident.

Epistemic Uncertainties - Epistemic uncertainties from the overall distribution of source terms calculated for the various accidents are not considered in the proposed accident source term. An uncertainty analysis was performed in SAND2011-XXXX to address the aleatory (statistical) uncertainties.

There are several issues related to epistemic uncertainties. These are the use of point estimate characterizations of source term parameters given that there is uncertainty, the particular point estimate chosen (i.e., the confidence level), the need for explicit consideration of epistemic uncertainties in the MELCOR calculations, and how the treatment of epistemic uncertainties is described in the documentation. Each of these issues is discussed below.

1. Use of point estimates is considered reasonable as a way to simplify the source term and its application. An alternative would be to specify a distribution for each source term parameter and perform a Monte Carlo analysis to determine a distribution on release to containment or some other intermediate result. While this could be done, it represents a significant increase in complexity and level of effort, not only for NRC but also for the applicant, it would negatively impact transparency of the licensing analysis, and it is not really necessary as discussed further below.

2. The point estimate used in NUREG–1465 was the mean or 75th percentile, whichever was less (this was done to address the tails issue). NUREG–1465 indicated that the median was considered, but the mean or 75th was felt to provide better appreciation of the range of values lying above the median. In the present work, the MELCOR calculated value is reported along with a statistical uncertainty range which reflects the limited sample size (limited number of accident sequences). So effectively, the present work used a best estimate. This is considered acceptable for two main reasons. First, the fact that containment integrity is maintained as part of the design basis accident (DBA) accident source term dose calculation (i.e., containment is assumed to function as designed with respect to mitigation system performance and leak rate) removes the largest source of uncertainty in fission product release to the environment. Challenges to containment integrity are addressed in detail outside the design basis in plant specific PRAs performed by each licensee and the NRC studies such as NUREG–1150 and the recent SOARCA project. Second is the fact that the accident source term effort and the MELCOR model development and validation work are rooted in the overall source term research undertaken or supported by NRC over the last several decades such as the

Phébus tests, the VERDON experiments, ARTIST, and the STEM project. For these two reasons no major downstream surprises are expected in terms of accident source term parameters that would cause a significant impact on control room or offsite DBA dose.

3. With respect to the need for explicit consideration of epistemic uncertainties in the MELCOR calculations, this could be done similar to what is being done for SOARCA. It is, however, a very substantial undertaking for which there are practical limitations of resources and computer time, and for the same two reasons in 2. does not seem to be worth the effort. Future updates of the accident source term could address this as the methods and computer tools evolve. Having said this, some of the SOARCA uncertainty results would be applicable to accident source term and could be used to strengthen the high B/U, MOX source term work with respect to addressing uncertainties.

4. With regard to how the treatment of epistemic uncertainties is described in the documentation, page 21 of SAND2011-XXXX, "Accident Source Terms for LWR Nuclear Power Plants Using High B/U or MOX Fuel," states that… "No attempt has been made to systematically quantify the state of knowledge uncertainty (epistemic uncertainty) in the results." Further elaboration on what has been done generally by NRC as noted in 2. and why a systematic quantification of uncertainties is not really necessary would be a good idea.

A final comment on epistemic uncertainties is the fact that it is incredible that the operators would not take measures that would mitigate releases in some way. This is not considered in the proposed accident source term, in part because, as pointed out in discussions as part of the peer review process, operator actions that go beyond the design basis are plant-specific and in many cases, the procedures are proprietary. This would make it difficult to include specific information in the proposed accident source term documentation. However, the fact that operator actions are not credited in the proposed accident source term further strengthens the notion that the source terms developed here would not be exceeded by source terms from any accidents considered credible.

Breakdown into Phases - NUREG–1465 assumed constant release rate. In addressing release rate, Section 4.4 of SAND2011-XXXX includes two figures that show calculated in-vessel release rate with higher short term aerosol mass in containment compared to constant in-vessel release rate, and the question arose as to whether the proposed accident source term should use constant or variable release rates.

Constant release rates are probably more representative of the spectrum of higher probability accidents, though NRC may want to retain the option of plant-specific release rates, esp. small modular reactors (SMRs). Another reason for using constant release rates is that specifying a variable (i.e., front loaded) release could drive the design toward a containment fission product mitigation system that is limited in duration and mitigates the front portion of the release but is not functioning later when a significant portion of the release may actually occur. (Note that this was the case for the AP600 passive, accumulator-driven spray system which had capacity for 30 minutes and with the TID-14844 source term (instantaneous release into containment at time zero) would have been designed to actuate at time zero, thus missing the real accident release.)

Generally, it appears to me that for the range of accident sequences being considered, there is no definitive trend toward high release rates early in the accident. The basis for this is that the fuel heatup and oxidation process is incoherent, and other than medium to large sized LOCAs, which are relatively low in probability, the release to containment is relatively constant, possibly with a spike late due to lower head failure. Figure 20 of SAND2008-6664 (which is for a Sequoyah SBO with no AFW), illustrates this behavior.

On a related matter, in discussions as part of the peer review process, Sandia indicated that the sequence used as the basis for the Figures 4 and 5 of Section 4.4 of SAND2011-XXXX, showing calculated in-vessel release rate with higher short term aerosol mass in containment, was a relatively slow station blackout (SBO). The SAND2011-XXXX, Figure 4 and 5 plots look more like large LOCA as opposed to slow SBO (i.e., based on the red-dashed, calculated in-vessel release rate, release starts at ~0.5 hr and ends at ~2 hr which is not a slow SBO). In any event, the document should state what sequence is the basis for these figures.

Treatment of Gap Release - With regard to gap fraction, there appears to be justification based on the high B/U, MOX work to eliminate a separate Gap Release Phase. This makes sense since it simplifies application of the proposed accident source term by the licensee, simplifies NRC review, and adds to transparency of the source term by not implying that there is a separate “gap release” consisting of fission products from the gap. A separate effort may be necessary to address release from reactivity induced accidents (neither NUREG–1465 nor the high B/U, MOX work address reactivity induced accidents). In addition, if the separate gap phase is eliminated, the basis for Table 3 of RG 1.183 will need to be defined. While it is not explicitly stated in RG 1.183, NUREG 1465 gap releases support Table 3. Particularly for fuel handling accident (FHA), RG 1.183, Table 3 or equivalent is needed.

Treatment of Early In-Vessel Release - The proposed accident source term defines early in-vessel release duration as the release occurring up to the time of lower vessel head failure. One of the things the MELCOR calculations are showing is that the damaged core is colder than previously believed (due to relocation starting at 2600 degrees K and better modeling of heat loss mechanisms) and lower head failure takes longer, i.e., of the order of many hours depending upon the sequence. Defining early in-vessel release duration as the release occurring up to the time of lower vessel head failure is one option which is transparent and easy to explain. On the other hand, as noted above it is incredible that the operators would not take measures that would mitigate releases in some way.

One solution to this dilemma for operating plants is to ignore operator actions for purposes of defining early in-vessel release duration, but cite operator actions as a source of conservatism. A more informed solution which could be applied to new plants might be to allow individual design certification applicants to justify a shorter in-vessel release duration based on their design and associated operator actions which they are able to justify.

Not Differentiating Between Low B/U, High B/U, and MOX - SAND2011-XXXX, Accident Source Term for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel,” state in Section 6 that in terms of fractional releases, source terms developed for high B/U fuel and MOX fuel do not differ markedly from source terms developed by similar means for lower B/U and/or low enriched uranium (LEU) fuel. In the peer review meetings, it was discussed whether

there should be differentiation between source terms developed for high B/U and/or MOX fuel vs. low B/U, LEU fuel.

NUREG–1465 differentiated between BWR source terms and PWR source terms. Table C.5.1 shows parameter values for gap plus early in-vessel iodine release magnitude and gap plus early in-vessel release duration. As can be seen from the table, the difference in iodine release magnitude between PWRs and BWRs in NUREG–1465 was one third, i.e., PWR iodine release magnitude is 4/3 times BWR iodine release magnitude. There were technical reasons for this (though the difference between BWRs and PWRs is in the other direction for the updated source term).

Table C.5.1 Parameter Values for Gap Plus Early In-Vessel Iodine

Source Term Estimate	PWR		BWR	
	I Release (fraction of core inventory)*	Release Duration (hr)*	I Release (fraction of core inventory)*	Release Duration (hr)*
NUREG–1465	0.40	1.8	0.30	2.0
Low B/U, LEU Update	0.307	5.63	0.543	9.0
High B/U, LEU Update	0.374	4.72	0.472	8.16
MOX	0.508	4.76	N/C	N/C

*Gap plus early in-vessel

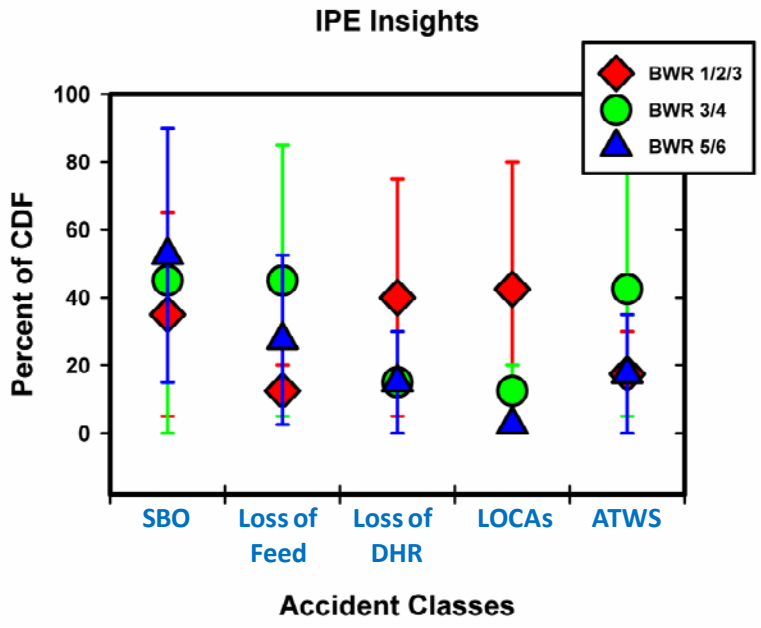
Looking at the values in Table C.5.1, and given the precedent of differentiation between BWRs and PWRs in NUREG–1465, it would seem appropriate to differentiate between low B/U LEU, high B/U, and MOX as well as between BWRs and PWRs. While slightly more complicated to have several source terms, it is basically the same approach taken in NUREG–1465 which has worked well from this standpoint, and it would not unduly penalize the non-MOX plants. If there is differentiation between MOX and non-MOX plants, care should be taken not to unnecessarily dramatize the higher MOX release as it is not significant from a risk standpoint.

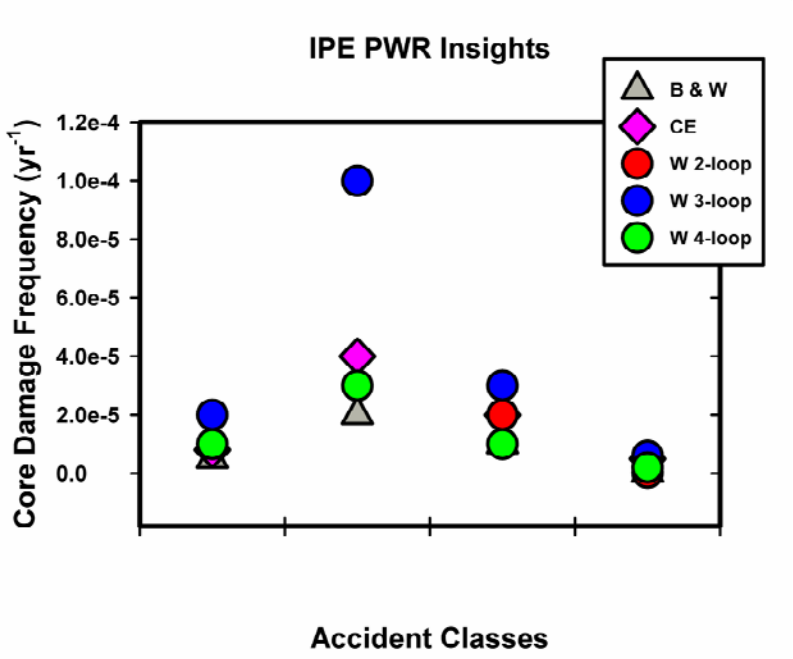
C.5.3 Selection and Treatment of Accident Scenarios

During the peer review, questions were raised with respect to selection of accident sequences, the weighting process, i.e., instead of weighting each of the accident analysis results equally (see SAND2011-XXXX, “Accident Source Terms for LWR Nuclear Power Plants Using High B/U or MOX Fuel,” page 18), recognize the relative probability of the various sequences being considered, and use of the median of the cumulative distribution of accident sequences for the source term parameter.

The selection of accident sequences appeared to be very reasonable and complete, covering all of the dominant core damage scenarios for a variety of both BWRs and PWRs. With regard to the weighting process, Sandia indicated that given that the level of aggregation adopted for this work is BWR and PWR, aggregation is based on results reported in the IPE Insights document [NUREG–1560] which shows that any defensible level of weighting would be about equal for each class with the possible exception of ATWS (see figures below).

While accident frequency weighting could be done, the approach used (no weighting by accident frequency) simplifies the analysis and is reasonable for PWRs where, with the exception of Westinghouse 3-loop, which is less than 20% of the PWR population, the accident class frequencies are quite close (see also Figure 3.19 of NUREG–1560 which applies to Westinghouse 4-loop which comprise a significant plurality of PWRs). It is also reasonable for





BWRs where, with the exception of BWR 5/6 which is about 25% of the BWR population, the accident class frequencies are quite close (see also Figure 3.6 of NUREG–1560 which applies to BWR 3/4 which are the majority of BWRs).

With regard to use of the median, the peer review group discussed use of the median vs. the mean. Use of the median as explained on page 18 of SAND2011-XXXX appears reasonable and is slightly conservative relative to the mean, at least for iodine (which is the most important contributor to dose) for both release magnitude (median > mean) and release timing (median < mean). It should also be noted that NUREG–1465 used the lesser of mean and 75% and the high B/U, MOX study is using median which is greater than the mean.

C.5.4 Proposed PWR and BWR Release Quantities and Associated Characteristics

With regard to the proposed source term parameters, the discussion above under General Approach applies to release rate, release duration, gap release, and differentiation of MOX vs. non-MOX plants. Other than these parameters, I have no issues with the proposed source term parameters.

Two final comments on uncertainties and experimental bases are as follows:

1. The three largest sources of uncertainty with respect to fission product release to the environment (and thus risk to the public and the environment) are containment integrity, iodine chemical form, and low volatile fission product release to containment (in contrast to volatile fission product release where the uncertainty is limited since for iodine, for example, the release can only be a factor of 2 or 3 higher).

   If containment integrity is maintained (e.g., low leak rates of the order of a few tenths of volume percent per day) as is assumed for DBA radiological dose analysis (for which the proposed accident source term is intended) the release to the environment will be small.

   Iodine chemical form is important because of the volatility and radiological importance of iodine. Iodine chemical form is discussed in SAND2011-XXXX where it is noted that iodine chemical form is the subject of continuing debate and research. The outcome of this debate and research are uncertain at this point, but one possible outcome is a significantly higher gaseous iodine fraction in containment which would result in greater iodine release to the environment. Because of the potential importance of this issue, it is suggested that the section on iodine chemical form be expanded to include a summary of recent references on the ongoing experimental and analytical work. This would be useful to new plant applicants such as SMRs which may need to stay abreast of the iodine work and reflect it in their radiological DBA analyses.

   The cerium group in-vessel release magnitude was addressed extensively in the peer review discussions. The issue was raised mainly due to the fact that the proposed accident source term Ce group in-vessel release (1.3E-7 for high B/U BWRs and 1.5E-7 for high B/U PWRs) is almost 3 orders of magnitude lower than what is in NUREG–1465 and the fact that this estimate applies to plutonium which is very important with respect to radiological effect and is in the Ce group.

In my view, the peer review discussion did not provide a basis for changing the MELCOR calculated value for the Ce group. Reasons for sticking with the MELCOR calculated value are as follows:

- Use of the MELCOR-based estimate provides a consistent approach, and while there are uncertainties, there is no data that contradicts the MELCOR result.
- The expert elicitation process was the basis for the parameter values in NUREG–1465 and as noted above, one expert's estimate can significantly bias the result; this is a less reliable process than physical modeling, even with the fact that the MELCOR model has some uncertainties.
- The higher Pu releases in the range of the in-vessel release from NUREG–1465 have very little effect on DBA dose which is the subject of this work.
- Finally, the fact that as more is learned, including the wealth of information that will no doubt come from cleanup and dissection of Fukushima over the ensuing years, the proposed accident source term can be updated.

However, because of the optics of the difference in Ce group release fraction between the proposed accident source term and NUREG–1465, there should be further explanation in SAND2011-XXXX of the reasons for the difference, and a summary of what is known today about Pu release from Phébus, the ORNL VI tests, VERCOUR, VEGA, and Fukushima.

2. The Fukushima accident bears some discussion here. Above and beyond its effect on Japan, this accident is significant in two ways - (1) its potential impact on the U.S. industry as well as globally, and (2) the visibility of the accident and the public attention that it has drawn and will continue to draw for years. Because of this, not mentioning the accident in the high B/U, MOX documentation strains the credibility of the NRC effort. There needs to be some acknowledgment of the accident somewhere in the high B/U, MOX source term documents, if only brief and even with the recognition that there is still a great deal we don't know about what happened. This brief acknowledgment could, for example, indicate that there is nothing known at this point from Fukushima that would contradict the results of the proposed accident source term, or perhaps that there is still so little known with certainty that no conclusion can be drawn but reliable accident data will be considered when available.

   This matter is important enough that I believe the peer review group should have a one day meeting in which Sandia and NRC present what data they have on the accident progression (e.g., results of MELCOR calculations simulating the accidents at Units 1, 2, and 3) and the fission product release from the core and RCS. Peer review members could also present any information or insights they believe to be relevant. The objective of this meeting would be to inform the peer review group so as to allow comments to NRC on the need for acknowledgment of the accident in the documentation and what could be said at this point.

C.5.5 Documentation of Proposed Source Terms

The quality and transparency of the documentation of the source term work is crucial to its acceptance and credibility. Generally, the documentation reviewed by the peer review group was excellent and quite complete. This having been said, there are several areas where the documentation can be improved. These were pointed out in my June 27, 2011, comments and are summarized as follows:

- In SAND2008-6664, "Accident Source Terms for PWRs with High B/U Cores Calculated with MELCOR 1.8.5," it appears that there are inconsistencies between the plots in Section 6 and the tables in Section 5 from which the plots were taken. Also, it was not possible to reconcile the data in SAND2008-6664 Appendix C with the tables in Section 5. An exchange with Sandia indicated that these matters were being addressed. In conjunction with this, some examples would help with regard to going from tables in Section 5 to plots in Section 6.

- SAND2008-6664 Appendix D should indicate which Surry sequence is being presented (1A or 1D). Also, several of the figures only have one set of curves (LBU or HBU?). Sandia responded that the plots in Appendix D are for the HBU Surry SBO, not an AFW sequence (Case 1A) and that the text should have stated this. However, further checking indicates that the Appendix D sequence appears to be neither 1A nor 1D. For example, the Appendix D sequence shows accumulator injection beginning at 12 hours whereas 1A and 1D have accumulator injection starting in the range of 3 to 5 hours. Also, the opening paragraph of Appendix D states that there are plots for both low B/U and high B/U calculations, but the graphs only have one plot.

- IN SAND2008-6664, Appendix D type data for additional sequences would be useful. Similarly, including figures corresponding to Figures 20 and 21 would be useful and would more fully document the work done to support the source term. There are two reasons that at least some of these results are needed. First is that these results will provide confidence in the proposed accident source term release parameters since it allows the user to see where the results come from. Second is the fact that these results may be useful in the future, e.g., for applicants who desire to propose alternatives to aspects of the regulatory guidance or for SMR designers who may have to justify their own source term. Note that this NRC effort to upgrade the accident source term is probably a once every 10 to 20 year exercise and if we don't capture this type of information now it could be lost forever to users.

- It was not possible to confirm the SAND2011-0128 Table 11 values (i.e., median PWR HBU source term parameters) with SAND2008-6664 values. During the peer review, Sandia indicated that the SAND2008-6664 values are incorrect (and out of date), that these values had not been used in SAND2011-0128, and that SAND2008-6664 would be withdrawn and reworked with new numbers that are correct and were actually used.

C.5.6 Limitations on the Use of the Proposed Source Terms

Limitations that should be stated and discussed include the following:

- Source terms not intended for non-LWRs
- Source terms should be justified on a design specific basis if used for LWR SMRs
- Applicable for burnups up to 62 GWD/t for peak rod
- Intended for use for radiological DBA dose analysis to address offsite and control room dose; while useful information is provided, the proposed source terms are not really intended for beyond DBA applications such as PRA and emergency planning zone sizing

C.5.7 Observations Concerning Fukushima

Referring to the CSARP presentations concerning Fukushima, it is noted that the Fukushima accident evaluation results are limited and preliminary at this point in time, and that much will be learned in the coming months and years as the accident remains are examined and the models are improved and made more plant-specific. Nonetheless, it is appropriate as part of the proposed accident source term peer review to examine what is known about the Fukushima accident at this point in time that could shed light on the proposed accident source term. Based on my review, there is nothing that has come from the evaluation of the Fukushima accident to date that would contradict the results of the proposed accident source term. Specific comments are as follows:

The key BWR parameters for design basis accident (DBA) source terms are the time of beginning of fission product release, the duration of release, the magnitude of release (to containment prior to RPV failure), and iodine form. The table below shows the first three of these parameters for Fukushima Dai-ichi Units 1, 2, and 3, and the corresponding updated BWR proposed accident source term numbers.

	Beginning of Release	**Duration of Release**	**Magnitude of Iodine Release to DW [6]**
1F1	~3 - 4 hr	~5 – 10 hr	~45% [1]
1F2	~75 - 80 hr	~5 – 12 hr	Small (~10% or less) [2,3]
1F3	~36 - 40 hr	~36 hr	Small (~1%) [4]
Draft Updated BWR AST	0 hr [5]	8.2 hr	47%

[1] ~50% of I and Cs also released to wetwell.
[2] I, Cs release to DW at time of RPV failure may have been somewhat less.
[3] ~40% of I and Cs also released to wetwell.
[4] Large fraction of I and Cs in wetwell.
[5] Updated AST has gap release beginning at time zero and fuel release beginning at about 10 minutes.
[6] Fraction of core inventory

In comparing the Fukushima parameters with the proposed accident source term, it should be noted that the proposed accident source term is not intended to match a specific sequence or to be bounding, but rather to simply be representative of LWR severe accidents up to the time of RPV failure and to provide a source term which challenges mitigation system and containment design. The following observations are offered from these results:

- There is a range of Fukushima values which reflects the several analyses which were reported at the CSARP meeting on the accident evaluation. Results are preliminary as noted above, and uncertainties exist due to limited availability of measurements, behavior of systems, and code uncertainties.

- Only iodine magnitude is given in the table. The updated BWR source term has a Cs release of 13%. With one exception, the Fukushima evaluations did not report separate I and Cs values, and for this exception the I and Cs release from the fuel was essentially the same, although the transport in the RCS and containment differed (greater Cs retention in the RCS).

- Even with the caveats on the Fukushima accident data, the draft updated BWR source term parameters are representative of the accident data. The timing is conservative (the one aspect of the proposed accident source term where this is intended), and while less conservative, the magnitude is still reasonable and challenging.

- Not shown in the table are the estimated Fukushima accident releases to the environment (generally of the order of about 1% for I and Cs). In the proposed accident source term as applied to DBAs, the accident release to the containment atmosphere (DW for the BWR) is considered up to the time of RPV failure, and the containment is assumed to be intact, leaking to the environment at the design basis leak rate. The associated 10 CFR 50.67 dose limits to the public must be met. Under these assumptions, I and Cs releases to the environment would be significantly smaller than the 1% value. While the SAND2011-0128 does provide a challenging source term for mitigation system performance and containment design for DBAs, given (1) the Fukushima accident which exceeded the design basis and where the collateral damage effects go beyond dose to the public (e.g., hydrogen explosion, worker dose, land contamination), and (2) the fact that like it or not we are in a zero-fault tolerant industry, the question arises as to whether existing safety goal and related risk limit guidance are specific enough with respect to source term and other severe accident matters. While outside the scope of SAND2011-0128, this is an area that deserves some thought in the future, perhaps as part of work on beyond design basis accidents.

With regard to iodine form, the Fukushima accident measurements available to date do not provide data that would necessarily contradict the 5% gaseous iodine source term specification in SAND2011-0128. While TEPCo's measurement indicated more than half of the iodine in the reactor building was gaseous, this can be explained by the fact that iodine as aerosol is less likely to leak from containment due to settling and scrubbing inside containment and plugging in small leak paths. There is, however, much work ongoing to address experimental observations from Phébus on iodine form in containment which when available is expected to engender a review of this part of the iodine specification.

In summary, based on my review there is nothing that has come from the evaluation of the Fukushima accident to date that would contradict the results of the updated SAND2011-0128. It is suggested that at some point in the future, depending upon the pace of availability of Fukushima accident information and related evaluations, NRC perform another review of the proposed accident source term against what is being learned from Fukushima.

C.5.8 Conclusions and Recommendations

The high B/U, MOX project is well-conceived and provides a much improved basis for a proposed accident source term, incorporating what has been learned over the last 20 years on severe accident core damage progression and fission product release and transport. In addition, the work is well-documented though as noted here improvements can be made in documentation. The peer review process was thorough, transparent, and well-managed.

The following specific conclusions and recommendations apply:

- The use of MELCOR is a much improved basis for a proposed accident source term vs. the expert judgment approach that was used in NUREG–1465. Reasons for this are the significant MELCOR model development and validation has been performed over the last several years, and the fact that use of MELCOR provides a consistent and transparent basis for specifying a proposed accident source term that avoids the problems which are inherent in the expert elicitation process.

- While there are still uncertainties in MELCOR results due to use of simplified models to limit run time, limited experimental data in some areas, and selection of accident sequences to be calculated, the body of severe accident work performed by the global nuclear community since the early 1990s and the recent model development and validation of MELCOR provides confidence in the results such that no big surprises are expected downstream. In this regard, it will have been roughly two decades since the NRC issued the original basis for accident source terms. It is reasonable that the basis can be revisited again 10 to 20 years in the future as more is learned from operating experience, severe accident research, and understanding of the Fukushima accident.

- Given the fact that containment is intact, containment mitigation systems are assumed to work, and the source term work is rooted in the overall source term research undertaken or supported by NRC over the last several decades, it is agreed that it is unnecessary to explicitly consider epistemic uncertainties in the high B/U, MOX source term work. The benefit of doing so does not seem to be worth the effort.

- It is incredible that the operators would not take measures that would mitigate the calculated releases in some way. This is not considered in SAND2011-0128, in part because operator actions that go beyond the design basis are plant-specific and in many cases, the procedures are proprietary. This would make it difficult to include specific information in the documentation. However, the fact that operator actions are not credited further strengthens the notion that the source terms developed here would not be exceeded by source terms from any accidents considered credible. This notion should be included in the source term documentation.

- Given the precedent of differentiation between BWRs and PWRs in NUREG–1465, it would seem appropriate to differentiate between low B/U LEU, high B/U, and MOX as well as between BWRs and PWRs. While slightly more complicated to have several source terms, it is basically the same approach taken in NUREG–1465 which has worked well from this standpoint, and it would not unduly penalize the non-MOX plants. If there is differentiation between MOX and non-MOX plants, care should be taken not to unnecessarily dramatize the higher MOX release as it is not significant from a risk standpoint.

- The quality and transparency of the documentation of the source term work is crucial to its acceptance and credibility. Generally, the documentation reviewed by the peer review group was excellent and quite complete. This having been said, there are several areas where the documentation can be improved. These were pointed out in my June 27, 2011 comments and are summarized in the main body of this report.

- Two additional points on documentation concern Ce group release and iodine chemistry:

    - While no change in the Ce group release is considered necessary, because of the optics of the difference in Ce group release fraction between SAND2011-0128 and NUREG–1465, there should be further explanation in SAND2011-XXXX of the reasons for the difference, and a summary of what is known today about Pu release from Phébus, the ORNL VI tests, VERCOUR, VEGA, and Fukushima.

    - Iodine chemical form is important because of the volatility and radiological

    - Importance of iodine. Iodine chemical form is discussed in SAND2011-XXXX where it is noted that iodine chemical form is the subject of continuing debate and research. The outcome of this debate and research are uncertain at this point. Because of the potential importance of this issue, it is suggested that the section on iodine chemical form be expanded to include a summary of recent references on the ongoing experimental and analytical work. This would be useful to new plant applicants such as SMRs which may need to stay abreast of the iodine work and reflect it in their radiological DBA analyses.

- It is considered that not mentioning the Fukushima accident in the high B/U, MOX documentation strains the credibility of the NRC effort. There needs to be some acknowledgment of the accident somewhere in the high B/U, MOX source term documents, if only brief and even with the recognition that there is still a great deal we don't know about what happened. This matter is important enough that I believe the peer review group should have a one day meeting in which Sandia and NRC present what data they have on the accident progression (e.g., results of MELCOR calculations simulating the accidents at Units 1, 2, and 3) and the fission product release from the core and RCS. Peer review members could also present any information or insights they believe to be relevant. The objective of this meeting would be to inform the peer review group so as to allow comments to NRC on the need for acknowledgment of the accident in the documentation and what could be said at this point.

C.6 Dr. Sudarshan Loyalka

I wanted to thank the Sandia team for their reports, presentations, response to my queries, and discussions over the last six months. I also wanted to thank you and your staff, and NRC and its participating staff for facilitating the review, and the opportunity for my own participation. Discussions with all of them, and the review team members, have been very helpful. We have conducted the review within a set of objectives and framework that were defined in our first meeting, and I will not restate these (and the reference documents) here as I am assuming that these will be all noted in the overall report. As we had all agreed I am providing my own individual comments below for use, as appropriate, in the review report (in the following, the phrases Accident Source Term and source terms etc. have been used interchangeably). I would be very willing and anxious to revise the review below in case I have misunderstood, misstated, or erroneously analyzed any issue.

C.6.1 General Approach

I want to emphasize that in my view use of MELCOR (1.8.5) is a major step forward beyond the use of STCP (NUREG–1465). It is helpful that MELCOR has been used for assessment of severe accident source terms in light water reactors with HBU and MOX cores, and also with regular (LBU) cores so that the results can be compared with previous NUREG–1465 estimations for regular cores. Not only does MELCOR have better modeling in several areas that I am very familiar with, but its integrated nature makes it possible to both review and improve the modeling and estimations in a more organized and convenient fashion. Expert judgments are valuable, but these are even more useful (and open to greater scrutiny and transparency) when these can be implemented (even with some ambiguity) in a computer program. Thus, I recommend use of MELCOR and estimations based on it for the accident source term in place of the existing STCP based estimations (NUREG–1465) for all situations in so far as is possible and practicable.

The focus of this review is not MELCOR, but since MELCOR is the central tool for the analysis, it is important to have a good understanding of its strengths and limits. Throughout the review, it has been noted that MELCOR ensures mass and energy conservation in computations. MELCOR developers have been cognizant of experimental data that have become available from recent tests such as PHEBUS , VERCORS, VEGA, etc. and they have incorporated understandings from VERCORS and PHEBUS in MELCOR, for example in the fission gas release models (from the fuel) and formation of Cesium and Molybdenum compounds and aerosols. These steps have obviously improved agreements of MELCOR computations with the available data from these experiments (primarily PHEBUS and VERCORS), and also have had a significant impact on estimations of cesium releases in particular. In the absence of (to my knowledge) availability of any other integrated simulation program that has as much details and up-to-date information built in as in MELCOR, this also provides some substance to the assertion that for the present review MELCOR should be regarded as the state of the art.

It will be helpful however to define the “state of the art” clearly and unambiguously. MELCOR clearly has some models that are new or the latest available, but it also has models that have been frozen over the years and which can be upgraded in view of the progress of the last many years. It is difficult to predict though as to what impact these factors (or the CFD simulations for

certain situations/phenomena) would have on the overall results. Certainly any changes in the models/code structures are expensive undertakings. Yet such factors are part of all computations and will always remain so as our understandings and capabilities improve. Identification of such issues, and improvements in MELCOR and reviews of methodologies and models would be helpful and should be a continuing and vigorous effort.

The study has addressed well a range of (relatively) high frequency core damage sequences, and variations within these sequences with respect to failure/non-failure and timing of several functions. For each case, model parameters have been chosen based on previous experiences with the models, underlying phenomenology and assumptions, sensitivities of estimations to parameters, and judgments of the Sandia team (some of the assumptions and parametric choices are described in Appendix A of SAND2006-XXXX and Appendix A of SAND2008-6665, but this list is certainly not complete as many modeling assumptions and other parameters are also involved). The results are thus the best estimates, and related confidence levels due to epistemic uncertainties have either not been obtained or not reported (such uncertainties almost always exist because of the complexity of a problem and limits of modeling, and phenomena not modeled or not adequately modeled). Since the parametric space is very large, and the phenomena span many time and length scales, these are well recognized and difficult issues, and these are also difficult to fully document in reasonable space. Still, it will be useful to address these topics more fully sometime in the future to provide directions for future research and estimations.

The MELCOR computations simulate the time evolution of an accident, and thus there is no clear (and artificial) breakdown of different phases such as gap release, (early) in-vessel release, ex-vessel release, and the late in-vessel release, but still the releases can be viewed within this classification if one so chooses. The classification was convenient in earlier times, and it still has some value, but it is not clear if the concept of “gap release” is still valuable (the concept is most useful when only a single rod in an experimental situation is involved, but with multitude of rods in different stages of failure in different regions of the core, there is no clear phase when only the gap releases occur).

It is also useful to note that the accident source terms reported here are patterned after the source terms developed in NUREG–1465. These account for retention of radionuclides (vapors, gases, aerosols) in the reactor coolant system, but do not account for the effects of natural and engineered processes that lead to their deposition in the reactor containment building or leakage from it. In many circumstances, the natural deposition and retention processes, as well as engineered processes and operator intervention can play an important role in available airborne radionuclide inventories in the containment, and these should be considered in the future.

C.6.2 Selection and Treatment of Scenarios

The study has focused on sequences that cover the majority that lead to large core damage. These sequences (accidents) have not been randomly selected, but the selection was deliberate based on regulatory interest and risk profiles of the selected plants. The results on the timing (of various release phases) and the amount of releases have been described in terms of cumulative distributions with associated uncertainties through use of non-parametric

statistics. The sequence results were not weighed according to estimated sequence frequency, and the emphasis has been on the median rather than the mean values. This analysis is both appropriate and useful.

C.6.3 Proposed PWR and BWR Release Quantities and Associated Characteristics

Release from fuel - The radionuclide (RN) inventories in the fuels have been obtained through use of ORIGEN code and nodalization of the reactor core. The RN releases from a fuel rod are obtained through the use of the diffusion equation in the fuel, with diffusion coefficients that have been adjusted to reproduce the release rates obtained in the VERCORS experiments for both LBU and HBU-MOX fuels (these procedures and models have variously come to be known as Booth, ORNL-Booth and MOX-Booth models). The models and procedures, together with other topics have been described in two supplemental Sandia reports (SAND2010-1633 and SAND2010-1635). The simple diffusion equation cannot capture all the complexities of the radionuclide transport in a nuclear fuel (the Sandia team is very cognizant of it) and the diffusion coefficients that are obtained (extracted from the data) for the two fuels have parametric constants that are quite different between the two sets of fuels, and should not be regarded as altogether realistic from a physics point of view. Still the descriptions are quite useful in reproducing the VERCORS results in a useful fashion. I carried out independent analytical and numerical calculations (noticing an oversight in the two Sandia reports with respect to reporting of release rates), obtaining the diffusion coefficients by an inverse calculation, and my results confirmed the accuracy of the values obtained by the Sandia team. Thus I am satisfied that MELCOR calculations for the resulting radionuclide release from the fuel are in general consistent with VERCORS experiments.

Transport to containment - Transport of radionuclides from the vessel to the containment is a very complex physical and chemical process involving molecules, aerosols, gases, vapors, liquids and solids in high temperatures and pressures. Fully realistic descriptions of such a complex process are presently not feasible, and certainly there is a need for much development here. The study team therefore has adopted an approach in which guidance from the results of the PHEBUS experiments (and associated theoretical analysis) has been adopted with respect to cesium, molybdenum and tellurium chemical forms and transport. These models have substantial impact on transport of these radionuclides to the containment (cesium and molybdenum combine to form cesium-molybdate, which because of a lower vapor pressure deposits and may vaporize later as an accident progresses, tellurium is assumed in chemical forms that do not deposit in the vessel and transport to the containment). It is important that computations are guided by experiments and theory both, and this indeed is the case here.

Timing of release - The timing of releases is different from those obtained in NUREG–1465. This has to do with improved models and methodologies in MELCOR, and the releases and transport discussed above. The releases in general are of longer duration, and in many sequences there is a large early release followed by a much slower later release to the containment. This suggests that the assumption of constant release in a phase (as in NUREG–1465) is open to discussion.

Experimental bases and uncertainties - The Sandia team gave a very good update on experimental bases and continuing experimental efforts on releases and transport (which

appear mostly international in scope). Some experiments and uncertainties in those (primarily VERCORS and VEGA) were also described and discussed by members of the review team particularly engaged and well versed in them. I do not have anything more to add, except that it should be clear to all involved that good experiments and theory at all levels are crucial to improving our understandings in areas critical to accident source term estimations.

C.6.4 Documentation of Proposed Source Terms

The documents and presentations have been very informative, and are of very good quality. Several members of the review team had suggested some clarifications and corrections during the meetings and exchanges with the Sandia team, and in so far as any document has not been published in a final form, revisions should be carried out as appropriate. In other cases where the documents have been published in apparently final forms (e.g. SAND2010-1635, page 17, Figure 3) the authors may wish to take appropriate opportunities to make the corrections in future related publications.

C.6.5 Limitations on the Use of the Proposed Source Terms

The proposed source terms are representative (not bounding or otherwise conservative), and have been calculated for some existing LWRs (BWRs and PWRs) with HBU and MOX Fuels and LBU fuels. Since the calculations were specific to LWRs of existing designs and specific types of fuels, and while these will provide guidance and approximate estimations, their use for other types of existing reactors, or future reactors with different designs and/or fuels will not be fully justified. In most such cases, extra modeling and calculations will be needed.

C.6.6 Conclusions and Recommendations

1. The present study is a major step forward beyond NUREG–1465. It provides Accident Source Term estimations based mainly on a single integrated computer program MELCOR. The program has utilized improved modeling, computing, and information from some recent separate effects and integral tests, which are beyond the state of the art that formed the basis for NUREG–1465.

2. The new estimations indicate that the Accident Source Terms are very similar for HBU and MOX fuels, and LBU fuels in similar plants for similar accident sequences. The study has provided credible reasons for this as much of the release occurs at higher temperatures where the releases from the different fuels are similar. The selection of accident sequences and the analysis of results are appropriate and well described.

3. The study has compared new source terms with the NUREG–1465 source terms. The new terms are lower in the case of cesium, higher for tellurium, and substantially lower (three orders of magnitude) for the Cerium group. These are all credible in view of the computations and accompanying reasons. Still, as the cerium group releases were already quite low in NUREG–1465 estimations, allowance for some greater uncertainty in the new cerium group estimations might be appropriate (that is a reduction of one or two orders of magnitude rather than three).

4. It seems to me that both the timing and amount of releases are important with in a phase itself, given the longer duration of releases in individual phases (as given by the new calculations) and the amount of releases should be viewed as variable (function of time) during an individual phase itself. I recognize that there are good arguments for averaging releases during an individual phase also, but those arguments are stronger when the phases are of shorter durations.

5. While good arguments exist for retaining the "gap release" phase in certain contexts, there does not seem to be a compelling scientific/technical reason for its retention in the context of the Accident Source Term (regulatory reasons are another matter, and I do not offer any views there).

The review team was asked to accept MELCOR 1.8.5 as the "state of the art" for the purposes of the review. It will be helpful however to define the "state of the art" with respect to MELCOR clearly and unambiguously. Since MELCOR is under continual development (some later versions of MELCOR have already emerged), such designations are both necessary (to provide a fixed frame for review) as well as challenging (not all aspects of MELCOR can be viewed as state of the art as some would understand it). Certainly if some new important experimental data or theoretical/computational information emerges that might potentially have significant impact on estimations, then there should be scope for revisions both during a review and later.

APPENDIX D: PRESENTATIONS AT THE MEETINGS

February 2-3, 2011

Dose Analysis for Design Basis Accidents, M. Blumberg

Revising NUREG–1465 for MOX and HBU LWR fuel, M. Salay and R. Lee

Fission Product Release Models and Experimental Bases, R. Gauntt

Results of Core Inventory Calculations for HBU and MOX Fuels, R. Gauntt

Assessment of Fission Product Release Modeling, D. Powers

Deposition Characteristics Affected by Release Model. D. Powers

Modification to Booth Release Model for MOX and HBU, D. Powers

Thoughts on Treatment for Air Oxidation Conditions, D. Powers

Outlines of VEGA Experimental Results on Radionuclide Release from Irradiated Fuel under Severe Accident Conditions, A. Hidaka and T. Kudo

The VERCORS experimental programme, B. Clement

May 10-11, 2011

Assessment of Responses to Questions, Review Findings, Conclusions and Recommendations, B. Clement

Technical Basis For Release Of Semi-Volatile Radionuclides, B. Clement

Assessment of Responses to Questions, Review Findings, Conclusions and Recommendations, R. Denning

AST Update Review Findings, D. Lever

Responses to Initial Peer Review Comments, D. Powers

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APPENDIX E: COMMITTEE MEMBER BIOGRAPHIES

Bernard Clément graduated 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.

He participated in the start-up of the Phébus reactor devoted to safety research experiments. As a test director, he was a member of the team that achieved the first experiments of the Phébus LOCA programme on the single rod and the 25-rod bundles at Cadarache. He also participated in the development and calibration of the two-phase flow measurements used for the blowdown phase of the experiments. Subsequently, he was responsible for the development and calibration of the in-pile instrumentation for the Phébus. The Cabri and Scarabee experiments were used for in-pile safety experiments for sodium-cooled fast breeder reactors. His duties were extended to the completion of the in-pile test sections, a challenging task given the extreme conditions of the experiments both for the 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 in the design of the Phébus FP test section, experimental circuits and instrumentation.

In 1991 he became Experimental Project Leader for the Phébus FP programme and then the Scientific Project Leader for the Phébus FP and the international Source Term programmes. He chairs the Scientific Analysis Working Group of both programmes. He is also a 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 in countries from the former USSR. Currently, he is a Senior Expert at IRSN.

Richard Denning spent most of his career at Battelle Memorial Institute before joining the Ohio State University (OSU) faculty in 1999. He is an internationally recognized expert in the fields of risk analysis, nuclear analysis, nuclear safety, and severe accident behavior of nuclear reactors. He has managed safety and risk studies of different types for a variety of nuclear facilities including commercial nuclear power plants and a number of the DOE non-reactor nuclear facilities. He was a primary contributor to the development of methods of Probabilistic Risk Assessment. He assisted the NRC in the development and oversight of its severe accident research program. He was a consultant to the TMI Special Inquiry Group. He was a member of the DOE Advisory Committee on Nuclear Facility Safety 1987-1991. From 1995 to 2007, he had responsibility for the oversight of safety hardware upgrades in the DOE program to improve the safety of former Soviet Union reactors (PNNL was the DOE lead laboratory). Dr. Denning served on the independent review and advisory committee for the No-Action Alternative Analysis for the Yucca Mountain Environmental Impact Statement in 1998-1999. 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 OSU on an interim basis from July 1999 to June 2001 and from March 2006 to June 2007. He is a fellow of the American Nuclear Society. He holds a joint appointment with Idaho National Laboratory.

Akihide Hidaka received his BS degree in physics from Tohoku University in 1980 and his PhD degree in quantum science and energy engineering from Tohoku University in 1998. He entered former Japan Atomic Energy Research Institute (current Japan Atomic Energy Agency) in 1980. He joined the development of SPEEDI (System for Prediction of Environmental Emergency Dose Information) code for atmospheric dispersion and radiation dose estimation due to an accidental release, and had a responsibility for WIND (Wide range piping INtegrity Demonstration)/WAVE (Wide range Aerosol model VErification) experiments on aerosol behavior in reactor coolant system and VEGA (Verification Experiments of radionuclide Gas/Aerosol release) program for radionuclide release from fuel under severe accident conditions. He participated in the IRSN's PHEBUS/FP program from 1992 to 1994 as a resident researcher and attended the first USNRC's source term applicability panel in September 2001. From 2004 to 2009 and since 2010, he was temporarily transferred to secretariat of the nuclear safety commission in Japanese government and supports the introduction of risk informed regulation into Japanese nuclear regulation and the change of regulatory positioning of severe accident management.

Thomas S. Kress is a past chairman of USNRC's Advisory Committee on Reactor Safeguards (ACRS) and is currently working as an independent consultant to ACRS and other organizations. He received his BS and MS degrees in Mechanical Engineering and his PhD degree in Engineering Science from the University of Tennessee. Before his retirement in 1994, he worked for 35 years in various capacities at Oak Ridge National Laboratory (ORNL), where he was involved in the design and safety aspects of LWRs, LMFBRs, Molten-Salt Reactors, Gas-Cooled Reactors, and Space Nuclear Applications. For several years, he managed ORNL's Severe Accident Programs for NRC, which dealt with all aspects of core degradation and source terms for LWRs. He was a member of the OECD/CSNI Group of Experts on Source Terms and Group of Experts on Aerosols. He helped develop a special source term report for CSNI. He was a technical expert elicited for NUREG–1150 and helped NRC develop NUREG–1465 (the current LWR design basis source term). He served as a technical expert for IAEA's evaluation of the Chernobyl accident and helped develop an IAEA TECDOC on design basis source terms for future LWRs.

David Leaver has more than 30 years of experience in the nuclear power industry, including extensive work in reactor safety, probabilistic risk assessment (PRA), radiological source term and accident analysis, terrorist risk studies, and emergency planning, including 16 years as a founder and principal at Polestar Applied Technology, now a WorleyParsons company. Dr. Leaver performed some of the earliest PRA studies of nuclear plants, was the industry leader in development and application of the alternative source term to advanced light water reactors (ALWRs) and operating LWRs, and led efforts for the industry over the last few years to improve and update the technical basis for emergency planning. Recently he has provided technical support for the development of source term and dose analysis for small modular reactors, high temperature gas reactors, and the Idaho National Laboratory (INL) Advanced Test Reactor (ATR). He was a participant in the 2002 NRC expert group on the effects of high burnup and mixed oxide fuels on the alternative source term, and is currently a participant in the peer review group for the NRC State-of-the-Art Reactor Consequence Analysis (SOARCA) effort and the peer review group for NRC's source term update. Dr Leaver holds a PhD in Mechanical Engineering from Stanford University and a BS in Electrical Engineering from the University of Washington.

Sudarshan Loyalka was educated at the University of Rajasthan, India (BE 1964), and Stanford University (MS 1965, PhD 1967), and is currently a Curators' Professor of Nuclear Engineering at the University of Missouri in Columbia. His educational and research interests are in transport theory, aerosol mechanics, the kinetic theory of gases, and neutron reactor physics and safety. He is a Fellow of both the American Physical Society (since 1982, for elucidating the role of gas-surface interactions on molecular transport) and the American Nuclear Society (since 1985, for contributions to reactor physics and safety and aerosol mechanics). He is a recipient of the David Sinclair Award (1995) of the American Association for Aerosol Research and the Glenn Murphy Award (1998) of the American Society for Engineering Education. Professor Loyalka's research has been supported by NSF, EPA, NASA, DOD, NIH, DOE, NRC, and the industry. He has also consulted with national laboratories and several companies both in the nuclear (fission product transport and heat transfer) and environmental areas (molecular and aerosol transport). He is a co-author/co-editor of 5 books, including, "Aerosol Science: Theory and Practice with special applications to the Nuclear Industry," by M. M. R. Williams and S. K. Loyalka, Pergamon (1991). Professor Loyalka is a member of the editorial advisory boards of both the Progress in Nuclear Energy and the Annals of Nuclear Energy, and he is a registered professional engineer in Missouri.

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