ACCIDENT SOURCE TERMS FOR LIGHT-WATER NUCLEAR POWER PLANTS: HIGH BURNUP AND MIXED OXIDE FUELS

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

ERI/NRC 02-202

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FIRST DRAFT FOR REVIEW AND COMMENT

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United States Nuclear Regulatory Commission
Washington, D.C.
Under Contract Number NRC-04-97-040

PREFACE

This report has been produced based on work performed by an expert panel organized by the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, to develop recommendations for changes, if necessary, to the revised source term as published in NUREG-1465, for application to high burnup and mixed oxide fuels.

Dr. Brent Boyack of Los Alamos National Laboratory served as the panel facilitator, and Energy Research, Inc. has been responsible for the preparation of the final report. Individual contributors to this report include:

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Substantial technical input was provided to the panel, by Mr. A. Hidaka of the Japan Atomic Energy Research Institute (JAERI), and Mr. J. Evrard of the Institut de Protection et de Sûrete Nucléaire (IPSN).

This work has been performed under the auspices of the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under contract number NRC-04-97-040. Mr. Jason Schaperow, the NRC Project Manager, provided considerable input to the conduct of the panel activities and the preparation of this report.

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BWR	Boiling Water Reactor
FSAR	Final Safety Analysis Report
HPME	High Pressure Melt Ejection
IPSN	Institut de Protection et de Sûrete Nucléaire
JAERI	Japan Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LEU	Low Enriched Uranium
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MOX	Mixed Oxide
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
SE	Steam Explosion
SNL	Sandia National Laboratories
STCP	Source Term Code Package
TR	Total Release

1. INTRODUCTION

1.1 Background

The history of the use of postulated accidental releases of radioactive material in the regulatory practices of the U.S. Nuclear Regulatory Commission (NRC) is summarized in Reference [1].

The NRC's reactor site criteria of 10 CFR Part 100 [2] require, for licensing purposes, that an accidental fission product release resulting from "substantial meltdown" of the core into the containment be postulated to occur and that its potential radiological consequences be evaluated assuming 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 to the environment." The radiological release is obtained from the containment leak rate and a knowledge of the airborne radioactive inventory in the containment atmosphere. The radioactive inventory within containment is referred to as the "in-containment accident source term."

The expression "in-containment accident source term," as used in this document is identical to that of Reference [1] which is defined as the radioactive material composition and magnitude, as well as the chemical and physical properties of the material within the containment that are available for leakage from the reactor to the environment. It is noted that the "in-containment accident source term" is normally a function of time and involves consideration of both fission products being released from the core into the containment, and the fission product removal by various mechanisms including natural processes and operation of engineered systems (e.g., containment sprays).

For currently licensed plants, the characteristics of the fission product release from the core into the containment are set forth in Regulatory Guides 1.3 [3] and 1.4 [4] and have been derived from the 1962 report, TID-14844 [5]. This release consists of:

- 100% of the core inventory of noble gases
- 50% of the core inventory of iodine (half of which are assumed to deposit on interior surfaces)
- 1% of the remaining solid fission products

Note that the latter was removed from consideration in Regulatory Guides 1.3 and 1.4. However, the 1% of the solid fission products are considered in certain areas such as equipment qualification [1].

As stated in Reference [1], Regulatory Guides 1.3 and 1.4 specify that the iodine chemical form is assumed to be predominantly (91%) in elemental (I_2) form, with 5% assumed to be particulate iodine and the remaining 4% is assumed to be in organic form (CH_3I, etc.). In addition, releases are assumed to occur instantaneously into the containment, following a postulated design basis accident.

Using the results of research on severe accidents and fission product releases since the publication of TID-14844, NUREG-1465 [1] proposed a revised set of source terms to the containment for Light-Water Reactors (LWRs), which are based on more realistic assumptions as related to release duration, release quantity, fission product aerosol retention, and chemical forms, to replace the TID-14844 based source term in licensing applications. Regulatory Guide 1.183 ("Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Reactors") was developed by NRC to support the final rule that amended 10 CFR Part 21, 50, and 54. The revised source term as proposed in NUREG-1465 [1] is primarily based on experiments and analytic studies applicable to low burnup (i.e., < 40 GWd/t) UO_2 fuel. Note that the use of the revised source term (NUREG-1465) has been voluntary by the licensees.

Additional research has been completed since the publication of NUREG-1465. This includes experimental and analytic studies in France [6-8] and Japan [9], and studies performed by others [10-13]. The French experimental programs [6] include the fission product release HEVA VERCORS experiments, the fission product vapor deposition condensation DEVAP studies, the aerosol behavior TUBA, TRANSAT, PITEAS, and FUCHSIA experiments, the CARAIDAS spray experiments, the small scale EPICUR, CAIMAN, and CHIP experiments dealing with iodine, and the Phebus-PF integral experiments. In addition, data have been collected in France through measurements to determine the gap inventory of the spent fuel rods from reactors at different burnup levels (up to 60 GWd/t) for UO_2 and mixed oxide fuels. Furthermore, in-pile FLASH experiments [8] have been performed to measure the fission product release from a fuel rod under Loss Of Coolant Accident (LOCA) conditions, including one experiment which involved high burn-up fuel (i.e., 50 GWd/t).

The VEGA experimental program [9] at the Japan Atomic Energy Research Institute (JAERI) is focusing on investigation of fission product releases from irradiated PWR and BWR fuel with fuel burnup ranging from 26 to about 61 GWd/t. Tests are also planned at JAERI that will include the spent MOX fuel from several European power plants with burnup levels reaching 80 GWd/t.

The regulatory applications of the revised source (gap and in-vessel release phases) as used for LOCA design basis analyses include:

- Assessment of individual dose at the Exclusion Area Boundary, at the Low Population Zone distance, and in the control room
- Containment isolation valve closure time (start time of gap release)
- Integrated dose used to qualify equipment in containment
- Post accident shielding, sampling, and access
- Hydrogen generated by radiolytic decomposition of water

In addition, NRC envisions the utilization of the revised source term for other regulatory applications.

1.2 Objectives

The primary objective of this report is to assess the applicability of NUREG-1465, and if possible, to define a revised accident source term for regulatory application to reactors using high burnup (i.e., burnup up to 75 GWd/MTU) low enriched uranium (LEU) and mixed oxide (MOX) fuels. Those aspects of the source term as proposed in NUREG-1465 that are potentially impacted by the use of high burn-up or MOX fuels will be addressed. The specific aspects addressed are chemical and physical forms, release timing, and release magnitude. Otherwise, the recommendations of NUREG-1465 are expected to be applicable.

1.3 Approach

The approach used in this effort is based on reconstitution of the source term panel that developed the source term uncertainty distributions for the NUREG-1150 study, which also served as the technical basis for NUREG-1465 [1]; and by considering (a) the data and insight that have been generated since NUREG-1465 was published, and (b) the physical phenomena that affect fission product release and transport mechanisms for high burnup and MOX fuels.

The panel was requested to provide specific recommendations for changes, if necessary, to the revised source term as published in NUREG-1465 [1], for high burnup and MOX fuels.

The panel meetings and deliberations were conducted during three meetings held at the Nuclear Regulatory Commission (NRC) Headquarters in Rockville, Maryland. All panel deliberations were transcribed. The list of panel members, and short biographies of each of the panel members are presented in Appendix A.

The panel was also requested to provide specific recommendations on the need for research to help establish appropriate source terms for high burnup and MOX fuels applications. Each letter from the panel members on the specific research recommendations is included in Appendix B.

2. BASIS FOR REVISED (NUREG-1465) SOURCE TERM

In 1995, the U.S. Nuclear Regulatory Commission published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" [1], which defined a revised accident source term for regulatory application. NUREG-1465 utilized current technical knowledge and understanding of LWR severe accident phenomenology to present, for regulatory purposes, a more realistic portrayal of the fission products present in the containment from a postulated severe accident. NUREG-1465 presents a representative accident source term for a Boiling Water Reactor (BWR) and for a Pressurized Water Reactor (PWR). These source terms are characterized by the composition and magnitude of fission product release into containment, the timing of the release into containment, and the physical and chemical forms in containment.

This chapter provides a summary of the technical basis for the NUREG -1465 accident source term. A brief qualitative discussion of the phenomenology of fission product release and transport behavior during the progression of severe accidents is presented and the technical basis for characterizing the revised accident source term parameters (composition, magnitude, timing and physical and chemical forms) is described.

2.1 Progression of Severe Accident Sequences and Release Phases

Radiological releases into containment under severe accident conditions can be generally categorized in terms of phenomenological phases associated with the degree of core damage and degradation, reactor pressure vessel integrity, and, as applicable, attack upon concrete below the reactor cavity by molten core materials. The general phases, or progression, of a severe LWR accident are shown in Figure 2.1.

Initially there is a release of coolant activity associated with a break or leak in the reactor coolant system (RCS). The radiological releases during the coolant activity release phase are negligible in comparison to the releases during the subsequent release phases. Assuming that the coolant loss cannot be accommodated by the reactor coolant makeup systems, or the emergency core cooling systems, fuel cladding failure would occur. Upon failure of the cladding, a small quantity of fission products that resides in the gap between fuel pellets and the fuel cladding would be released. This release, which is termed the gap release, would consist mostly of the volatile nuclides, particularly noble gases, iodine, and cesium.

As the accident progresses, core degradation begins, resulting in loss of fuel geometry accompanied by melting and relocation of core materials to the bottom of the reactor pressure vessel. During this period, the early in-vessel phase, significant quantities of the volatile nuclides in the core inventory as well as small fractions of the less volatile nuclides are released into containment. The fission products and other materials, which are released from the fuel, are likely to be transported through the various portions of the RCS before reaching the containment. As they move through the RCS, fission products may be retained as a result of various types of interactions. The extent of this retention depends on the fission product chemical and physical form and the thermal hydraulic

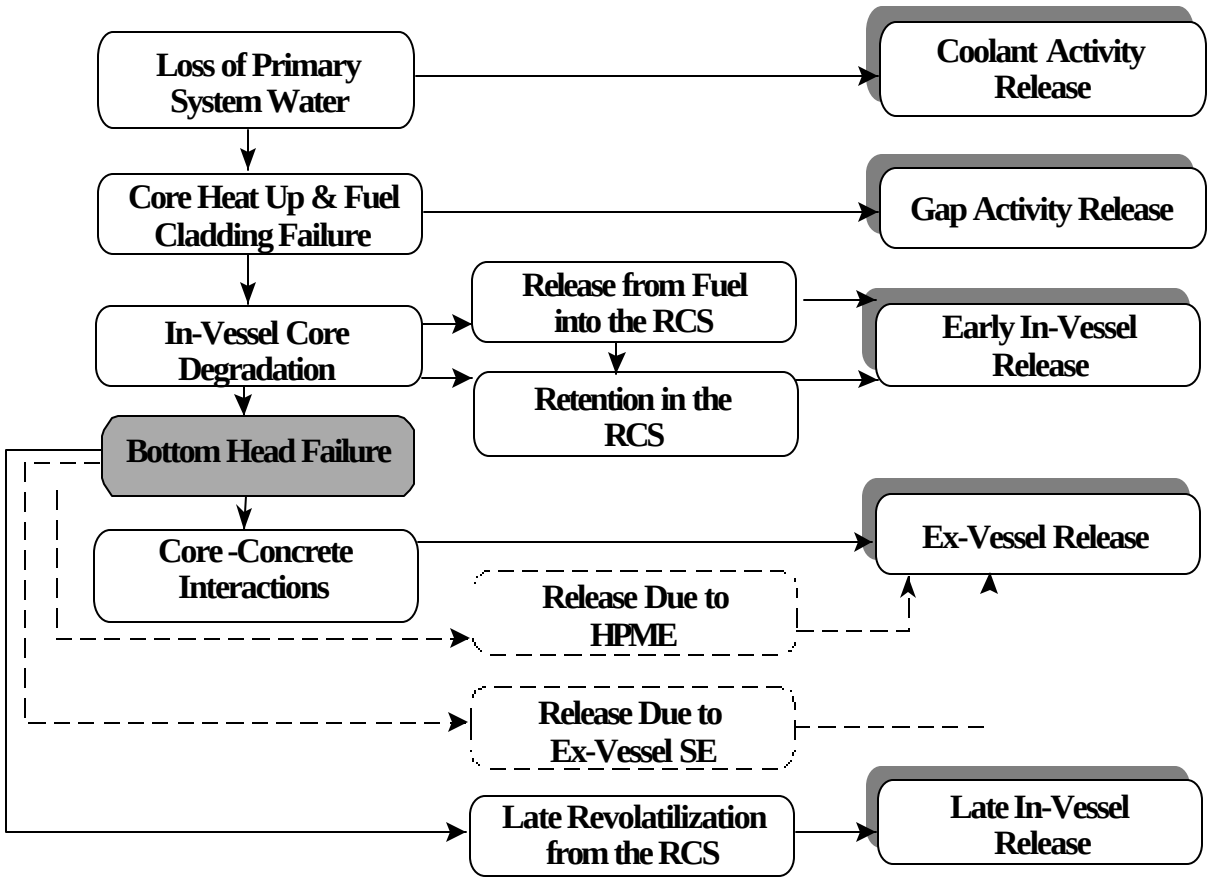

Figure 2.1 Progression of Severe Accident Sequences and Associated Release Phases

conditions along the flow path. The more volatile fission products would tend to enter the RCS as gases while the less volatile elements would tend to condense. The released fission product gases could absorb or condense onto aerosols and RCS surfaces, or react chemically with other species in the RCS atmosphere or with RCS surfaces. The amounts of fission products released into containment during the early in-vessel release phase are influenced by the residence time of the radioactive material within the RCS during core degradation. High-pressure sequences result in long residence times and significant retention of fission products within the RCS, while low-pressure sequences result in relatively short residence times and less retention within the RCS and consequently higher releases into containment.

If the bottom head of the reactor pressure vessel fails, two additional release phases (ex-vessel and late in-vessel) may occur. Following the bottom head failure, the molten core debris will be released from the reactor pressure vessel to the containment. Contact of molten core debris with the concrete structural materials of the cavity/pedestal below the reactor could lead to core-concrete interactions. As a result of these interactions,

significant quantities of the volatile radionuclides not already released during the early in-vessel phase as well as lesser quantities of non-volatile radionuclides are released into the containment atmosphere. Core-concrete interactions liberate copious amounts of concrete decomposition gas products. These gases may, in turn, sparge some of the less volatile nuclides, such as barium and strontium, and small fractions of the refractory nuclides into the containment atmosphere. Large quantities of non-radionuclide aerosols may also be released as a result of core-concrete interactions. The presence of water in the reactor cavity/pedestal overlaying any core debris can significantly reduce the ex-vessel releases (both radioactive and non radioactive) into the containment, either by cooling the core debris, or at least, by scrubbing the releases and retaining a large fraction in the water.

Two other phenomena that affect the ex-vessel release of fission products into containment could also occur. The first of these is referred to as "high pressure melt ejection" (HPME). If the RCS is at high pressure at the time of failure of the bottom head of the reactor pressure vessel, quantities of molten core materials could be ejected into the containment at high velocities. In addition to a potentially rapid rise in containment temperature, a significant amount of radioactive material could also be added to the containment atmosphere, primarily in the form of aerosols. The occurrence of HPME is precluded at low RCS pressures. A second phenomena that could affect the release of fission products into containment is a possible steam explosion (SE) as a result of interactions between molten core debris and water. This could lead to fine fragmentation of some portion of the molten core debris with an increase in the amount of airborne fission products. While small-scale steam explosions are considered quiet likely to occur, they will not result in significant increases in the airborne activity already within the containment. Large-scale steam explosions on the other hand, could result in significant increases in airborne activity, but are much less likely to occur. In any event, releases of particulates or vapors during steam explosions will also be accompanied by large amounts of water droplets, which would tend to quickly sweep released materials from the atmosphere.

Following the failure of the bottom head of the reactor pressure vessel, some of the volatile nuclides may be released into containment as a result of re-volatilization of the material, which had deposited on RCS structures during the in-vessel phase, or volatilization of material remaining in the reactor pressure vessel after vessel breach. This late in vessel release phase proceeds simultaneously with the occurrence of the ex-vessel phase. However, the late in-vessel releases have generally a longer duration than that of the ex-vessel releases.

2.2 Accidents Considered

In order to determine revised accident source terms for regulatory purposes, the NRC sponsored studies [13-15] that analyzed the timing, magnitude, and duration of fission product releases into containment. In addition, an examination and assessment of the chemical form of iodine likely to be found within containment as a result of a severe accident was also carried out [16].

A range of severe accidents that were previously analyzed for LWR plants was examined. Evaluation of a range of severe accident sequences was based upon work done in support of NUREG-1150 [17] study. The sequences studied progressed to a complete core-melt, involving failure of the reactor pressure vessel and including core-concrete interactions.

The revised source terms presented in NUREG-1465 were intended to be representative or typical, rather than conservative or bounding values, of those associated with a low pressure core melt accident, except for the onset of the release of gap activity, which was chosen conservatively. The release fractions for the revised source term were not intended to envelop all potential severe accident sequences, or to represent any single sequence.

2.3 Fission Product Composition

WASH-1400 [18] examined the spectrum of fission products and grouped 54 radionuclides into 7 major groups on the basis of similarity in chemical behavior. The effort associated with the STCP further analyzed these groupings and expanded the 7 fission products groups into 9 groups.

Both the results of the STCP analyses and the uncertainty analysis reported in NUREG/CR-5747 [14] indicated minor differences between Ba and Sr releases. Hence a revised grouping of radionuclides was developed that groups Ba and Sr together. A study of relative importance to offsite health consequences of the radioactive elements in a nuclear power reactor core, reported in NUREG/CR-4467 [19], found that other elements such as curium could be important for radiological consequences if released in sufficiently large quantities. For this reason, curium (Cm), americium (Am), and cobalt (Co) were also added to radionuclide elements. The revised radionuclide groups used in NUREG-1465 including the elements comprising each group are shown in Table 2.1.

Table 2.1 Revised Radionuclide Groups

Group	Elements in Group
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se
Barium, Strontium	Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium Group	Ce, Pu, Np

2.4 Fission Product Magnitude

The core inventory release fractions for the NUREG-1465 accident source term, by radionuclide groups, are listed in Table 2.2 for BWRs and Table 2.3 for PWRs.

Table 2.2 BWR Core Inventory Release Fractions for the Revised (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 2.3 PWR Core Inventory Release Fractions for the Revised (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

Based on WASH-1400 [18], the inventory of volatile fission products residing in the gap between the fuel and the cladding is no greater than 3 percent except for cesium, which was estimated to be 5 percent. NUREG/CR-4881 [13] reported a comparison of more recently available estimates and observations indicating that releases of the dominant fission product groups were generally below the values reported in WASH-1400.

NUREG-1465 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 over the duration of the gap release phase.

The early in-vessel, ex-vessel and late in-vessel release fractions for the revised accident source terms were derived from the simplification of NUREG-1150 [17] source terms documented in NUREG/CR-5747 [14]. NUREG/CR-5747 [14] utilized the current technical knowledge and understanding of the source term phenomenology to develop simplified formulation for realistic estimates of source term release into containment in terms of timing, nuclide types and quantities. Source terms parameters were quantified based on the detailed examination of available information, including results of the integrated Source Term Code Package (STCP) computer codes calculations and the insights from the NUREG-1150 expert elicitation on source terms issues [20]. Uncertainty analyses were also performed for releases into containment by utilizing the probability distributions for source term parameters used in NUREG-1150 study.

The early in-vessel release fractions presented in Tables 2.2 and 2.3, 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 [14]. The range of release estimates for the volatile nuclides, such as noble gases, iodine, cesium, and to some extent tellurium, spans about one order of magnitude. Therefore, it was concluded that for this group of nuclides, use of the mean value is a reasonable estimate of the release fraction. In contrast, the range of release estimates for the low volatile nuclides, such as barium, strontium, cerium and lanthanum, spans about four to six orders of magnitude. For the latter group of nuclides, the mean is controlled by the upper tail of the distribution, and the details of the whole distribution may be more indicative of the uncertainty than the "bottom line" results such as a mean value. Hence, in the final NUREG-1465 report, 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. It should be noted that , in the final NUREG-1465 report, the in-vessel release fraction for the tellurium was reduced somewhat in response to comments that tellurium will be retained via reaction with the Zircaloy cladding, with major releases occurring only on extensive oxidation of the clad.

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 [14]. The in-vessel releases due to heat up and degradation of the residual fuels following the vessel failure were not considered explicitly. However, it was assumed that the entire core participates in core-concrete interactions and therefore the volatile species (Halogens and Alkali Metals groups) remaining in the reactor pressure vessel at the reactor vessel breach were assumed to be released during the ex-vessel release phase. It should also be noted that, in the final NUREG-1465 report, the ex-vessel releases for tellurium and the low volatile nuclides were reduced as a result of comments received.

The late in-vessel release fractions presented in Tables 2.2 and 2.3 are also generally the mean values of the uncertainty distributions for a typical low-pressure core-melt accident scenario documented in NUREG/CR-5747 [14]. These releases are associated with the revaporization of radionuclides retained in the RCS and their subsequent release into the containment after the vessel failure.

2.5 Timing of Release Phases

Table 2.4 tabulates the onset and duration for each release phase of the revised accident source terms for BWRs and PWRs. The specified onset is the time following the initiation of accident (i.e., time=0). It should be also noted that the rate of release of fission products into the containment is assumed to be constant during the time duration shown.

Table 2.4 Timing of Release Phases

Release Phase	BWRs		PWRs	
	Onset	Duration	Onset	Duration
Gap Release	30 sec*	0.5 hr	10-30 sec*	0.5 hr
Early In-Vessel	0.5hr	1.5 hr	0.5 hr	1.3 hr
Ex-Vessel	2. hr	3 hr	1.8 hr	2 hr
Late In-Vessel	2 hr	10 hr	1.8 hr	10 hr

The timing was selected to be typical of a low-pressure core-melt scenario, except for the onset of the release of gap activity (duration of coolant activity), which was chosen conservatively to be based on the earliest calculated time of fuel rod failure under accident conditions (i.e., large-break LOCA).

In order to provide a realistic estimate of the shortest time for fuel failure, calculations were performed using the FRAPCON2, SCDAP/RELAP5 MOD3.0, and FRAPT6 computer codes for two PWR Plants (a B&W Plant with a 15x15 fuel rod array and a W Plant with a 17x17 fuel rod array [15]. The minimum time from the time of accident initiation until first fuel rod failure was calculated to be 13 and 24.6 seconds for the B&W and W plants, respectively. As noted in Reference 4, these estimate are valid for a double-ended rupture of the largest pipe. For a 6-inch break, the time until first fuel rod failure was estimated to be greater than 6.5 and 10 minutes for the B&W and W plants, respectively. It was expected that the CE plants would have coolant activity durations similar to the W plants. The review of the Final Safety Analysis Reports (FSARs) for BWRs indicated that fuel failure may occur significantly later, in the order of several minutes or more. At the time of the publication of NUREG-1465, no calculations for BWRs and CE plants had been performed, using the aforementioned suite of codes.

The gap activity phase ends when the fuel pellet bulk temperature has been raised sufficiently that significant amounts of fission products can no longer be retained in the

fuel. NUREG/CR-5747 [14] provided estimates for the onset of significant fission product release into the containment based on the review of STCP calculated results for 6 reference plants. Significant fission product releases were estimated to commence no earlier than about 30 minutes and 60 minutes for PWRs and BWRs, respectively, after the onset of the accident. More recent calculations [21] for the Peach Bottom Plant, using the MELCOR code, indicated that the duration of gap release for three BWR accident sequences were about 30 minutes as well. On this basis, the duration of gap release phase (onset of early in-vessel release phase) was selected to be 0.5 hours, for both PWRs and BWRs. This is in contrast to a large-break LOCA, during which the onset of significant fission product release can be in much less than 30 minutes.

The early in-vessel release phase ends when the bottom of the reactor pressure vessel fails, allowing molten core to fall onto the concrete below the reactor pressure vessel. NUREG/CR-5747 provided estimates for the early in-vessel release durations based on the review of STCP calculated results for seven reference plants. The early in-vessel release duration was found to be somewhat longer for BWR plants than for PWR plants. This is due to the lower power to moderator ratio and the lower core power density in BWRs, which would delay the time for core degradation and vessel failure. Representative early in-vessel release durations were selected to be 1.3 hours and 1.5 hours, for PWR and BWR plants respectively.

NUREG/CR-5747 [14] also provided estimates for the ex-vessel release durations based on the review of STCP calculated results. Although releases from core-concrete interactions are predicted to take place over a number of hours after vessel breach, the bulk of the fission product releases (about 90%), with the exception of tellurium and ruthenium, are expected to be released over a 2-hour period for PWRs and 3-hour period for BWRs. For tellurium and ruthenium, ex-vessel releases extend over 5 and 6 hours, respectively for PWRs and BWRs. Based on analysis in Reference [14], the ex-vessel release duration for the revised accident source terms was taken to be 2 and 3 hours, respectively, for PWRs and BWRs.

The duration for the late in-vessel release phase was taken to be 10 hours as recommended in Reference [14].

2.6 Iodine Chemical Form

The chemical form of iodine and its subsequent behavior after entering containment from the reactor coolant system was investigated in NUREG/CR-5732 [16]. On the basis of this work, NUREG-1465 concluded that iodine entering containment from the reactor coolant system is composed of at least 95% cesium iodide (CsI), with no more than 5% I plus HI. Once within containment, highly soluble CsI will readily dssolve in water pools and plate out on wet surfaces in ionic form. Radiation-induced conversion of the ionic form to elemental iodine could be an important mechanism for revolatilization. If the pH is controlled to a level of 7 or greater, conversion to elemental iodine will be minimal. If the pH is not controlled, however, a relatively large fraction (greater for PWRs than BWRs) of iodine dissolved in containment pools in ionic form could be converted to

elemental iodine in the long-term. With the exception of elemental iodine and noble gases, fission products are assumed to be in aerosol form.

3. PANEL RECOMMENDATIONS AND BASIS FOR RECOMMENDATIONS FOR HIGH BURNUP AND MOX FUELS

As stated discussed earlier, the initial "source term" postulated for the purposes of calculating off-site doses in accordance with 10 CFR Part 100 [2] was published in 1962 in TID 14844 [5]. Over the next 30 years, substantial knowledge about severe light-water reactor (LWR) accidents and the resulting behavior of the released fission products was developed. NUREG-1465 provided a postulated fission product source term released into containment that was based upon the understanding of LWR accidents and fission product behavior developed between 1962 and 1995.

Several elements of NUREG-1465 are noted here for comparison with the current endeavor to develop source terms for high burnup low enriched uranium (LEU) and mixed oxide (MOX) fuels.

First, the release fractions for the source terms of NUREG-1465 were intended to be representative or typical, rather than conservative or bounding values, of those associated with a low-pressure core-melt accident. The source term applicability panel employed the same approach in specifying the source terms for high burnup LEU and MOX fuels.

Second, NUREG-1465 recognized that the source term in the report, particularly gap activity, might not be applicable for fuel irradiated to high burnup levels, stated to be burnup levels in excess of 40 GWd/t. Clearly, the same is true for MOX fuel, given its different composition.

Third, the source terms appearing in NUREG-1465 were developed after extensive examinations of both relevant data and calculations that had been developed prior to 1995 combined with expert elicitation. The current development of the source terms for high burnup LEU and MOX fuels, is best described as solely an expert elicitation, albeit an expert elicitation informed by recent test data and insights available for high burnup LEU and MOX fuels [6-9, 12].

3.1 Panel Organization and Elicitation Process

The following process was used to organize the panel and subsequently elicit panel member input for the gap release, early in-vessel, ex-vessel, and late in-vessel releases into the containment for (1) high burnup PWR fuel, (2) high burnup BWR fuel, and (3) MOX fuel.

1. Panel members were selected based upon their expertise. Three experts participated in the development of the NUREG-1150 source term (J. Giseke, T. Kress, and D. Powers), which were used as the basis for the development of the revised source terms in NUREG-1465. These panel members were enlisted to partially reconstitute the expert's group providing input to NUREG-1150 (and NUREG-1465). The remaining panel members (B. Clement and D.

Leaver) brought similar expertise to the source term applicability effort. Brief biographies of each of the panel members are presented in Appendix A.

2. Recent additions to the databases for high burnup LEU and MOX fuels were presented and reviewed [6-9, 12] before beginning the elicitation process for each fuel.

3. The specific characteristics of the fuel and accident sequence type were established and documented (e.g., fuel composition, burnup, cladding type, low-pressure loss-of-coolant accident [LOCA]).

4. The starting point for panel deliberations for high burnup PWR and BWR fuels and MOX fuel was NUREG-1465, which contains tabulations of both PWR and BWR releases into the containment. Releases for nine radionuclide groups (Noble Gases, Halogens, Alkali Metals, Tellurium, Barium/Strontium, Nobel Metals, Cerium, and Lanthanides groups) for each of four accident phases (gap release, early in-vessel, ex-vessel, and late in-vessel) are tabulated.

5. The input of each panel member was elicited and recorded for each fuel, for each accident phase and for each radionuclide group.

6. For the PWR source term, each release fraction and the rationale were elicited from each panel member. If all panel members agreed on the release fraction for a given radionuclide group, a single value was entered into the table. If one or more panel members offered different values for the release fraction for a given radionuclide group, the value offered by each expert was entered in the table. A summary rationale was prepared for each entry in the table.

   The historical approach to addressing the many radionuclide species has been to group species with similar chemical properties and behaviors together. NUREG-1465 employs nine such groups. For the present effort, if one or more panel members felt that a subdivision of one of the NUREG-1465 radionuclide groups was needed to reflect the different release fractions represented in the database, the group was divided into subgroups and the release fractions elicited for the subgroups (see, for example, the Cerium group in Table 3.1). Subgroups were developed for the original NUREG-1465 groups of Noble Metals, Cerium, and the Lanthanides.

   The approach to specifying the source term in France differs from that in the United States. In France, a value is specified for the gap release, and a second value is specified for the in-vessel release. However, this latter value is the total release (TR) used for the remainder of the accident. Thus, values for the ex-vessel and late in-vessel releases were not provided for this effort, by the French expert (B. Clement). The total release values offered for the in-vessel phase of the accident are flagged in the appropriate tables.

A. Hidaka presented test data collected by the Japan Atomic Energy Research Institute (JAERI) at the first panel meeting. However, a JAERI representative was unable to attend the final two panel meetings. As these were the meetings at which the PWR, BWR and MOX source term values were elicited, only five panel members participated in the elicitation.

The PWR releases into containment for high burnup fuel and summary rationales are presented in Section 3.2 (Table 3.1 and Tables 3.2 through 3.10, respectively).

7. For the BWR source term, the value provided by each panel member was recorded but a rationale for each value offered by each panel member was not elicited because of time constraints. General statements about the similarities and dissimilarities of PWR and BWR were discussed and a summary of the general statements for the BWR source term is presented in Section 3.3. The BWR releases into containment for high burnup fuel are presented in Table 3.11.

8. For the MOX source term, both the value and the rationale offered by each panel member were recorded. The MOX releases into containment and summary rationales are presented in Section 3.4 (Table 3.12 and Tables 3.13 through 3.21, respectively).

9. During the first panel meetings, the experts indicated that the physical and chemical forms of the revised source terms as defined in NUREG-1465 are also applicable to high burnup and MOX fuels. Therefore, during subsequent meetings, the panel members did not discuss this issue.

3.2 PWR Accident Source Term

3.2.1 PWR Accident Sequence

The PWR accident sequence selected for NUREG-1465 [1] was a low-pressure core-melt accident. The panel followed the precedent established for NUREG-1465 and selected a low-pressure core-melt accident as the scenario for which the source term for high burnup PWR fuel was to be evaluated.

3.2.2 PWR Fuel Characteristics

The following fuel characteristics were specified for the high burnup PWR fuel.

- A maximum burnup of 75 GWd/t
- A core average burnup of approximately 50 GWd/t
- Zirlo cladding

As a prelude to the elicitation step, panel members considered the effect of high burnup on PWR fuel through discussions of governing phenomenological processes and issues that can potentially impact core degradation and fission product release behavior of high burnup fuels as compared to LEU fuels [11-12, 22].

3.2.3 PWR Accident Source Term

The PWR releases into containment, as specified by the source term applicability panel, are summarized in Table 3.1.

The NUREG-1465 values for PWR releases into containment are shown in parentheses for each phase for the duration and each radionuclide group. For those cases for which the panel agreed upon a single value, only one value is entered e.g., the duration of the gap release phase or the halogen release fractions during all four phases. However, when the panel members did not agree on a single value for a given release, one value was entered for each panel member. As explained previously, the approach to specifying the source term in France differs from that in the United States. A value is specified for the gap release. A second value is specified for the in-vessel release. However, this latter value is the total release, designated as TR in the table, and there are no further entries for the later phases of the accident.

Both the release fraction for each radionuclide group and an accompanying rationale were elicited from each panel member. A summary rationale was prepared for each release fraction and the panel reviewed this summary. The resulting rationales for each accident phase and for the duration and each radionuclide group are presented in Tables 3.2 through 3.10.

The panel also considered additional research needs during the course of its deliberations. These are also summarized in Tables 3.2 through 3.10. In addition, the panel members were also asked to consider research needs separately from the panel meetings and to document these in a letter. The individual panel member letters identifying research needs are provided in Appendix B.

Table 3.1 PWR Releases Into Containment (High Burnup Fuel)

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.4 (0.5)[1]	1.4 (1.3)	2.0 (2.0)	10.0 (10.0)
Noble Gases	0.05; 0.07; 0.07; 0.07 (0.05)	0.63; 0.63; 0.63; 0.65 (0.95)	0.3 (0)	0 (0)
Halogens	0.05 (0.05)	0.35 (0.35)	0.25 (0.25)	0.2 (0.1)
Alkali Metals	0.05 (0.05)	0.25 (0.25)	0.35 (0.35)	0.1 (0.1)
Tellurium group	0.005 (0)	0.10; 0.30; 0.30; 0.30 (0.05)	0.40 (0.25)	0.20 (0.005)
Barium, Strontium	0 (0)	0.02 (0.02)	0.1 (0.1)	0 (0)
Noble Metals	(0)	(0.0025)	(0.0025)	(0)
Mo, Tc	0	0.15; 0.2; 0.2; 0.2; 0.7TR[2]	0.02; 0.02; 0.2; 0.2; TR	0; 0; 0.05; 0.05; TR
Ru, Rh, Pd	0	0.0025; 0.0025; 0.01; 0.01; 0.2TR	0.0025; 0.02; 0.02; 0.02; TR	0.01; 0.01; 0.01; 0.10; TR
Cerium group	(0)	(0.0005)	(0.005)	(0)
Ce	0	0.0002; 0.0005; 0.01; 0.01; 0.02TR	0.005; 0.005; 0.01; 0.01; TR	0
Pu, Zr	0	0.0001; 0.0005; 0.001; 0.002; 0.002TR	0.005; 0.005; 0.01; 0.01; TR	0
Np	0	0.001; 0.01; 0.01; 0.01; 0.02TR	0.005; 0.005; 0.01; 0.01; TR	0
Lanthanides (one group)	0; 0; 0; 0; 0 (0)	0.0005; 0.002; 0.01 (0.0002)	0.005; 0.01; 0.01 (0.005)	0 (0)
La, Eu, Pr, Nb		0.0002; 0.02TR	0.005; TR	
Y, Nd, Am, Cm		0.0002; 0.002TR	0.005; TR	
Nb		0.002; 0.002TR	0.005; TR	
Pm, Sm		0.0002; 0.002TR	0.005; TR	

[1] The numbers in parenthesis are those from NUREG-1465, Accident Source Terms for PWR Light-Water Nuclear Power Plants (Table 3.13).
[2] TR = total release. The practice in France is to assign all releases following the gap release phase to the early in-vessel phase.

Table 3.2 Rationales for Duration Entries in Table 3.1

Gap Release	The NUREG-1465 duration for this interval is 0.5 hr. The panel concluded that for high burnup fuel, the duration of the interval is 0.4 hr. The end point of the gap release phase is defined as the release of significant fission products and this process is accelerated with high burnup fuel. The shortened time reflects the qualitative understanding that the fuel has restructured, putting more gas near the periphery and accelerating the release kinetics of volatile fission products. Data from French VERCOR experimental program shows that early in-vessel release starts earlier with high burnup fuel. Needs: Perform calculations with a code validated with applicable data such as is being generated in the VERCORS experiment.
Early In-Vessel	The NUREG-1465 duration for this interval is 1.3 hr. This phase ends when the bottom head of the reactor pressure vessel fails. The panel concluded that the total time for accident initiation to the end of the early in-vessel phase should be the same as the value in NUREG-1465, i.e., 1.8 hr., thereby fixing the duration of this interval at 1.4 hr. Panel members thought degradation processes might change with high burnup fuel, but that such outcomes can only be understood with integrated experiments. Needs: Need integrated bundle experiments with high burnup fuel, e.g., PHEBUS. Need to translate the data into models, such that whole-core calculations can be made. Need to investigate experimentally the degradation of cores with high burnup fuel to see if a qualitatively different core degradation model is needed.
Ex-Vessel	The NUREG-1465 duration for this interval is 2 hr. The panel concluded that the same value was applicable for high burnup fuel. The dominant ex-vessel process is the high rate of zirconium oxidation. A significant change in this process is not expected for high burnup fuel.
Late In-Vessel	The NUREG-1465 duration for the late in-vessel phase is 10 hr. The panel concluded that the same value was applicable for high burnup fuel. Several processes dominate during this phase. They are (1) revaporization of deposited radionuclides, (2) degradation of residual fuel in core region, (3) air ingress. Existing analyses only address the first two. Data to assess the contribution of the third process is lacking. Releases due to degradation of residual fuel to the ex-vessel category are considered in the ex-vessel category to narrow the time frame from 10 hours to 2 hours. Until additional data are available, the panel has no justification for changing the duration of this phase. Needs: Experimental investigation of air ingress to investigate the competition between degradation and fission product release is essential.

Table 3.3 Rationales for Noble Gases Entries in Table 3.1

Gap Release	The NUREG-1465 gap release was 0.05. A majority of the panel concluded that an increased release fraction is applicable for high burnup fuel. A value of 0.07 was selected. Data from the Japan Atomic Energy Research Institute (JAERI) was cited. The JAERI test featured high reactivity insertion rates characteristic of a rod ejection accident. The data show an acceleration of gap release with burnup, with a discernable threshold near 42 GWd/t. The highest fuel burnup level discussed for the JAERI test was 50 GWd/t. French data for high burnup fuel indicate a value of about 5% at 60 GWd/t (FLASH 5 Test). A release fraction as high as 0.10 was considered by the panel, but the lower value of 0.07 was adopted to account for the slower rate of energy deposition during a large LOCA and in recognition of the French data. However, the specified burnup for panel consideration is 75 GWd/t, causing, possibly, a higher release than the 5% measured in the French test. One panel member weighted the French results higher than the JAERI results on the basis of prototypicality of the experimental conditions to a low-pressure LOCA and concluded that a lower release rate of 0.05 was more applicable. Needs: The JAERI data was cited by the panel was an important factor in selecting an increased release fraction. However, the panel recognized that the high-energy deposition rate could be an important factor in the increased release fraction. Therefore, the panel concluded that further experiments with high burnup fuel under conditions more representative of a large LOCA are desirable to determine if higher gap releases are realized with high burnup fuel in slower events.
Early In-Vessel	The NUREG-1465 early in-vessel release was 0.95. The panel concluded that a reduced release fraction is applicable but that this conclusion was independent of fuel burnup, i.e., the panel concluded that the release fraction reported in NUREG-1465 is too high. The panel noted that only a portion of the core is involved in the accident through this phase. Therefore, noble gases can only be released from that portion of the core involved in the accident. For the LB LOCA, the panel believes 60-70% of the core is involved in the accident by this stage. A release fraction of 63% was selected.

Table 3.3 Rationales for Noble Gases Entries in Table 3.1 (continued)

Ex-Vessel	The NUREG-1465 ex-vessel release was 0.0. i.e., all noble gases were released by the end of the early in-vessel phase. The panel concluded that the release fraction for high burnup fuel for this phase should be 0.3, which constitutes the remainder of the release for noble gases. The relocated core debris does not contain any noble gases. The panel noted that the gap release and early in-vessel phases are discrete phases in time. However, the ex-vessel and late in-vessel phases overlap in time. The panel recognized that releases would occur in-vessel during the late in-vessel phase. However, the panel elected to bias releases occurring from residual fuel early in the late in-vessel phase by attributing the release to the ex-vessel phase. The panel chose this course because assigning the release to the late in-vessel phase would allocate the release to the 10-hour period associated with the late in-vessel phase. Because the releases are expected to occur early in the late in-vessel phase, the panel concluded that issue of timing, with respect to release fractions, is best handled by associating the release with the ex-vessel phase. NUREG/CR-5747, which was referenced in NUREG-1465 as the basis for the release fraction, would have the total release through the early in-vessel phase of about 80%. The NUREG/CR-5747 values were based upon CORSOR calculations, supplemented by expert elicitation. The CORSOR calculations are known to be conservative for noble gases and thus the panel concluded that a lower release of 0.70 for the early in-vessel phase should be used.
Late In-Vessel	The NUREG-1465 late in-vessel release was 0.0. The panel understands that some release of noble gases in this phase is possible but concluded that the release would be early in the phase. Given the length of the late in-vessel phase (10 hours), the panel elected to assign all releases following the early in-vessel phase to the ex-vessel phase.

Table 3.4 Rationales for Halogen Entries in Table 3.1

Gap Release	The NUREG-1465 gap release was 0.05. The panel concluded that the same value was applicable for high burnup fuel. French reactor data for high burnup fuel indicates that the amount of gas in the gap reaches a value of about 5% at about 60 GWd/t. An in-pile experiment for fuel with a burnup of 50 GWd/t and under LOCA conditions indicates a release of 5%.
Early In-Vessel	The NUREG-1465 early in-vessel release was 0.35. The panel concluded that the same value was applicable for high burnup fuel. Several panel members noted that the releases reported in NUREG-1465 were too large. However, because the NUREG-1465 releases were predicated based on CORSOR calculations, which is believed to overestimate Halogen releases, the two effects are thought to offset, and the release fraction remains the same. Need: Experiments needed to understand how high burnup fuel degrades.
Ex-Vessel	The NUREG-1465 ex-vessel release was 0.25. The panel concluded that the same value was applicable for high burnup fuel. The panel noted that the gap release and early in-vessel phases are discrete phases in time. However, the ex-vessel and late in-vessel phases overlap in time. The panel recognized that releases would occur in-vessel during the late in-vessel phase. However, the panel elected to bias releases occurring from residual fuel early in the late in-vessel phase by attributing the release to the ex-vessel phase. The panel chose this course because assigning the release to the late in-vessel phase would allocate the release to the 10-hour period associated with the late in-vessel phase. Because the releases are expected to occur early in the late in-vessel phase, the panel concluded that issue of timing, with respect to release fractions, is best handled by associating the release with the ex-vessel phase.
Late In-Vessel	The NUREG-1465 late in-vessel release was 0.1. The panel concluded that the release for this phase should be 0.2. The increased release was variously attributed to self-heating of deposited fission products and revaporization off the piping system associated with air ingress and the circulation of air through the piping. Needs: The post-test analyses of PHEBUS data should be reviewed for information and insights regarding revaporization.

Table 3.5 Rationales for Alkali Metals Entries in Table 3.1

Gap Release	The NUREG-1465 gap release was 0.05. The panel concluded that the same value was applicable for high burnup fuel. The gap release is via vaporization into the void volume and neither the chemistry of the Cs and Rb nor temperatures changed significantly for high burnup.
Early In-Vessel	The NUREG-1465 early in-vessel release was 0.25. The panel concluded that the same value was applicable for high burnup fuel. Two factors for retaining the value were indicated. Various cesium compounds are released, some of which have reduced vapor pressures and are, therefore, less volatile. However, hot spots are thought to increase the release. These two processes are thought to offset and the release fraction remains the same. One panel member was supportive of a release fraction of 0.3 based upon separate effects release rates just slightly below the release rates for iodine.
Ex-Vessel	The NUREG-1465 early ex-vessel release was 0.35. The panel concluded that the same value was applicable for high burnup fuel. The panel noted that the gap release and early in-vessel phases are discrete phases in time. However, the ex-vessel and late in-vessel phases overlap in time. The panel recognized that releases would occur in-vessel during the late in-vessel phase. However, the panel elected to bias releases occurring from residual fuel early in the late in-vessel phase by attributing the release to the ex-vessel phase. The panel chose this course because assigning the release to the late in-vessel phase would allocate the release to the 10-hour period associated with the late in-vessel phase. Because the releases are expected to occur early in the late in-vessel phase, the panel concluded that issue of timing, with respect to release fractions, is best handled by associating the release with the ex-vessel phase.
Late In-Vessel	The NUREG-1465 late in-vessel release was 0.1. The panel concluded that the same value was applicable for high burnup fuel. Needs: Experiments are needed to better characterize the revaporization of cesium.

Table 3.6 Rationales for Tellurium Group Entries in Table 3.1[1]

Gap Release	The NUREG-1465 gap release is 0. The panel concluded that a small but increased release fraction of 0.005 is applicable for high burnup fuel. A non-zero value was felt appropriate because gap releases increase with time. The panel felt that the uncertainty was high and that additional test data was needed (see Needs). Needs: Experiments with high burnup fuel under LOCA conditions to measure gap releases are needed. The applicability of the proposed PHEBUS LOCA experiments should be reviewed.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.05. The panel concluded that an increased release fraction is applicable for high burnup fuel. A value of 0.30[1] was specified by a majority of the panel. The panel noted that PHEBUS FPT-1 test results indicate high in-vessel releases of Tellurium, which decays to Iodine. As the PHEBUS tests were designed and directed to fission product measurement, there is justification for basing source term applicability on PHEBUS results. Other tests show lower Te releases, apparently due to binding of the Te with other materials. The Te release is very much dependent upon the manner in which the accident scenario evolves. This can lead to a large variation in Te release. Another issue discussed was the degree to which the Te subsequently interacts with other materials, which would reduce the release to containment. The range of values offered by the panel is between 15% and 35%. The panel noted that even if select the lowest value of 15% is used, this value represents a significant change in regarding the tellurium source term. One panel member stated that in the PHEBUS test, the tellurium release lagged the cesium release. The other factors cited were that the measured values from TMI-2 (TMI-2 core average burnup was 8 GWd/t) shows low tellurium releases and the event duration of 1.4 hours for the PWR source term applicability selected by the panel is shorter than the duration of the FPT-1 test. This panel member stated that a lower release rate of 0.1 to 0.15 was more applicable. Needs: The differences between the PHEBUS, VERCORS and ORNL tests should be reconciled. The perception that Tellurium doesn't come out until 95% of the cladding is oxidized comes from the ORNL and the SFD tests. Experiments with high burnup fuel under both severe accident and LOCA conditions are recommended.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.25. The panel concluded that an increased release fraction is applicable for high burnup fuel. A value of 0.40 was specified. The panel concluded that approximately 60% of the tellurium is released from the fuel, 40% remains in fuel, which will be released from residual fuel after the vessel head fails or from the core-concrete interaction. The panel also recognized that chemical interactions with vapors from the control rods and other materials, tellurium transport is via tellurides, not as elemental tellurium. Consequently tellurium doesn't chemically react with the piping system. Note: The tellurium releases occur in-vessel but after the time that the bottom head fails and so is counted as an ex-vessel release. The release is actually occurring in the vessel and during the transit through the vessel.

Table 3.6 Rationales for Tellurium Group Entries in Table 3.1[1] (continued)

Late In-Vessel	The NUREG-1465 late in-vessel release is 0.005. The panel concluded that an increased release fraction is applicable for high burnup fuel. A value of 0.20 was specified. In contrast to the NUREG-1465 assessment in which the total release fraction through the late in-vessel phase was 0.305, the panel concluded that all tellurium will be released by the end of this phase. Several important processes were identified. First, the amount of tellurium present is significant and it is not bonded to surfaces. Rather, it is bonded to aerosols and more easily released. Second, once the piping system is opened and oxygen partial pressure increases, all the tellurides are oxidized to TeO and telluric acid, both of which are highly volatile.

[1] The changes in the Tellurium releases from NUREG-1465 reflect new insights from test programs and are not associated with high burnup fuel specifically.

Table 3.7 Rationales for Barium and Strontium Entries in Table 3.1

Gap Release	The NUREG-1465 gap release is 0.0. The panel concluded that the same value was applicable for high burnup fuel.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.02. The panel concluded that the same value should be retained for high burnup fuel. Modeling results were considered[1]. Barium and strontium are modeled separately and a release of 0.03 is predicted. Given the modeling uncertainties, there was insufficient justification to alter the NUREG-1465 value. There is experimental evidence that barium is much more volatile than strontium; the VERCORS and HI/VI (ORNL) experiments cited. The panel discussed whether the two should be considered separately but retained the NUREG-1465 grouping. [2] Needs: experimental investigations to resolve competing effect and the absolute value of the release fraction are needed for high burnup fuel. The increase of the diffusion coefficient for these species and the oxygen potential that depresses the volatility of both species were identified as the competing effects.
Ex-Vessel	The NUREG-1465 early ex-vessel release is 0.1. The panel concluded that the same value should be retained for high burnup fuel. Thermodynamic calculations performed by the French support a combined release of 0.1 with .01 being barium and 0.09 strontium. The release of barium and strontium ex-vessel is proportional to the amount of zirconium metal that comes ex-vessel. However, the degree to which high burnup affects the amount of ex-vessel metal is unknown. Lacking this information, which again supported the retention of the NUREG-1465 value. Needs: An improved understanding of high burnup fuel degradation is needed; experimental data is required.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0. The panel concluded that the same value should be retained for high burnup fuel. Uncertainties in the resuspension assessment were noted. Large deposits of barium and strontium have been found above the fuel and the potential for release of this material exists (See Needs). Need: A better understanding of resuspension and revolatized processes is needed; experimental data are required.

[1] Results were presented to the panel by T. Kress

[2] Although barium and strontium behave differently, they are treated as a class and a representative value of the release fraction used. For background, the panel was informed that in the draft NUREG-1465, barium and strontium were treated separately. They were combined for the final document because they were judged at that time to have similar behavior.

Table 3.8 Rationales for Noble Metals Entries in Table 3.1

Gap Release	The NUREG-1465 gap release is 0. The panel noted that the release is non-zero but sufficiently small that the NUREG-1465 value was retained.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.0025. The range of experimental results is high and uncertainties are large. The ability to differentiate via prediction does not exist. The four species in this group exhibit differences in behavior. Several of the species are very volatile (Mo, Tc). Based on VERCORS data, Mo and Tc release 90% of the noble metals from fuel and 70% to containment. The Ru release is 10% from the fuel and 2% to the containment. The Rh release is 30% from the fuel and 6% to the containment. The panel created two subgroups. The first contains Mo and Tc and the second contains Ru and Rh. Several factors led to the creation of two groups. Within the original single grouping, there were differences in volatility and biological effects. Temperature increases were cited as an important factor affecting release rates. It was noted that even if temperatures remain nearly constant and fuel motion occurs, the release rates are affected. Similarly, gas release rates at constant temperature can affect release fractions. The original groupings were influenced by MELCOR capabilities and modeling approaches. The choice by the panel to divide the Noble Metals into two subgroups may impact analytical code requirements. Needs: Need data that can be used in model (various releases at various times and temperatures; either data with time/temperature plateaus or several tests as a function of burnup). Need to distinguish between Ru and Mo).
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.0025. Ruthenium is under predicted in VANESSA by perhaps three orders of magnitude. However, this information was available to the NUREG-1465 panel and should, therefore, be factored into the NUREG-1465 release fraction. Needs: correct code models to handle increased Ru releases.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0.0. The noble metals are released and deposited but whether they are revaporized is unclear (See Needs). Needs: An improved understanding of revaporization processes for noble metals is needed; experimental data are required.

Table 3.9 Rationales for Cerium Group Entries in Table 3.1

Gap Release	The NUREG-1465 gap release was 0. The panel concluded that the same value was applicable for high burnup fuel.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.0005. Several of the panel members supported a higher release fraction. Based on VERCORS and other experiments the Np release is 0.10 from the fuel and 0.02 to the containment. The Pu release from the fuel is 0.01 and 0.001 to the containment. A significant release of Ce was measured in the VI5 test and this outcome was also seen in some VERCORS experiments. By analogy with the release of La as measured in VERCORS, a Ce release from the fuel of 0.1 and 0.02 to the containment is predicted. The panel created three subgroups. The first consists of Ce, the second contains Pu and Zr and the third consists of Np. Several factors led to the creation of the three groups. The differing biological impacts of the radionuclides in the three subgroups was indicated. Also considered were the indications of different release rates obtained in the available experiments. Finally, there were differences in the inventory of the species that favored creation of three subgroups. The choice by the panel to divide the Cerium group into three subgroups may impact analytical code requirements. The panel moved the fission product Zr from the Lanthanides group to the Cerium group. The panel noted that Zr had a very low volatility, much like Pu. Also, it is a tetravalent species, as are the other species in the Cerium group. Needs: An improved understanding of how the oxidation potential in high burnup fuel is behaving. Transient data with two or more plateaus for use in models.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.005. Two components for the release were noted, mechanical release as bubbles burst at the surface and vapor release dependent upon the amount of zirconium present.
Late In-Vessel	The NUREG-1465 gap release was 0. The panel concluded that the same value was applicable for high burnup fuel. There is no late in-vessel release per the panel findings. Needs: An improved understanding of the volatilities as well as revaporization of the species in the cerium group is needed; experimental data are required.

Table 3.10 Rationales for Lanthanides Entries in Table 3.1

Gap Release	The NUREG-1465 gap release is 0. The panel concluded that the same value was applicable for high burnup fuel.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.0002. Three of the panel members concluded that the original single group was still applicable because they exhibited similar chemical behaviors. Another panel member concluded a different release rate was warranted only for Nb and only for the early in-vessel phase. Otherwise, a single release rate was deemed adequate by this panel member to represent the release rates for the remaining species within the group as well as for the other phases of the accident. Another panel member felt that there was justification for creating two subgroups. The first subgroup consists of La, Eu, Pr and Nb. These species are characterized by a 0.1 release from the fuel and 0.02 total release to the containment for the early in-vessel phase. The second group consists of the remaining species. Needs: Data on the Lanthanides lacking; data to be obtained for and processed through models.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.005. The panel concluded that the same value was applicable for high burnup fuel.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0. The panel concluded that the same value was applicable for high burnup fuel. Needs: Assuming the existence of La deposits, an improved understanding of the resuspension is needed; experimental data are required.

3.3 BWR Accident Source Term

3.3.1 BWR Accident Sequence

The BWR accident sequence selected for NUREG-1465 [1] was a low-pressure core-melt accident.

The panel followed the precedent established for NUREG-1465 and selected a low-pressure core-melt accident, as the scenario for which the source term for high burnup fuel was evaluated.

3.3.2 BWR Fuel Characteristics

The following fuel characteristics were specified for the high burnup BWR fuel.

- A maximum burnup of 75 GWd/t.
- A core average burnup of approximately 50 GWd/t.
- The cladding material is Zircaloy-2, most predominately in the annealed, fully recrystallized condition with a zirconium-based inner liner, although cold-worked stress relieved material and non-liner applications also exist. The zirconium liner can contain varying amounts of alloy additions, intended for post-defect corrosion resistance.

3.3.3 BWR Accident Source Term

The BWR releases into containment, as specified by the source term applicability panel, are summarized in Table 3.11.

The NUREG-1465 values for BWR releases into containment are shown in parentheses for each phase and for the duration and each radionuclide group. For those cases for which the panel agreed upon a single value, only one value is entered (e.g., the duration of the gap release phase or the halogen releases during the gap release phase). However, when the panel members did not agree on a single value for a given release, one value was entered for each panel member.

As France does not deploy BWRs, the French panel member, while contributing to general technical discussions about the source term, did not participate in the specification of numerical values for BWR releases into containment.

The panel operated under time constraints that limited the collection of rationales for values offered by each panel member for the BWR releases into containment. In an effort to accommodate to the time constraints, the panel first discussed PWR and BWR differences that might affect the source term. The following items were discussed as a prelude to eliciting values for the BWR releases into containment.

- BWRs have a higher zirconium inventory than PWRs due to the presence of channel boxes. This factor was accounted for in NUREG-1465.
- The BWR power density is lower, approximately 60% of a PWR, and the system contains more water that would tend to slow the progression of accident sequences.
- Steam separators in the BWR upper plenum can serve as deposition sites. A panel member questioned whether this effect had been considered in NUREG-1465.
- BWRs employ boron carbide control rods rather than the silver-indium-cadmium control rods used in PWRs. Several panel members noted that the different materials would affect tellurium behavior. This effect was thought to have been considered in NUREG-1465.
- The BWR power profile is thought to be flatter. This could alter, relative to the PWR, the fraction of the core affected by an accident and thus the amount of core experiencing early melting and relocation. It was the belief of some panel members that this factor had not been accounted for in NUREG-1465.

Also considered by the panel was the power upgrade effort for BWRs that will lead to a 20% increase in core power for all or nearly all operating BWRs.

Individual rationales were not elicited for each accident phase and radionuclide group for the BWR.

The panel did not consider additional BWR-specific research needs during the course of its deliberations. As with the PWR source term, the panel members were asked to consider research needs separate from the panel meetings and to document these in a letter. The individual panel member letters identifying research needs are provided in Appendix B.

Table 3.11 BWR Releases Into Containment (High Burnup Fuel)

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.4 (0.5)[1]	1.6 (1.5)	3.0 (3.0)	10.0 (10.0)
Noble Gases	0.05; 0.07; 0.07; 0.07 (0.05)	0.65; 0.76; 0.76; 0.93 (0.95)	0; 0.17; 0.17; 0.3 (0)	All 0 (0)
Halogens	All 0.05 (0.05)	0.25; 0.30; 0.35; 0.40 (0.25)	0.15; 0.2; 0.3; 0.35 (0.30)	0.1; 0.1; 0.12; 0.2 (0.01)
Alkali Metals	All 0.05 (0.05)	0.2; 0.25; 0.3; 0.4 (0.20)	0.1; 0.3; 0.35; 0.35 (0.35)	0.1; 0.1; 0.2; 0.22 (0.01)
Tellurium group	All 0.005 (0)	0.05; 0.06; 0.06; 0.1 (0.05)	All 0.25 (0.25)	0.005; 0.01; 0.01; 0.01 (0.005)
Barium, Strontium	0 (0)	0.02 (0.02)	0.1 (0.1)	0; 0; 0; 0.01 (0)
Noble Metals	(0)	(0.0025)	(0.0025)	(0)
Mo, Tc	0	0.0025; 0.2; 0.2; 0.2	0.0025; 0.02; 0.2; 0.2	0; 0; 0; 0.05
Ru, Rh, Pd	0	0.0025; 0.0025; 0.0025; 0.01	0.0025; 0.0025; 0.02; 0.02	0; 0.01; 0.01; 0.10
Cerium group	(0)	(0.0005)	(0.005)	(0)
Ce	0	0.0002; 0.0005; 0.01; 0.01	0.005; 0.005; 0.01; 0.01	0
Pu, Zr	0	0.0001; 0.0005; 0.001; 0.002	0.005; 0.005; 0.01; 0.01	0
Np	0	0.001; 0.01; 0.01; 0.01	0.005; 0.005; 0.01; 0.01	0
Lanthanides (one group)	0; 0; 0; 0; 0 (0)	0.0005; 0.002; 0.01 (0.0002)	0.005; 0.01; 0.01 (0.005)	0 (0)
La, Eu, Pr, Nb		0.0002; 0.02TR	0.005; TR	
Y, Nd, Am, Cm		0.0002; 0.002TR	0.005; TR	
Nb		0.002; 0.002TR	0.005; TR	
Pm, Sm		0.0002; 0.002TR	0.005; TR	

[1] The numbers in parenthesis are those from NUREG-1465, Accident Source Terms for BWR Light-Water Nuclear Power Plants.(Table 3.12)

3.4 MOX Accident Source Term for Light-Water Nuclear Power Plants

The panel agreed upon an approach to be taken in considering the MOX releases into containment. The approach consisted of the following elements.

- The tabulated fission product release fractions are for MOX assemblies only.
- Assume MOX assemblies are distributed uniformly throughout the core.
- Whether MOX or LEU, the assembly undergoes the same thermal transient.
- The MOX assembly passes through a temperature transient that damages the fuel. LEU assembly also passes through a temperature transient that damages the fuel.
- At the end of the early in-vessel phase, 50% of the core is badly damaged, some molten and some otherwise disrupted and will be released immediately upon failure of the vessel lower head.
- Some portion of the 50% of the core remaining in the vessel is also damaged and participates in the early in-vessel release. Three-fifths of the remaining core loses one-half of its volatile inventory.
- During the ex-vessel phase, 100% of the core eventually ends up outside the vessel and on the concrete floor.
- For the late in-vessel phase (10 hour phase), the material deposited on the reactor cooling system internals is revaporized.

The panel also addressed and developed a methodology for applying the values for the MOX releases into containment to an entire core containing both MOX and LEU fuel assemblies. The application process is as follows.

- To apply results of these tables to a core containing both MOX and LEU assemblies, define the fraction of the MOX fuel in the core to be "f". Define the fraction of LEU to be "1-f".

  $I_{MOX}(i)$ = inventory of the i^{th} radionuclide in the MOX fuel
  $I_{LEU}(i)$ = inventory of the i^{th} radionuclide in the LEU fuel
  $RF_{MOX}(i)$ = release fraction of the i^{th} radionuclide from the MOX fuel
  $RF_{LEU}(i)$ = release fraction of the i^{th} radionuclide from the LEU fuel

  Then, the release fraction from the mixed core is:

$$RF(i) = \frac{f * RF_{MOX}(i) * I_{MOX}(i) + (1-f) * RF_{LEU}(i) * I_{LEU}(i)}{f * I_{MOX}(i) + (1-f) * I_{LEU}(i)}$$

- For high burnup LEU fuel, the values are those in the tables generated by the panel (i.e., Tables 3.1 and 3.11, for PWRs and BWRs, respectively).
- For LEU fuel that is not at high burnup, use the values from NUREG-1465, even though some of the values could be updated based upon the database now available.

3.4.1 MOX Accident Sequence

The accident sequence is identical to the low-pressure accident sequence of Section 3.2.1.

3.4.2 MOX Fuel Characteristics

The following fuel characteristics were specified for the MOX fuel.

- The MOX fuel pellets are 5% PuO_2 with approximately 95% depleted UO_2.
- The plutonium is approximately 93% Pu-239 and 6% Pu-240.
- The cladding is M5.
- The fuel assembly is identical to LEU assemblies, except for the fuel pellet.
- The cycle length is 18 months. MOX assemblies are typically withdrawn after two cycles.
- The maximum burnup on an assembly basis is approximately 46 GWd/t
- The typical burnup on an assembly basis is approximately 42 GWd/t
- The burnup limit on a pin is 50 GWd/t.
- The planned core loading is for approximately 40% MOX assemblies.
- The MOX assemblies will be irradiated in 4-loop Westinghouse PWRs with ice condenser containments.

Several differences between LEU and MOX fuels were noted. First, MOX fuel experiences high power at higher burnup levels. Second, the plutonium in discharged high burnup LEU fuel is approximately 1%. MOX assemblies are discharged with approximately 3% plutonium.

3.4.3 MOX Accident Source Term

The MOX releases into containment, as specified by the source term applicability panel, are summarized in Table 3.12. A summary rationale was prepared for each release fraction, and this summary was reviewed by the panel. The resulting rationales for each accident phase and for the duration and each radionuclide group are presented in Tables 3.13 through 3.21.

Numerous times during the elicitation of MOX releases into the containment, panel members emphasized that their input was based upon partial and preliminary data regarding MOX characteristics and behavior under severe accident conditions.

MOX releases were provided by all five panel members for the phase duration and for the noble gases, halogens, alkali metals, and tellurium groups. However for the remaining four groups (barium and strontium, noble metals, cerium group, and lanthanides), some panel members elected to make no entry because it was felt that there was insufficient data available upon which to make an informed judgment. The panel members noted that all the noble gases, halogens, alkali metals and tellurium group are released from a MOX fuel. Thus, it was possible to make judgments as to the phase in which they were

released. However, for barium and strontium, the noble metals, cerium group, and lanthanides, only fractional releases occur and the database was deemed insufficient by some panel members to support a specific value for a release fraction.

The NUREG-1465 values for PWR releases into containment are shown in parentheses for each phase and for the duration and each radionuclide group. For those cases for which the panel agreed upon a single value, only one value is entered e.g., the duration of the gap release phase or the halogen releases during all four phases. However, when the panel members did not agree on a single value for a given release, one value was entered for each panel member.

A discussed in Section 3.1, the approach to specifying the source term in France differs from that in the United States. A value is specified for the gap release. A second value is specified for the in-vessel release. However, this latter value is the total release used for the remainder of accident. Thus, values for the ex-vessel and late in-vessel releases were not provided for this effort. The total release values offered for the in-vessel phase of the accident are flagged in the appropriate tables.

The panel only briefly considered additional MOX-specific research needs during the course of its deliberations. As with the PWR and BWR source term, the panel members were asked to document their opinions regarding research needs. The individual panel member letters identifying research needs are provided in Appendix B.

Table 3.12 MOX Releases Into Containment

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.3; 0.4; 0.4; 0.4; 0.4 (0.5)[1]	1.3; 1.4; 1.4; 1.4; 1.5 (1.3)	2.0 (2.0)	10.0 (10.0)
Noble Gases	0.05; 0.05; 0.05; 0.05; 0.07 (0.05)	0.65; 0.65; 0.75; 0.93; 0.95 TR[2] (0.95)	0; 0.2; 0.3; 0.3; TR (0)	0 (0)
Halogens	0.05; 0.05; 0.05; 0.05; 0.07 (0.05)	0.325; 0.35; 0.35; 0.375; 0.95TR (0.35)	0.15; 0.2; 0.25; 0.25; TR (0.25)	0.2; 0.2; 0.2; 0.2; TR (0.1)
Alkali Metals	0.05; 0.05; 0.05; 0.05; 0.07 (0.05)	0.30; 0.30; 0.30; 0.30; 0.65TR (0.25)	0.25; 0.25; 0.30; 0.30; TR (0.35)	0.10; 0.15; 0.15; 0.15; TR (0.1)
Tellurium group	0; 0; 0; 0.005; 0.005 (0)	0.1; 0.15; 0.3; 0.35; 0.7TR (0.05)	0.4; 0.4; 0.4; 0.4; TR (0.25)	0.1; 0.2; 0.2; 0.2; TR (0.005)
Barium, Strontium	NE[3], NE, NE; 0; 0 (0)	NE, NE, NE; 0.1; 0.1 (0.02)	NE, NE, NE; 0.1; 0.1 (0.1)	NE, NE, NE; 0; 0.05 (0)
Noble Metals	(0)	(0.0025)	(0.0025)	(0)
Mo, Tc	NE, NE, NE; 0; 0	NE, NE, NE; 0.1; 0.05	NE, NE, NE; 0.01; 0.01	NE, NE, NE; 0.1; 0.01
Ru, Rh, Pd	NE, NE, NE; 0; 0	NE, NE, NE; 0.1; 0.1	NE, NE, NE; 0.01; 0.01	NE, NE, NE; 0.1; 0.01
Cerium group	(0)	(0.0005)	(0.005)	(0)
Ce	NE, NE, NE; 0; 0	NE, NE, NE; NE; 0.01	NE, NE, NE; 0.01; 0.01	NE, NE, NE; NE; 0
Pu, Zr	NE, NE, NE; 0; 0	NE, NE, NE; NE; 0.001	NE, NE, NE; 0.002; 0.001	NE, NE, NE; NE; 0
Np	NE, NE, NE; 0; 0	NE, NE, NE; NE; 0.01	NE, NE, NE; 0.01; 0.02	NE, NE, NE; NE; 0
Lanthanides	NE, NE, NE; 0; 0 (0)	NE, NE, NE; NE; 0.005 (0.0002)	NE, NE, NE; NE; 0.01 (0.005)	NE, NE, NE; NE; 0 (0)

[1] The numbers in parenthesis are those from NUREG-1465, Accident Source Terms for PWR Light-Water Nuclear Power Plants (Table 3.13).
[2] TR = total release. The practice in France is to assign all releases following the gap release phase to the early in-vessel phase.
[3] NE = No entry; the panel member concluded that there was insufficient information upon which to base an informed opinion.

Table 3.13 Rationales for Duration Entries in Table 3.12[1]

Gap Release	The NUREG-1465 duration for this interval is 0.5 hr. The panel concluded that for MOX fuel, the duration of this interval should be shortened. Four panel members specified a duration of 0.4 hr to indicate the direction of the change, while still acknowledging the limited data available. Data from the VERCORS RT2 experiment as well as the Halden data were considered by these panel members as a sufficient indication of trend to warrant a reduction in the length of the period. Based upon the same information, one panel member specified a duration of 0.3 hr. Needs: Data on duration of gap release and progression of fission gas loading of cladding for representative scenarios.
Early In-Vessel	The NUREG-1465 duration for this interval is 1.3 hr. Three panel members concluded that the time to melt through the bottom head, i.e., the total of the gap release and early in-vessel phases, is the same for LEU and MOX at 1.8 hrs. Given that the gap release phase for MOX was shorter than for LEU, the early in-vessel phase duration was determined to last 1.4 hr. Another panel member also concluded that the time to failure of the lower head was 1.8 hr. Having specified a duration of 0.3 hrs for the gap release phase, the early in-vessel phase duration was determined to last 1.5 hr. The final panel member concluded that the total time to vessel failure was shorter (i.e., 1.7 hr), and having specified a duration of 0.4 hr for the gap release phase, determined a value of 1.3 hr. was appropriate for the duration of the early in-vessel phase Needs: MOX bundle degradation test to characterize fuel relocation and associated fission product transport.
Ex-Vessel	The NUREG-1465 duration for this interval is 2 hrs. The panel concluded that the same value was applicable for MOX fuel. The release for this phase is composed of two parts. The first is the continued degradation and expulsion of the core remaining within the vessel at the end of the early in-vessel phase. The second is the release with the core-concrete interaction. The panel felt that neither the early in-vessel release nor the release with core-concrete interactions were changed by a sufficient amount to change the duration of this phase.
Late In-Vessel	The NUREG-1465 duration for this interval is 10 hrs. The panel concluded that the same value was applicable for MOX fuel. Resuspension and revaporization are the key processes for this phase of the accident. The 10 hr. period applies if the late in-vessel releases during this phase are the same as for LEU. However, if there are substantial releases of Te or Sr, these would enhance revaporization and could alter the duration of this phase.

1 Panel member inputs for Table 3.13 and the remaining MOX rationale tables are based upon partial and preliminary data regarding MOX characteristics and behavior available to the panel at the time the source term table input was prepared.

Table 3.14 Rationales for Noble Gases Entries in Table 3.12

Gap Release	The NUREG-1465 gap release for LEU fuel was 0.05. A majority of the panel concluded that the same gap release was applicable for MOX fuel and specified 0.05 as the release fraction to the containment. Based upon the information presently available, these panel members concluded that the 0.05value for the release was still applicable. This value is based upon the database available to the panel, the limitations on burnup of MOX fuel, and a conclusion that while the MOX inventories for noble gases are higher than for LEU, they are still likely to be within the 0.05 level. One panel member concluded that the noble gas gap inventory was larger for MOX fuel and specified a value of 0.07. Needs: For future large LOCA experiments, would recommend measurements of noble gases.
Early In-Vessel	The NUREG-1465 gap release for LEU fuel was 0.95, i.e., the entire inventory of noble gases was released during the early in-vessel phase. One panel member concluded that the entire inventory of noble gases in MOX fuel was released in this phase and provided a release fraction of 0.93. Note that the gap release fraction offered by this panel member was 0.07. As noted in Section 3.1, The approach to specifying the source term in France differs from that in the United States. A value is specified for the gap release. A second value is specified for the in-vessel release. However, this latter value is the total release used for the remainder of accident. Thus, values for the ex-vessel and late in-vessel releases were not provided for this effort. The total release (TR) value for the noble gases was 0.95. The remaining panel members concluded that only a portion of the noble gases were released during the early in-vessel phase and that subsequent releases occurred during the ex-vessel phase. Two panel members estimated the release fraction to be 0.65, based upon an estimate of the amount of fuel either with failed cladding or melted down. One panel member estimated the release to be 0.75.
Ex-Vessel	The NUREG-1465 ex-vessel release for LEU fuel was 0.0. Three panel members determined that there would be a release during this phase with two providing values of 0.3 and the third a value of 0.2. One panel member estimated the release to be 0. The final panel member was the French delegate and a release of 0 (TR) was specified per the French convention.
Late In-Vessel	The NUREG-1465 late in-vessel release for LEU fuel was 0.0. The panel concluded the same value applied for MOX fuel.

Table 3.15 Rationales for Halogens Entries in Table 3.12

Gap Release	The NUREG-1465 gap release was 0.05. A majority of the panel concluded that the same gap release was applicable for MOX fuel and specified 0.05 as the release fraction to the containment. Based upon the information presently available, these panel members concluded that the 0.05 value for the release was still applicable. This value is based upon the database available to the panel, the limitations on burnup of MOX fuel, and a conclusion that while the MOX inventories for halogens are higher than for LEU, they are still likely to be within the 0.05 level. One panel member concluded that the halogen gap inventory was larger for MOX fuel and specified a value of 0.07. Needs: The panel noted that there is essentially no data for halogen releases from MOX fuel and that data is needed.
Early In-Vessel	The NUREG-1465 early in-vessel release was 0.35. Ex-vessel releases between 0.325 and 0.375 were specified by the panel members. One panel member concluded that the early in-vessel release was 0.325 based upon the observation that there is a higher deposition of vapors and aerosols on the piping system for MOX fuels and thus a smaller release fraction. Two panel members concluded the halogen release fraction is 0.35, the same as for LEU. One panel member concluded that there were no significant changes in the assumptions associated with halogens releases from MOX fuel that would require a change in the release. The logic used by the second panel member considered both the fraction of MOX assemblies leaving the vessel as well as the LEU and a total release of the halogens for these materials to arrive at the 0.35 release. One panel member concluded that the halogens release was 0.375. The rationale was that for MOX there is a larger release than for LEU and at a faster rate. The numerical logic presented was based upon one-half the core in LEU assemblies and the other half in MOX, assemblies. Melting and/or significant thermal damage to 70 percent of the core is taken and assuming a total release of the halogens, this leads to a release fraction of 0.35. Given the higher releases from the MOX, could lead to a value of 0.4 but this was reduced to 0.375 to account for the greater deposition rates. A total release or TR of 0.95 was specified by the French panel member and this release was for the totality of the remaining scenario. Needs: As with the gap release, the panel noted that there is essentially no data for halogen releases from MOX fuel and that data is needed. The panel also noted that data encompassing damage progression is crucial.

Table 3.15 Rationales for Halogens Entries in Table 3.12 (continued)

Ex-Vessel	The NUREG-1465 ex-vessel release was 0.25. Ex-vessel releases between 0.15 and 0.25 were specified by the panel members, one at 0.15, one at 0.20 and two at 0.25. The value of 0.15 was specified based upon a calculation algorithm used by the panel member. The value of 0.2 was provided without a discussion of rationale. The panel members offering the value of 0.25 stated that they had no basis for changing the NUREG-1465 value. A total release or TR of 0 was specified by the French panel member as explained for the early in-vessel phase. Needs: The panel identified the need for information about core damage progression and also for inventory data and data related to air ingress and revaporization.
Late In-Vessel	The NUREG-1465 late in-vessel release was 0.1. The panel concluded that a value of 0.2 should be used for the late in-vessel release of MOX fuel, although different rationales were given. Two panel stated that they had no basis for changing the NUREG-1465 value, one based the value on the results a calculation algorithm, and the final number was provided with no discussion. A total release or TR of 0 was specified by the French panel member as explained for the early in-vessel phase. Needs: The panel identified the of lack information on revaporization but noted that this deficiency is not specific to MOX; the same deficiency also applies to LEU. The lack of air ingress data was also noted, i.e., the impact of air on the revaporization process.

Table 3.16 Rationales for Alkali Metals Entries in Table 3.12

Gap Release	The NUREG-1465 gap release is 0.05. Four panel members concluded that the same gap release was applicable for MOX fuel and specified 0.05 as the release fraction to the containment. The rationale was that while the inventory is probably a little higher, the value of 0.05 has sufficient margin to reflect the increase, particularly when the burnup is limited to 40 GWd/t. One panel member, citing consistency with his contributions for the noble gases and halogens specified a higher alkali metals gap inventory of 0.07. Needs: The panel stated that the primary need was to understand the Cs data that currently exists from the VERCOR program
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.25. The alkali metals release fraction for MOX during the early in-vessel phase is 0.30. The panel stated that the MOX alkali metals release fraction is greater than for LEU. A total release or TR of 0.65 was specified by the French panel member and this release was for the totality of the remaining scenario. Needs: See gap release
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.35. Two panel members concluded the ex-vessel release was 0.25 and two panel members concluded the release was 0.3. Rationales were not provided during the panel meeting. The final panel member was the French delegate and a release of 0 (TR) was specified per the French convention. Needs: See gap release.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0.1. One panel member concluded that the late in-vessel release was 0.10 and three panel members concluded the release fraction was 0.15. No rationale was provided during the meeting for the 0.10 value. For the 0.15 release fraction, it was noted that the fractional releases of the alkali metals during the late in-vessel phase are about the same as for NUREG-1465, but the environment is such that more deposition of cesium on the parent piping system is expected, creating more refractory compounds, and reducing the efficiency of the revaporization process. The final panel member was the French delegate and a release of 0 (TR) was specified per the French convention. Needs: See gap release.

Table 3.17 Rationales for Tellurium Group Entries in Table 3.12

Gap Release	The NUREG-1465 gap release is 0. For MOX fuel, three panel members concluded the gap release is 0. One noted that based upon its volatility, the tellurium releases should be as large as for the halogens. However, it also binds with the metallic elements in the fuel, reducing its volatility. Two panel members felt that a small gap release of 0.005 was appropriate.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.05. Early in-vessel tellurium releases between 0.1 and 0.35 were offered by the panel members. Given MOX and M5 cladding, the panel member offering the 0.1 value expects slightly higher releases of molybdenum and ruthenium for MOX and the same for tellurium. However there will be higher concentrations of the reactive forms of tellurium in the release and these will interact with the surfaces, causing a significant fraction of the released products to be deposited on the piping system. The release fraction of 0.1 is the net of all these processes. A value of 0.15 was offered by another panel member who notes that there is an increased release relative to LEU but not as high as 0.3 offered by other panel members. Values of 0.3 and 0.35 were offered. No rationale was provided during the meeting for the former. For the latter, it was assumed that tellurium is released in the same amount as for the halogens. A total release or TR of 0.7 was specified by the French panel member and this release was for the totality of the remaining scenario. Needs: Data are needed on the interaction of MOX fuel and M5 cladding.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.25. The panel concluded that a release rate of 0.4 applies to the ex-vessel release. One panel member noted that melt-concrete interactions are well understood as experiments have been done in which tellurium interactions were explicitly considered. The final panel member was the French delegate and a release of 0 (TR) was specified per the French convention. Needs: See early in-vessel.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0.005. Three panel members concluded the late in-vessel release was 0.2 and one panel member concluded the release was 0.1. The rationale for the 0.2 release fraction was that a lot of tellurium is released from the fuel and deposited on the piping system during the earlier phases of the accident. For reactors having air-filtered containments there will be air ingress and this will react to turn any tellurides into tellurium oxide, a volatile compound that will be released. The rationale for the 0.1 release was that once tellurium gets tied up on the surfaces, it won't subsequently be released (revaporized). The final panel member was the French delegate and a release of 0 (TR) was specified per the French convention. Needs: See early in-vessel.

Table 3.18 Rationales for Barium and Strontium Entries in Table 3.12

Gap Release	The NUREG-1465 gap release is 0. Three panel members concluded that the gap releases for barium and strontium were 0. They noted that the temperature is low and these species are not very volatile and will not vaporize at these temperatures. One panel member declared there was insufficient data to support a value. The French panel member also declined to provide a specific value and stated that the analysis of applicable data was in progress but not yet to the point where the results could be disseminated. Although a value was not entered, releases for which the on-going analyses might result in a change from the NUREG-1465 value were "flagged" and a brief comment provided. Need: Perform a LOCA test with MOX and measure the gap release.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.02. Two panel members declared they were unable to provide a value[1]. The French panel member also declined to provide a value but did describe two flags. The first flag was that data from PHEBUS and data from VERCORS are not consistent. This is one of the areas currently being analyzed by the French. The second flag was that the results for strontium have not yet been sufficiently analyzed. One panel member concluded that the early in-vessel release fraction was 0.01 and noted that the fuel is inherently oxidizing, which would tend to suppress barium releases in concert with increasing noble metals in response to the oxidizing environment. The stated release fraction was accompanied by a statement that confidence in the value was low. Two panel members concluded that a release of 0.1 was appropriate, based upon observations and a consideration of qualitative representations of the French data. Again, confidence in the value was said to be low.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.1. Two panel members declared they were unable to provide a value. No flags were provided by the French panel member. Three panel members specified a release of 0.1 and noted that the release is driven by Zr presence in the initial transient. Zirconium reduces everything down to barium metal and barium metal vaporizes.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0. Two panel members declared they were unable to provide a value. No flags were provided by the French panel member. Two panel members stated the release was 0. The final panel member stated the release was 0.05 because his values for releases from the vessel were large.

[1] The panel member elected to make no entry because it was felt that there was insufficient data available upon which to make an informed judgment. The panel members noted that all the noble gases, halogens, alkali metals and tellurium group are released from a MOX fuel. Thus, it was possible to make judgments as to the phase in which they were released. However, for barium and strontium, the noble metals, cerium group, and lanthanides, only fractional releases occur and the database was deemed insufficient by some panel members to support a specific value for a release fraction.

Table 3.19 Rationales for Noble Metals Entries in Table 3.12

Gap Release	The NUREG-1465 gap release is 0. Two panel members concluded a release fraction of 0 was appropriate; no rationale was presented during the meeting. Three panel members declined to make an entry[1]. No flags were declared by the French panel member.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.0025. The noble metal group was further subdivided into two subgroups, the first containing Mo and Tc and the second containing Ru, Rh and Pd. Three panel members declined to make an entry for the noble metals group. For Mo and Tc, the remaining two panel members each concluded a release fraction of 0.1 is appropriate. The rationale provided for the release was that the uranium matrix surrounding the particles is where the fission products reside and this matrix oxidizes the fuel and creates the volatile forms of these radionuclides. The French panel member flagged this group to indicate that additional analysis was needed to determine if the value should be different than that specified for LEU fuel. For Ru, Rh, and Pd one of the panel members concluded that a release of 0.05 is appropriate. The second panel member concluded that a release of 0.1 was appropriate.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.0025. Two panel members declined to make an entry. The French panel according to the French approach to source term assigns a total release to the early in-vessel phase and does not declare values for the ex-vessel and late in-vessel phases. For Mo and Tc the remaining two members each concluded that a release fraction of 0.01 is appropriate. For Ru, Rh, and Pd the remaining two members each concluded that a release fraction of 0.01 is appropriate.

Table 3.19 Rationales for Noble Metals Entries in Table 3.12 (continued)

Late In-Vessel	The NUREG-1465 late in-vessel release is 0. Two panel members declined to make an entry. The French panel according to the French approach to source term assigns a total release to the early in-vessel phase and does not declare values for the ex-vessel and late in-vessel phases. For Mo and Tc the remaining two members each concluded that a release fraction of 0.1 is appropriate. For Ru, Rh, and Pd the remaining two members each concluded that a release fraction of 0.01 is appropriate.

1 The panel member elected to make no entry because it was felt that there was insufficient data available upon which to make an informed judgment. The panel members noted that all the noble gases, halogens, alkali metals and tellurium group are released from a MOX fuel. Thus, it was possible to make judgments as to the phase in which they were released. However, for barium and strontium, the noble metals, cerium group, and lanthanides, only fractional releases occur and the database was deemed insufficient by some panel members to support a specific value for a release fraction.

Table 3.20 Rationales for Cerium Group Entries in Table 3.12

Gap Release	The NUREG-1465 gap release is 0. Two panel members concluded a release fraction of 0 was appropriate; no rationale was presented during the meeting. Three panel members declined to make an entry[1]. No flags were declared by the French panel member.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.0005. The cerium group was further subdivided into two subgroups, the first containing Ce and Np and the second containing Pu, Zr, and Np. Four panel members declined to make an entry for the cerium group. The French panel member would separate Ce and Np from Pu. Flags were provided on both groups to indicate that the release fractions may change from those offered for high burnup PWR fuel. For Ce and Np, the remaining panel members concluded a release fraction of 0.01 is appropriate. No rationale was provided during the meeting. For Pu, Zr, and Np the panel member concluded that a release fraction of 0.001 is appropriate. The reduced Pu release compared to the Ce and Np release is a direct consequence of considering a low-pressure scenario. If a high-pressure sequence was being considered, the Pu release would be higher.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.005. Three panel members declined to make an entry for the cerium group. For Ce and Np, the remaining two panel members concluded a release fraction of 0.01 is appropriate. This value is based upon molten-core-concrete-interaction experiments. For Pu, Zr, and Np the remaining two panel members concluded that a release fraction of 0.001 is appropriate.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0. Three panel members declined to make an entry for the cerium group. For Ce and Np, the remaining panel member concluded a release fraction of 0 is appropriate. F or Pu, Zr, and Np the remaining panel member concluded that a release fraction of 0 is appropriate.

[1] The panel member elected to make no entry because it was felt that there was insufficient data available upon which to make an informed judgment. The panel members noted that all the noble gases, halogens, alkali metals and tellurium group are released from a MOX fuel. Thus, it was possible to make judgments as to the phase in which they were released. However, for barium and strontium, the noble metals, cerium group, and lanthanides, only fractional releases occur and the database was deemed insufficient by some panel members to support a specific value for a release fraction.

Table 3.21 Rationales for Lanthanides Entries in Table 3.12

Gap Release	The NUREG-1465 gap release is 0. Two panel members concluded a release fraction of 0 was appropriate; no rationale was presented during the meeting. Three panel members declined to make an entry[1]. No flags were declared by the French panel member.
Early In-Vessel	The NUREG-1465 early in-vessel release is 0.0002. One panel member concluded a release of 0.005 is appropriate. As a point of comparison, it was noted that the lanthanides display less volatility than the cerium group. Confidence in the stated value was said to be low. Four members of the panel declined to make and entry for the lanthanides. The French panel member flagged the release for Eu only to indicate that the release fraction for this radionuclide might change from the value specified for high burnup LEU fuel.
Ex-Vessel	The NUREG-1465 ex-vessel release is 0.005. One panel member concluded a release of 0.01 is appropriate. Four members of the panel declined to make and entry.
Late In-Vessel	The NUREG-1465 late in-vessel release is 0. One panel member concluded a release of 0 is appropriate. Four members of the panel declined to make and entry.

1 The panel member elected to make no entry because it was felt that there was insufficient data available upon which to make an informed judgment. The panel members noted that all the noble gases, halogens, alkali metals and tellurium group are released from a MOX fuel. Thus, it was possible to make judgments as to the phase in which they were released. However, for barium and strontium, the noble metals, cerium group, and lanthanides, only fractional releases occur and the database was deemed insufficient by some panel members to support a specific value for a release fraction.

4. INSIGHTS AND RECOMMENDATIONS

It is emphasized that the radiological release fractions presented in this report are intended to be representative or typical, rather than conservative or bounding values, of those associated with low-pressure core damage scenarios, except for the initial appearance of fission products from failed fuel, which has been chosen conservatively. These release fractions into the containment are not expected to bound all potential severe accident scenarios, or to represent any single scenario/sequence.

It is noted that the panel did not have the benefit of the results of accident sequence analyses using accident analysis models validated by comparison to pertinent test results involving high burnup or MOX fuels. In addition, in many areas, the panel identified gaps in experimental data to support specific panel recommendations. Therefore, the members of the panel have attempted to mentally integrate the results of recent tests to predict fission product releases during accidents at nuclear power plants. They have extrapolated phenomenology of core degradation based on existing studies for conventional burnup of LEU fuels to anticipate fission product releases from fuel at burnup levels in excess of about 60 GWd/t. The panel members have also extrapolated the behavior of LEU fuels with conventional Zircaloy cladding to estimate the behavior of mixed oxide fuel with zirconium-niobium-tin alloy (M5) cladding.

In formulating the proposed changes to the NUREG-1465 source term, for application to reactor accident analyses for high burnup and MOX fuels, attention was also given to the changes in our understanding that have come about because of major experimental investigations of fission product behavior under reactor accident conditions, including the Phebus-FP and the VERCORS experiments.

4.1 High Burnup Fuel

The proposed accident source terms are listed in Tables 3.1 and 3.11, for application to PWR and BWR design basis accident analysis, respectively. Also shown on these tables are the release fractions as proposed in NUREG-1465 [1], for LWRs at lower levels of fuel burnup.

4.1.1 PWRs

The results of the panel recommendation for PWR high burnup fuel (Table 3.1) show that:

1. The **release durations** are not considered to be significantly different from the estimates in NUREG-1465. The shortened duration for the gap release phase reflects the expectation that the start of significant release of fission products is accelerated with high burnup fuel, as evidenced from the French VERCORS experiments. However, since the panel concluded that the total time for accident initiation to the end of the early in-vessel phase should be the same as the value in NUREG-1465, i.e., 1.8 hours, the duration of the "early in-vessel" phase was

increased to 1.4 hours. The potential impact of fuel burnup on core damage progression has been recognized as an area requiring further integral experiments. The duration of release for the late in-vessel and the ex-vessel phases remain unchanged as compared with NUREG-1465.

2. An increase in the **gap release**, as compared to NUREG-1465, is noted only for the noble gases and the tellurium groups. This increase has been influenced by the experimental data of the Japan Atomic Energy Research Institute (JAERI) [9].

3. The fission product releases in the **early in-vessel** phase show changes as compared with NUREG-1465 for all the fission product groups, except for the halogens, the alkali metals, and the barium/strontium groups. The reduction in the release of noble gas group reflects the current understanding that a near complete degradation of reactor fuel prior at the time of vessel breach is not considered to be plausible. Furthermore, the higher releases for tellurium as recommended by the panel, is influenced by the results of the Phebus experiments. The panel recommended regrouping the remaining nuclides to better reflect the differences in thermodynamic and chemical behavior of the various constituents in these groups. Higher releases are estimated for Mo/Tc; however, lower (as compared with Mo/Tc, but similar or higher as compared with NUREG-1465) releases are recommended for Ru, Rh and Pd. These release magnitudes were influenced by the observations from the French VERCORS experiments. Similarly, the panel also recommended dividing the cerium group into three subgroups consisting of Ce; Pu and fission product Zr[*]; and Np. Here again, due to the large uncertainties associated with the early in-vessel release of these nuclides, the range of releases to the containment assigned by the panel members spans over two orders of magnitude for Ce, and Pu/Zr; however, the majority of the panel members have suggested release magnitudes for Np which are higher than that in NUREG-1465 for the cerium group. The variability in the release estimates for the semi-volatile and refractory nuclides reflects the large uncertainties associated with these fission product groups. These changes as recommended by the panel are independent of fuel burnup and are also applicable to fuels at lower burn-up levels.

4. The changes in the recommended releases for the **late in-vessel** and the **ex-vessel** phases follow the observations as noted earlier for the in-vessel phase, where generally, no burnup dependence was noted. The most notable change is due to higher potential for revaporization for halogens, tellurium and noble metal groups, which are also expected to be applicable to lower burnup levels. The variations in the panel recommended releases for the semi-refractory and refractory nuclides as compared to NUREG-1465 reflect the general uncertainties associated with the release of these nuclides during core-concrete interactions, and are not indicative of expectations for any significant burnup dependence.

[*] Fission product Zr was moved from the lanthanide group due to very low volatility and the tetravalent nature of Zr

4.1.2 BWRs

The panel recommended source terms to the containment for BWRs, were developed considering the differences in BWRs and PWRs that can impact the progression of severe accidents, radionuclide releases and their transport characteristics. The recommended changes to the NUREG-1465 releases into BWR containment are also greatly influenced by factors such as the insights from the more recent experimental data, and the impact of new fuel design, and not by differences due to the higher fuel burnup.

The results of the panel recommendation for BWR high burnup fuel as listed in Table 3.11 show that:

1. The **release durations** are not considered to be considerably different from the estimates in NUREG-1465, and the rationale for the shortened duration for the gap release, and subsequently longer duration for the early in-vessel phase follows that for PWRs discussed earlier.

2. As for PWRs, an increase in the **gap release**, as compared to NUREG-1465, is also noted for the noble gases and the tellurium groups.

3. The fission product releases in the **early in-vessel** phase show changes as compared with NUREG-1465 for all the fission product groups, except for the barium/strontium group. These changes reflect more recent experimental observations and the current expectation for more incomplete core degradation prior to vessel breach. Note that the range of recommended releases are similar to those for PWRs as shown in Table 3.1, except for tellurium group, where a lower estimate for in-vessel release is recommended by the panel (NOTE TO PANEL MEMBERS – Please provide the rationale for this lower release of Te in BWRs). Here again, the recommended changes are not influenced greatly by the change in fuel burnup, rather by the more recent experimental evidence and results of current code calculations.

4. The changes in the recommended releases for the **late in-vessel** and the **ex-vessel** phases follow the observations noted earlier for PWRs, where generally, no significant burnup dependence was identified. The most notable changes are due to higher potential for revaporization of the halogens, the tellurium and the noble metal groups, which are also expected to be applicable to lower burnup levels. As for PWRs, the variations in the panel recommended releases for the semi-refractory and refractory nuclides as compared to NUREG-1465 reflect the general uncertainties associated with the release of these nuclides during core-concrete interactions, and are not indicative of expectations for any significant burnup dependence.

4.2 MOX Fuels

An approach has been proposed to calculate the source term to containment for an entire core containing mixed oxide and low enriched uranium fuel assemblies (see Section 3.4 for details). This approach apportions the releases into the containment based on the fraction of the core that contains MOX (and LEU) fuel. The release parameters for the LEU part of the core are to be taken from Tables 3.1 or 3.11, depending on the reactor type; while, the release fractions for the MOX assemblies can follow the panel source term recommendations listed in Table 3.12.

Table 3.12 shows the range of panel recommendation for the various release parameters, reflecting the uncertainties due to lack of an adequate database to characterize radiological releases for MOX fuels.

In general, the duration of release for the various phases are essentially identical to the LEU fuels, with the general expectation that the gap release would occur over a shorter time period, based on the observations from the VERCORS RT2 and Halden test data.

Table 3.12 shows that some panel members concluded that there was insufficient information upon which to base an informed opinion; therefore, these panel members did not provide specific recommendations for the release fractions for other than the more volatile radionuclides. The panel members noted that all the noble gases, halogens, alkali metals and tellurium group are released from a MOX fuel. Thus, it was possible to make judgments as to the phase in which they were released. For the remaining radionuclide groups, only fractional releases occur and the database was deemed insufficient to support a specific release fraction.

In general, the panel concluded that the gap release fractions are similar or slightly higher than those for LEU fuels.

The degree of variability in the recommended release parameters for noble gases, halogens, and alkali metals are not vastly different from those for PWRs and BWRs using LEU fuels. Nevertheless, some of the panel members were of the opinion that higher in-vessel releases (and faster rate of releases) are expected for MOX fuels as compared with LEU fuels. Note that some of the identified uncertainties are not specific to MOX fuels and are equally applicable to LEU and high burnup fuels (i.e., lack of data for effects of revaporization and air ingression, which impact late in-vessel releases). The largest quantitative differences are noted for tellurium, where the effect of zirconium-niobium (M5) cladding for MOX fuels is the most notable reasons listed by the panel members for the higher release fractions. In addition, there is a general opinion that there will be a higher concentration of the reactive forms of tellurium in the release, causing a significant fraction of the release to be deposited on reactor coolant system structures, with a higher propensity for late revaporization (late in-vessel phase).

The uncertainties in the release of semi-volatile and refractory species are stated as being very large, hampering the ability of the panel members to recommend estimates of quantitative release fractions with any confidence.

Finally, there is a general expectation that the physical and chemical forms of the revised source terms as defined in NUREG-1465 are applicable to high burnup and MOX fuels.

4.3 Panel Recommendations on Research Needs to Confirm Changes to the Revised Source Term

The limitations of the analysis and the available data make additional research to confirm the panel's estimates important. This section provides a summary of the specific recommendation by the panel members for research needs to confirm changes to the revised source term, as developed in the present report.

The following specific research recommendations are extracted from the letters reproduced in Appendix B, and ranked in accordance with the following prioritization scheme:

(NOTE TO PANEL MEMBERS: PLEASE CHANGE THIS PRIORITIZATION SCHEME AS NEEDED. THERE MAY BE A NEED TO ITERATE ON THIS SCHEME BEFORE WE SETTLE ON A FINAL PRIORITIZED LIST)

- High Priority – Research is required to develop confidence in the estimated releases to containment.

- Medium Priority – Research, if conducted, should help reduce some of the uncertainties in releases to containment.

- Low Priority – Research that is to help develop a greater understanding of core degradation and fission product releases over the long-term, but not essential for use of the revised source term in licensing applications.

NOTE TO PANEL MEMBERS: PLEASE PROVIDE YOUR RECOMMENDATIONS FOR RANKING. PLEASE REFER TO ITEM NUMBER, WHEN MAKING RECOMMENDATIONS FOR RANKING. ALSO, PLEASE ADD TO THE LIST OR EXPAND ON THE SPECIFIC RECOMMENDATIONS, AS APPROPRIATE (THIS IS ESPECIALLY IMPORTANT FOR THOSE PANEL MEMBERS THAT HAVE NOT YET SENT US THEIR LETTERS)

High Priority

1. Validation of accident analysis tools (i.e., MELCOR, VICTORIA) by comparison of predictions with results of major source term tests (e.g., PHEBUS-FP, VEGA and VERCORS with fuel of various burnup levels and MOX fuel) is needed. These comparisons that will lead to improvements and ultimately validation of the

computer codes are essential steps before analyses of significant accident sequences using accident analysis tools.

2. Experimental investigation of in-vessel core degradation following vessel failure is important in verifying the impact of air-ingression on producing radically different source term (e.g., verification of the Canadian tests showing a nearly complete release of radioactive ruthenium in air). This is also an important issue for the assessment of spent fuel pool accidents, fuel transportation and dry cask storage of fuel.

3. Acquire any available database on fission product release rates from high burnup and MOX fuels, in order help the panel to update the panel recommendations included herein. This data will also help to parameterize the available fission product release models in the systems codes used to analyze reactor accidents (see item 1 above).

4. Fuel burnup is expected to have an impact on the fuel melting point and fuel liquefaction process. The interaction of melting cladding with the fuel can be affected by the development of a restructured 'rim' region and by the formation of a significant oxide layer on the inner surface of the cladding. Perhaps of more significance is the possibility that the degradation of high burnup fuel will involve 'fuel foaming' rather than fuel candling as observed with fuel at lower burnup levels. This could change the core degradation process and consequently the release of fission products from the degrading fuel in qualitative ways that cannot be appreciated by simply extrapolating the results of tests with lower burnup fuel. Therefore, experimental investigation of fuel at high burnup, and with cladding material that include tin and niobium (Zirlo) or zirconium-niobium-tin alloys (M5), are essential in confirming the radiological release characteristics (e.g., effects of tin in M5 cladding on tellurium release) of fuels at high burnup and with new cladding material.

5. Revaporization is an important element of the revised source term as documented in NUREG-1465 and the present report. The actual magnitude of the revaporization component depends on the vapor pressures of the deposited radionuclides and these vapor pressures depend on the chemical form of the deposited radionuclide. Unfortunately, there is a limited understanding of the chemical forms of the deposited radionuclides. Consequently, empirical data are required on the vaporization of deposited radionuclides for comparison with predictions of models of the revaporization process.

Medium Priority

6. Assess the oxygen potential of MOX fuel in order to develop a better understanding of the chemical forms and volatility of various released constituents.

7. Perform separate-effects tests to resolve the issues of tellurium release; in particular, the potential dependence of tellurium release on the interaction of tellurium with zirconium in the cladding needs to be assessed experimentally.

8. Tests of core degradation with MOX fuel in order to assess damage progression behavior. These tests need to be done with fuel rod bundles to investigate the fuel liquefaction, fuel relocation and fission product releases during the degradation process.

9. Applicability of MOX data and models needs to be established. In particular, the differences, if any, in the fuel degradation behavior between the MOX fuel that has been prepared with reactor-grade plutonium dioxide to the fuel that has been prepared from weapons-grade plutonium dioxide (of primary interest in the United States), need to be assessed, analytically and/or experimentally.

10. Assess the accident sequence-specific aspects of release by considering thermal-hydraulic conditions other than low-pressure LOCA events in developing revised source terms for licensing applications.

11. Analyses of risk significant accident sequences in reactors fueled with MOX that follow a systematic code assessment process as recommended earlier for high burnup fuels.

Low Priority

12. The restructuring of high burnup fuel in the peripheral regions was identified as an issue that could lead to higher gap inventories of volatile radionuclides and higher release fractions of noble gases in the 'gap release' phase of an accident. Information on the gap inventories of fission gases (Xe and Kr), certainly, and possibly volatile radionuclides (Cs, I, and Te) and maybe even moderately volatile radionuclides (Ba, Sr, Sb, Mo) might also be derived from work being done in the Halden program on fuels taken to high burnup levels. If loss of coolant accident (LOCA) tests of high burnup fuel are to be done, some effort should be made in these tests to validate the predictions of the expert panel with respect to the gap releases including the prediction that the gap releases of cesium and iodine would not be affected significantly by high burnup Tests with longer rods will also provide information on the longer term fraction of the gap release fraction.

13. The understanding of fission product release during core debris interactions with concrete is fairly complete. Refinement of this understanding and the predictions of the release are not crucial for the revisions of the accident source term. This is true if for no other reason because the releases at this late stage of an accident are seldom used. Nevertheless, some of the known shortcomings of the codes used to predict the releases of fission products during core debris interactions with concrete should be addressed.

14. The only BWR core damage progression test to date was DF-4. Yet one-third of the U.S. fleet is BWR. Generally, it is thought that BWR core damage progression phenomena can differ from PWRs (e.g., candling of melt vs. forming a crucible, different control rod materials, larger amount of Zr which could make the accident environment more reducing, lower power density). While these differences may not make BWR releases substantially different from PWR releases (NUREG-1465 BWR releases do not differ from PWR releases other than a small difference for iodine and cesium), to the extent that additional Phebus tests are being contemplated, it makes sense to consider having one of these tests be for BWR fuel.

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APPENDIX A:

SOURCE TERM PANEL MEMBERS

A.1 Panel Members

Dr. Bernard Clément	Institut de Protection et de Sûrete Nucléaire (IPSN)
Dr. James Gieske	Consultant
Dr. David E. Leaver	Polestar Applied Technology, Inc.
Dr. Thomas S. Kress	Consultant
Dr. Dana A. Power	Sandia National Laboratories (SNL)

A.2 Facilitator

Dr. Brent Boyack	Los Alamos National Laboratory (LANL)

A.3 Vitae of Panel Members

Bernard Clément

TO BE PROVIDED

James Gieske

TO BE PROVIDED

David Leaver

TO BE PROVIDED

Thomas S. Kress

Thomas S. Kress is past chairman of USNRC's Advisory Committee on Reactor Safeguards (ACRS) and is currently serving his third term on that committee. He acquired a BS and MS in Mechanical Engineering and a PhD 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 design and safety aspects of LWRs, LMFBRs, Molten-Salt Reactor, 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 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, more recently, helped develop an IAEA TECDOC on design basis source terms for future LWRs.

Dana A. Powers

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

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

APPENDIX B:

PANEL RECOMMENDATIONS ON RESEARCH NEEDS TO CONFIRM CHANGES TO THE REVISED ACCIDENT SOURCE TERM

Mr. Jason Schaperow February 27, 2002
U.S. Nuclear Regulatory Commission
Mail Stop T-10-K-8
Washington, DC 20555

Re: **Research Needs to Confirm Revisions to the Reactor Accident Source Term**

Dear Mr. Schaperow:

Over the last several weeks our panel has been developing revisions to the reactor accident source term described in NUREG-1465. This effort to develop revisions to the reactor accident source term were prompted by interest in having source terms applicable to conventional reactor fuel taken to high burnups (55 to 75 GWd/t) and to mixed-oxide fuel (MOX) made with weapons-grade plutonium dioxide. In formulating the revisions, however, attention was also given to the changes in our understanding that have come about because of major experimental investigations of fission product behavior under reactor accident conditions such as the PHEBUS-FP program, the VERCORS tests, and VEGA tests. The assessments were done, however, without the benefit of accident sequence analyses using accident analysis models validated by comparison to pertinent tests involving fuel taken to higher burnups or MOX.

Members of the panel developing the revisions to the NUREG-1465 source term have attempted, then, to mentally integrate the results of specific, recent, tests to predict source terms during accidents at nuclear power plants. They have extrapolated phenomenology of core degradation known from studies of fuel taken to modest levels of burnup to predict the source terms from fuel burned to levels in excess of about 60 GWd/t. The panel members have also extrapolated the behaviors of conventional fuels with conventional Zircaloy cladding to estimate the behavior of mixed oxide fuel with zirconium-niobium (M5) cladding.

The limitations of the analysis and databases available to the expert panel make research to confirm the panel's estimates important. I outline below what I believe to be important confirmatory research to undertake to substantiate the experts' recommendations for changes to the reactor accident source term. I present these suggestions for research in approximate priority order beginning with the highest priority tasks. In developing the priorities, I have been aware that the more used portions of the accident source term are the gap releases and the invessel releases. I have also recognized that nearly all reactors now plan to take fuel to very high levels of burnup (at least 60 GWd/t and perhaps, someday, as high as 75 GWd/t), whereas only two reactors now plan to use mixed oxide fuel. A different approach to the prioritization could alter the order of the list. Especially the priority of work on mixed oxide fuel issues could be changed if different criteria had been used.

Task 1: Validate accident analysis tools by comparison of predictions with results of major source term tests.

A new generation of accident analysis tools, notably the MELCOR code[1] and to a lesser extent the VICTORIA code[2], are available for predicting reactor accident source terms. These tools have been developed based on research that led to the NUREG-1465 source term and research that has been done since his source term was published. These tools have not been validated, however, by comparison of their predictions with the results of major tests in the PHEBUS-FP program including the recent FPT-2 test involving reduced coolant flow and the FPT-4 test involving radionuclide release from debris expected to form in the later stages of core degradation. Comparisons of code predictions to the results of VERCORS tests with fuel of various burnups and mixed oxide fuel fabricated with reactor-grade plutonium dioxide need also to be done. These comparisons that will lead to improvements and ultimately validation of the computer codes are essential steps before the next high priority, confirmatory research task - analyses of significant accident sequences. Specific code comparison that are needed include comparisons of code predictions to the results of the FPT-1, FPT-4 and FPT-2 integral tests from the PHEBUS-FP program. More detailed comparisons to separate effects tests from the VEGA test program and VERCORS test program may be needed to properly parameterize models of radionuclide release from the degrading fuel.

The test results now available for comparison with code predictions have a distinct bias toward pressurized water reactor accident conditions. This has been in the past a significant concern since so few data directly applicable to accidents in boiling water reactors are available. This concern is being ameliorated in recent years as fuel and fuel configurations used in boiling water reactors have evolved to become more like the fuel and fuel configurations found in pressurized water reactors. An approximation that has been made in developing the recommendations that follow is that boiling water reactor specific data are not needed. That is, phenomenological tests recommended below will be suitable for validating codes for analysis of accidents in either pressurized water reactors or boiling water reactors. Explicit validation of this assumption is probably needed at some point.

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[1] R.O Gauntt, *et al.*, **MELCOR Computer Code Manuals**, NUREG/CR-6119, Volumes 1 and 2, SAND 2000-2417/1,2, Sandia National Laboratories, Albuquerque, NM, December 2000.

[2] N.E. Bixler, **VICTORIA 2.0: A Mechanistic Model for Radionuclide Behavior in a Nuclear Reactor Coolant System Under Severe Accident Conditions**, NUREG/CR-6131, SAND93-2301, Sandia National Laboratories, Albuquerque, NM, December 1998; T.J. Heames, et al., **VICTORIA: A Mechanistic Model of Radionuclide Behavior in the Reactor Coolant System Under Severe Accident Conditions**, NUREG/CR-5545, SAND90-0756, Revision 1, Sandia National Laboratories, Albuquerque, NM, December 1992.

Task 2: Reanalyze risk significant accident sequences using the validated accident analysis tools.

The original NUREG-1465 accident source term was developed based on examination of a variety of accident analyses most of which were done with the old Source Term Code Package. Similar sets of analyses obtained using improved, validated analytic tools need to be the basis of the revisions to the accident source term. One really must use accident analyses to define the source terms rather than the results of tests. Releases of radionuclides depend on time and temperature as well as accident phenomena. The times and temperatures of reactor accidents are not usually mimicked in tests to the accuracy needed to estimate source terms. Crucial issues that need to be considered in these analyses include:

- fraction of the core that remains within the core region following rupture of the reactor pressure vessel; this will have significant implications on the nature of the source term following vessel rupture.
- evidence from that the tests that fuel relocation, gas flow and gas composition are as important as fuel temperature in determining the rates of radionuclide releases.
- maximum temperatures reached by fuel prior to liquefaction and relocation which will have important ramifications on the release of the more refractory radionuclides.
- development of regions and periods of steam starvation in the core region which will affect the vaporization of alkaline earths (Ba, Sr) as well as refractory radionuclides such as Ce, Pu, and La.
- releases of tellurium (Te), molybdenum (Mo) and ruthenium (Ru).
- duration of core debris retention in the reactor pressure vessel.

An important outcome of these analyses will be determination of the need to draw additional distinctions within classes of radionuclides such as the distinctions drawn between Mo and Ru in the noble metal group and between Ce and Pu in actinide group. Predictions of prolonged retention of core debris within the reactor vessel might necessitate some changes in the timing of the accident source term and even the definition of new regimes of release.

Task 3: Experimental investigation of invessel core degradation following vessel rupture.

Modern accident analyses appear to predict with great consistency that a substantial fraction of the reactor core remains within the core region after reactor pressure vessel failure. The degradation of this residual fuel following penetration of the reactor vessel by core debris has been the subject of some speculation recently but little detailed

analysis. The degradation of this fuel is often known as the 'air ingression' issue[3] because it is likely that air will interact with this fuel following vessel failure. Air interactions with the residual fuel has the potential of producing a radically different source term based on results of tests in Canada showing nearly complete release of radioactive ruthenium in air[4]. But, we suspect that there is a competition between the rate of fuel liquefaction by interaction with molten cladding and fuel oxidation leading to extensive releases of ruthenium and tellurium. Competition arises because the reaction of air with the fuel cladding produces so much heat that it is possible that the fuel will melt and relocate from the core region before there is an opportunity for extensive fuel oxidation and releases of radioactive ruthenium and tellurium. This competition probably cannot be resolved based strictly on analysis. Experimental investigations will be needed. Experimental studies of this later phase of the core degradation process would not ordinarily rise so high in priority, but air interactions with fuel are also important to issues of spent fuel pool safety, fuel transportation and dry cask storage of fuel. These needs in combination with the significant potential change in the ex-vessel stage of the accident source term, experimental studies of core degradation in air are recommended with high priority.

Task 4: Acquire any available data base on fission product release rates from high burnup fuel.

The purpose of this task is to assemble data suitable for the parameterization of models of fission product release used in accident analysis codes to account for the effects of elevated burnup. To be useful, the data must be for fuel that has sufficiently high burnup that a significant 'rim' region of restructured fuel has developed. This typically means that the fuel pellet average burnup must exceed about 55 GWd/t and should preferably be in excess of 70 GWd/t. The data that are most useful are for semi-volatile and low volatile fission products. (Volatile fission products such as Cs and I are essentially completely released by fuel that is heated to the point of liquefaction and relocation. Details of the release rates for these volatile species are, then, less crucial to the accurate estimation of release fractions for the revised accident source term.) The best data will involve at least two temperature plateaux where substantial release occurs so that temperature dependencies can be determined.

Task 5: Experimental investigations of core degradation with high burnup fuel.

The profound, qualitative differences between the degradation of unburned uranium dioxide fuel and fuel that has been taken to burnups of 30-40 GWd/t have been

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[3] D.A. Powers, L.N. Kmetyk, and R.C. Schmidt, **A Review of the Technical Issues of Air Ingression During Severe Reactor Accidents**, NUREG/CR-6218, SAND94-0731, Sandia National Laboratories, Albuquerque, NM, September 1994.

[4] F.C. Iglesias, C.E.L. Hunt, F. Garisto, and D.S. Cox, "Measured Release Kinetics of Ruthenium from Uranium Oxides in Air", **Proceedings International Seminar On Fission Product Transport Processes During Reactor Accidents**, J.T. Rogers, editor, Hemisphere Publishing Corp., Washington, D.C., 1990.

remarkable. Even though burnup only modestly affects the melting point of the fuel, burnup very seriously affects the liquefaction of the fuel. Similar qualitative changes in the degradation of fuel that has experienced very high burnup are possible. Clearly, the interaction of melting clad with the fuel can be affected by the development of a restructured 'rim' region and by the formation of a significant oxide layer on the inner surface of the cladding. Perhaps of more significance is the possibility that the degradation of high burnup fuel will involve 'fuel foaming' rather than fuel candling as observed with fuel at lower burnups. Fuel foaming even if it is important only for transient periods could change the core degradation process and consequently the release of fission products from the degrading fuel in qualitative ways that cannot be appreciated by simply extrapolating the results of tests with lower burnup fuel.

The suitability of the fuel for tests of high burnup fuel degradation are similar to those mentioned above. That is, the fuel must have a well-developed 'rim' structure. Tests with fuel having an average burnup of about 70 GWd/t would be preferred. Another issue of importance is the choice of cladding. Until now most tests of core degradation have been done with Zircaloy clad fuel. It appears that many licensees are migrating toward cladding alloys involving both tin and niobium (Zirlo) or zirconium-niobium alloys (M5). Tests of core degradation with high burnup fuel should be done with a type of cladding that will get widespread use in the future.

Task 6: Analysis of accident sequences involving high burnup fuel.

Once reliable, validated accident analysis models are available, a range of representative accident sequences must be analyzed to develop a base of information from which accident source terms may be derived. The range of accident sequences must include both pressurized water reactors and boiling water reactors. The range must also include the range of patterns for loading fuels of various levels of burnup into the reactor core. An effort must be made to identify those things that qualitatively affect the progression of core damage and the releases of fission products to the containment.

Task 7: Revaporization Tests

A sage feature of the NUREG-1465 source term is the recognition that fission products deposited in the reactor coolant system my revolatilize later in the accident. The inclusion of this long-term, late source of radionuclides to the containment has been based on analyses. The actual magnitude of the revaporization source term will depend on the vapor pressures of the deposited radionuclides and these vapor pressures depend on the chemical form of the deposited radionuclide. Unfortunately, we have a limited understanding of the chemical forms of the deposited radionuclides. Experimental determination of the chemical forms of radionuclides on surfaces is quite difficult because concentrations can fall below the limits of common methods for determination of chemical form. Consequently, empirical data are required on the vaporization of deposited radionuclides for comparison with predictions of models of the revaporization process.

The tests that are needed involve heating deposits of radionuclides from prototypic surfaces in flowing atmospheres of the type to be expected in the reactor coolant system late in accident sequences. Such atmospheres could be quite oxidizing if the reactor coolant system has been penetrated by core debris and air circulates by natural convection through the reactor coolant system. Temperatures that need to be investigated are modest and certainly do not exceed the melting point of steel or even the temperatures at which heavy sections of steel creep rapidly.

Fortunately, it is probably not necessary to develop new test plans for the study of revaporization. Some revaporization tests are planned for the PHEBUS-FP program. Assuming that these tests do not fall victim to limitations of the budget, the tests should involve the revaporization of radionuclides from rather prototypic mixtures of deposited materials. It is, however, still important that once these test results come available that they are compared to predictions of models and appropriate adjustments to the models be made. Then, the models have to be used to predict the release fractions from revaporization in risk significant accident sequences.

Task 8: Separate-effects tests to resolve the issues of tellurium release.

The release of tellurium from degrading reactor fuel remains a mystery. Based on its physical and transport properties, tellurium should be as volatile as cesium and iodine from overheated reactor fuel. It is found, however, in some tests that tellurium release is greatly inhibited - as though it were interacting with the clad and would not release until the clad has been extensively oxidized. In other tests, it appears that tellurium is bound within the fuel matrix in some form that exhibits limited volatility. In the recent PHEBUS-FP tests, there is evidence of rather extensive tellurium release, but this release may have been delayed either until sufficient cladding oxidation has occurred or until sufficient fuel oxidation or degradation has occurred.

It appears some experimental resolution of this state of limited understanding of tellurium release is required. The need for this resolution may be even greater if there is an evolution within the nuclear industry away from cladding with high tin content to cladding made with zirconium alloyed with elements that interact less strongly with tellurium than tin.

Task 9: Acquire any test data on the release of fission products from MOX fuel.

The purpose of this task is to provide the data needed to parameterize the models of fission product release to account for the unique feature of mixed oxide fuel (MOX). The requirements for useful data are similar to those discussed for a similar task in connection with high burnup fuel (Task 4, above). The best data are for the modestly volatile and low volatile species such as Ba, Mo, Ce, etc. The tests should involve at least two temperature plateaux where significant (measurable) releases occur so that temperature dependencies

can be determined. At least some useful data are available from the VERCORS program[5] underway in France, though it may be necessary to obtain additional data to properly parameterize the models of release especially for the releases of actinides such as Pu, Cm, and Np. Releases of these actinides have been the subject of some public concern over the plan to use MOX in commercial nuclear power plants[6].

Task 10: Tests of core degradation with MOX fuel.

There are not test data now on the degradation of MOX fuel under accident conditions. Because the interaction of the cladding with MOX could be quite different than the interaction of cladding with conventional fuels, it is possible that degradation of MOX could be quite different than the degradation of conventional reactor fuels. Indeed, MOX may be more susceptible to fuel foaming types of degradation than candling degradation because of the high gas content of localized, plutonium-rich regions of the fuel. Such qualitative changes to the degradation process will affect the release fractions of fission products in a systematic way. Tests are needed, then, of the degradation of MOX. Such tests need to be done with fuel rod bundles to investigate the liquefaction and relocation of fuel. Fission product releases during the degradation process need to be monitored.

Task 11: Applicability of MOX data and models.

Data on fission product release from MOX fuel and degradation of MOX fuel will come, almost assuredly, from MOX that has been prepared with reactor-grade plutonium dioxide rather than weapons-grade plutonium dioxide that will be of primary interest in the United States. There may be other differences. For example, some have suggested that the size distribution of plutonium dioxide rich particles within the fuel to be used in the US will be different than the size distribution of particles in fuel prepared in Europe. Analyses or tests must be done, then, to confirm that data obtained with one type of MOX fuel are adequate, perhaps with some corrections, to adequately understand the behavior (degradation and radionuclide release) of the type of fuel of primary interest. Whether this task can be done analytically or requires prototypic testing really cannot be adjudicated until there are more data on some types of MOX fuel and some better phenomenological understanding of the things that most affect MOX behavior under accident conditions.

Task 12: Analyses of risk significant accident sequences in reactors fueled with MOX.

Task 12 is much like task 6 for high burnup fuel and task 2 for general changes to the source term. Once reliable, validated models of core degradation involving MOX and

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[5] P.P. Malgouyres, M.P. Ferroud-Plattet, G. Ducros, C. Poletiko, M. Tourasse, and D. Boulaud, "Influence of MOX Fuel in Fission Product Release Up to Meltdown Conditions", **Communications at the NURETH 9 ANS Meeting**.

[6] E.S. Lyman, **Public Health Consequences of Substituting Mixed-Oxide for Uranium Fuel In Light-Water Reactors**, Nuclear Control Institute, January 21, 1999.

fission product release from MOX are available, these codes must be used to analyze representative accident sequences for the plants that will use the MOX. This task is simplified by the fact that today only Westinghouse pressurized nuclear power plants in ice condenser containments will use MOX. The task is complicated, however, by the need to consider a range of patterns for the loading of MOX into the core, perhaps a range of MOX concentrations in fuel, and ranges of fuel burnup. The analyses of the representative accident sequences will provide the information in which a meaningful accident source term can be derived.

Task 13: LOCA tests with high burnup fuel.

The expert panel argued that restructuring of high burnup fuel in the peripheral regions should lead to higher gap inventories of volatile radionuclides and higher release fractions of noble gases in the 'gap release' phase of an accident. Information on the gap inventories of fission gases (Xe and Kr), certainly, and possibly volatile radionuclides (Cs, I, and Te) and maybe even moderately volatile radionuclides (Ba, Sr, Sb, Mo) might also be derived from work being done in the Halden program on fuels taken to high burnup levels. But, inventories are not easily translated into gap release fractions. **If** loss of coolant accident (LOCA) tests of high burnup fuel are to be done, some effort should be made in these tests to validate the predictions of the expert panel with respect to the gap releases including the prediction that the gap releases of cesium and iodine would not be affected significantly by high burnup. Such validation is complicated by the loss of gap inventory in the refabrication of rods for the tests. This can only be addressed by attempting to measure or estimate the loss of inventory. It may not matter whether tests are done with single rods or with multiple rods. More important will be the length of the rod. The gap release is composed of a prompt inventory venting and a longer term gap diffusion fraction. Releases from short rods will be dominated by the inventory venting. Tests with longer rods will also provide information on the longer term fraction of the gap release fraction.

Task 14: Upgrade models of fission product release during core debris interactions with concrete.

The understanding of fission product release during core debris interactions with concrete is fairly complete. Refinement of this understanding and the predictions of the release are not crucial for the revisions of the accident source term. This is true if for no other reason because the releases at this late stage of an accident are seldom used. Nevertheless, there are some nagging anachronisms in current codes used to predict these releases of fission products during core debris interactions with concrete. For example, it is known that the existing models underpredict the releases of ruthenium observed in tests because of inaccurate partitioning of ruthenium among the phases of core debris and omission of an important vapor species. The codes may also overpredict the releases of alkaline earths (Ba and Sr) because of improper activity coefficients. If resources are available some limited effort to correct these code deficiencies should be undertaken.

Sincerely yours,

Dana A. Powers
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cc:

R. Gauntt, SNL
R.Y. Lee, NRC
M. Khatib-Rahbar, ERI
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March 13, 2001

Mr. Jason Schaperow
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Mail Stop T-10-K-8
Washington, D.C. 20555

Dear Jason,

In the third meeting of the panel addressing the NUREG-1465 update for high burnup and mixed oxide (MOX) fuels, it was requested that panel members submit a letter discussing research needs. This letter is to provide my thoughts on this matter. In preparing this letter, I have received input from or discussed this subject with Duke Energy, the Electric Power Research Institute (EPRI), and Polestar people with experience in source term phenomena.

To address research needs, two categories of data have been considered:

(1) Data which could be made available in the short-term and used for the ongoing NUREG-1465 update for high burnup and MOX fuels
(2) Longer-term research which would be confirmatory, or which may be of interest to stakeholders

Observations on Short-Term Research

In the first category, there are several issues which have been discussed by the panel and which warrant discussion here:

French MOX fuel fission product release data From information presented by the French representatives, it is evident that there are significant data available on MOX accident releases. These data include that from VERCORS RT 2 (can be compared to RT 1) and VERCORS RT 7 (can be compared to HT 1). This would very likely be the most cost-effective, and only short-term, way for NRC to obtain MOX fuel accident release data. The panel should have the benefit of reviewing this data before concluding its work on the NUREG-1465 update if at all possible. Tables for RT 1, RT 2, and RT 7 similar to the HT 1 table entitled, "VERCORS HT1: FP release balance," which was in the French presentation would be very useful.

Oxygen Potential of MOX Fuel Figure 2. of reference [1] shows oxygen potential vs. temperature for LEU fuel materials. While significant differences are not expected, it would be useful to make some estimates of how the oxygen potential for MOX fuel (i.e., PuO_2 rich agglomerates in a UO_2 matrix) compares with this figure. As fuel damage progresses, oxygen potential of the damaged fuel is determined by the H_2/H_2O ratio in the gas space in the core region. Assuming chemical equilibrium exists at the high core region temperatures, oxygen potentials required for formation of oxides of various core

materials can be used to assess fission product chemical form. The chemical form of a material (metal vs. oxide) affects its volatility.

Phebus FPT-2 data Phebus FPT-2 was performed in October, 2000. While data analysis takes significant time, perhaps a quick look report of FPT-2 release data could be made available to the panel in the next few months. One interesting qualitative observation from FPT-2 is the fact that the Te release was quite low compared to FPT-1.

Weapons grade (WG) MOX vs. reactor grade (RG) MOX This matter was raised as a possible concern regarding the applicability of French MOX source term data, which is for RG MOX, to the U.S. Material Disposition program which will use WG MOX. The following is noted regarding RG MOX vs. WG MOX:

- Per reference [2], the MOX fuel fabrication process (master blend mix) will be adjusted for WG MOX fuel to produce plutonium-rich particles with approximately the same fissile content as RG MOX plutonium-rich particles, consistent with the European experience base. Thus the fission density and the fission product inventory will be the same in WG and RG MOX.

- The WG MOX fuel composition and neutronic performance tends to be in between that of RG MOX and low enriched uranium (LEU) fuel. This is illustrated in Figures 7.4 to 7.8 in reference [2].

- The ongoing Advanced Test Reactor (ATR) WG MOX fuel tests have progressed beyond 30,000 MWD/t with no indication of differences between WG MOX fuel and RG MOX fuel. These tests have included destructive hot cell post-irradiation examination (PIE) at intermediate burnups. The tests are to be continued to a burnup of 50,000 MWD/t, including additional hot cell PIE.

- The DCS Fuel Qualification Plan includes a WG MOX fuel lead assembly program – prototypical WG MOX fuel assemblies in one of the McGuire or Catawba reactors. The plan calls for irradiation of at least one assembly for three cycles (burnups in excess of 50,000 MWD/t), and hot cell PIE approximately one year following the third cycle. The hot cell PIE is intended to be confirmatory in nature – i.e., the NRC would issue a license for batch-scale irradiation of MOX fuel prior to the PIE.

Based on the above, the matter of WG vs. RG MOX is likely to be a second order effect for source term and need not be considered for additional research in the short-term. The ATR and MOX fuel lead assembly programs should be monitored to confirm that the data do not indicate significant differences between WG and RG MOX fuel.

Observations on Longer-Term Research

Effect of M5 Clad Material The MOX fuel assembly design will employ M5 cladding. M5 is an advanced bi-metallic (zirconium-niobium, no tin) cladding material. There was

speculation at the last meeting regarding the effects of M5 clad on source term, the main possible effect being on Te release.

There are several points relevant to this discussion:

- With tin present in the clad, for low clad oxidation the Te can be sequestered as SnTe in the unoxidized clad, thus limiting the Te release. However, even with tin, high local oxidation (such as the 85% active clad oxidation in FPT-1) can drive much of the SnTe out from the oxidized region.
- This clad sequestering effect will not be present without tin. In this case there are three possibilities: (1) the Te is in a reactive form (elemental Te or H_2Te) which will significantly increase the Te deposition velocity and thus decrease Te release, (2) some of the Te reacts with other materials such as silver (for PWRs with Ag-In-Cd control rods) which results in sequestering, or (3) some of the Te reacts with silver aerosol to form Ag_2Te aerosol which behaves like SnTe
- A final point is that use of M5 clad (and thus the absence of tin) is not exclusively a MOX fuel source term issue since M5 is being used in operating plants in the U.S. (LEU fuel) and Europe today.

Thus, use of M5 clad would not be expected to increase Te release, and in fact may decrease it. This, together with the fact that the M5 issue exists for LEU fuel, suggests that if research is performed on the effect of M5, it be done in the context of LEU fuel. Such research need not impact the ongoing MOX source term effort.

Non-LOCA events The panel's scope is limited to LOCA release. The gap release fraction is a relatively small part of the LOCA source term, most of the release being fuel release. Thus, changes in the gap release will not have a major effect on the total release for the LOCA. This is not the case for non-LOCA events (such as FHA, locked pump rotor, MSLB, SGTR, rod ejection) where most if not all of the release is the gap. Furthermore, these non-LOCA events involve different thermal conditions which can affect release: FHA involves a cool rod; locked pump rotor, MSLB, and SGTR involve a slowly heated rod; and rod ejection involves a rapidly heated rod. Thus, at some point in the future the NRC should consider updates for the non-LOCA releases.

BWR fuel release The only BWR core damage progression test to date was DF-4. Yet one-third of the U.S. fleet is BWR. Generally, it is thought that BWR core damage progression phenomena can differ from PWRs (e.g., candling of melt vs. forming a crucible, different control rod materials, larger amount of Zr which could make the accident environment more reducing, lower power density). While these differences may not make BWR releases substantially different from PWR releases (NUREG-1465 BWR releases do not differ from PWR releases other than a small difference for iodine and cesium), to the extent that additional Phebus tests are being contemplated, it makes sense to consider having one of these tests be for BWR fuel.

MOX core damage progression There was discussion at the last meeting on the possible need for a MOX fuel core damage progression test. Such a test is considered low priority based on the following:

- The MOX fuel assembly design is a state-of-the-art fuel assembly design by Framatome. A very similar design (without M5) was used for more than a decade in the McGuire and Catawba units, and is currently being used at Sequoyah. The same design (except for fuel pellet material) will be deployed at the North Anna units in about one year. Other than fuel pellet material, there is nothing special about the fuel assembly design, relative to LEU PWR fuel.
- MOX fuel has only small differences in thermal properties compared to LEU fuel (slightly lower decay heat, slightly lower thermal conductivity, slightly higher specific heat below ~2300 K, slightly lower specific heat above ~2300 K, slightly higher fuel temperature). These differences would not be expected to cause significant differences in severe accident behavior of MOX fuel compared to LEU fuel.
- The only aspect of MOX fuel which could impact severe accident progression is the fuel pellet microstructure (i.e., the occurrence of Pu fissions in PuO_2 rich agglomerates in the UO_2 matrix). However, this impact is expected to be minor since pellet microstructure will have little or no effect once fuel melting begins. In addition, any potential effect of MOX fuel is mitigated by the fact that the core is predominantly LEU fuel.

In any event, the VERCORS data, oxygen potential information, and the PIE from ATR WG MOX fuel irradiations noted above in the short-term research discussion should be used in deciding the priority of any MOX tests.

I appreciate the opportunity to provide this input for NRC consideration. Please contact me with any questions or comments.

Very truly yours,

David E. Leaver

Cc: Steve Nesbit, Duke Energy
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