FFRD Impact on the Containment Source Term

June 15 2021

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Abstract

The impact of fragmentation on the design-basis Containment Source Terms was evaluated. The considered source terms are that specified in RG1.183 Tables 1, 2, and 4 and a subsequent update to this source term for high-burnup fuel. This evaluation built upon considerable previous work on Severe Accidents, Source Term, and fuel fragmentation, relocation, and dispersal (FFRD) behavior during a loss-of-coolant accident, and on insights from the recent accident tolerant fuel (ATF) phenomena identification and ranking table (PIRT) exercise. Previous NRC documents regarding the impact of fragmentation focused on the recovered large break Loss-of-Coolant Accidents (LOCA)s involving functioning Emergency Core Cooling Systems (ECCS). Similar documents were not created to evaluate the impact of fragmentation on the Containment Source Term because of the expectation of a minimal impact. This document provides some of the reasoning behind this expectation and provides some simple scoping analyses supporting this expectation. The contribution of FFRD to the Containment Source Term is not significant, even when using unrealistically conservative assumptions regarding fuel fragment release magnitudes, transport behavior, concentration, and fragment heat up. When using more realistic assumptions for these effects, the source term with fragmentation is bounded by an equivalently developed Current Source Term without these effects. Two aspects of release in the Containment Source Term can be enhanced by fragmentation; enhanced early noble gas (NG) release upon clad burst, and earlier relocation of some of the decay heat carried by fragments to the lower head and the water remaining there. Both aspects affect the timing of radionuclide release to the containment. The magnitude of NG release from fragments cannot significantly increase the Containment Source Term NG release for high burnup because existing analyses already predict near-total NG release. Some NGs can release earlier because of high burnup. Given the expected fragment fraction and dispersal, it is considered unlikely that fuel collects in an unfavorable configuration, heats up, and releases large quantities of volatile radionuclides that would not otherwise have been released. The RG1.183 endorsed codes, ORIGEN, RADTRAD, and MELCOR, were briefly evaluated considering FFRD behavior. The codes are considered adequate for conservatively capturing FFRD behavior. Areas where calculations could be modified to better represent FFRD behavior were identified.

Introduction and Background

This document describes the expected impact of fuel fragmentation, relocation, and dispersal (FFRD) on the Light Water Reactor (LWR) design-basis (DB) Containment Source Term described in Regulatory Guide (RG)1.183 Tables 1, 2, and 4 and subsequent analyses to update this source term for high-burnup fuel. The Containment Source Term is used for demonstrating that plants remain within certain parts of their licensing bases. The previously developed high-burnup Containment Source Term did not include fragmentation effects.

The potential for FFRD at higher burnup has been a concern for ensuring that emergency core cooling systems (ECCS) remain within their acceptance criteria, as defined in 10 CFR 50.46. Fuel fragmentation has been extensively studied both by the Nuclear Regulatory Commission (NRC) and internationally. Analyses of ECCS acceptance involve the assumption of a functioning ECCS. Although rods may burst, the maximum temperature, and amount of radionuclide release, is limited. In such scenarios, a small fraction of the fuel released as fragments can potentially significantly increase radiological releases to the containment and can potentially affect ECCS performing their safety function of keeping the core cool.

The scenarios considered in the development of the Containment Source Term – which is used to demonstrate compliance with requirements in 10 CFR 50.34(a)(1)(ii)(D), 10 CFR 50.67, and 10 CFR 100.11) – typically do not involve functioning ECCS, unlike the ECCS acceptance criteria scenario. The Containment Source Term scenarios involve far greater radiological releases from the fuel. FFRD has not received comparable attention in The Containment Source term because the relative impact of expected fragmentation is limited in comparison to releases that result from a substantial meltdown of the core.

This report describes the containment source and approach to its development, provides a high-level status of the understanding of fragmentation, and assesses the potential impact of FFRD. It also includes some bounding scoping analyses of the contributing effects. Based on these analyses and observations, a conclusion is drawn about the expected impact of FFRD on the RG1.183 Containment Source Term and on the need for potential recommendations for future modeling of FFRD for Containment Source Term analyses.

Source Term Definitions:

"Containment Source Term" is used to satisfy 10 CFR 100.11 siting criteria. The source term for siting involves "substantial meltdown of the core" with "release of appreciable quantities of fission products". Regulatory Guide (RG) 1.183 Tables 1 and 2 provide gap release and early in-vessel phase radionuclide release fractions for BWRs and PWRs, respectively. RG 1.183 Table 4 provides the phase release timing for both boiling water reactors (BWRs) and pressurized water reactors (PWRs). ECCS is not assumed to be functional in the development of the source term. Although this source term is often referred to as the "LOCA (Loss-Of-Cooling Accident) Source Term" , it is referred to as the Containment Source Term in this report to distinguish it from the source term from the recovered design-basis LOCA 10CFR50.46 analysis and should not be confused with it.

The section on siting 10CFR100.11 is generally applicable and uses the same terminology to describe the source term: "Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products". This is the same text as in the 10 CFR 50.67, "Accident Source Term," definition for the alternative source term. The 10 CFR 50.67 source term only applies to plants with licenses issued prior to January 10, 1997.

The DB LOCA (10 CFR 50.46), "Recovered 10 CFR 50.46 DB LOCA" primarily consists of a thermal hydraulic calculation to ensure that Large Break LOCAs (or other LOCAs) does not result in widespread fuel damage that could inhibit core coolability. This scenario involves minimal radionuclide releases. Some rods burst releasing gap inventory (and fragments) although clad temperatures limited to 2200 °F (~1204 °C, ~1478 K). ECCS is assumed to be operable and to rapidly recover cooling to the core. Fragmentation can potentially increase the source term from these events. The 10 CFR 50.46 rule governs the acceptance criteria for emergency core cooling systems for light-water nuclear power reactors. This rule places limits on clad temperature, clad oxidation, and hydrogen generation. It requires the maintenance of a coolable geometry and assurances of long term coolability.

In addition to the Containment Source Term described above, RG 1.183 includes a source term for Non-LOCA design basis accidents. This source term is used for other accident scenarios that do not have as extensive releases as the Containment Source Term. This source term is provided in RG 1.183 Table 3. Accidents scenarios covered by this source term include Reactivity Insertion Accidents RIA, Locked rotor (if releases are expected), and the Fuel Handling Accident (FHA).

Objectives and Scope

The primary objective of this report is to characterize the impact of FFRD on a high burnup Containment Source Term used to satisfy 10 CFR 100.11 siting criteria for large light-water reactors (LWRs) with peak rod-average burnups up to 68 GWd/t. This assessment is intended to apply specifically to the high-burnup Containment Source Terms previously developed to update Tables 1, 2, and 4 in RG1.183 Rev. 1 to account for both high burnup fuel and for modeling advances [Powers, 2011]. It is also intended to be generally applicable for future high burnup source terms up to peak rod-average burnups up to 80 GWd/t developed in a similar manner.[1]

This report does not evaluate the impact of fragmentation on 10 CFR 50.46 design-basis LOCA analyses. It mentions previous 10 CFR 50.46 design-basis LOCA work to the extent needed to describe the differences in assumptions and release magnitudes. This report also does not cover the impact of fragmentation on "non-LOCA" events (RG. 1.183 Table 3)[2].

The Containment Source Term

To evaluate the impact of fragmentation on the Containment Source Term, an understanding of the regulatory basis, the technical basis, and the approach to developing this source term is needed. This section describes the regulatory basis, the technical basis, the current status, and the approach to Containment Source Term.

[1] See memorandum from Michael Case to Joseph Donoghue and Michael Franovich, entitled, "Applicability of Source Term for Accident Tolerant Fuel, High Burn up and Extended Enrichment." (ADAMS Accession Number ML20126G376)

[2] The table numbers are likely to change in future revisions for Source Term Guidance

Regulatory Basis

Use of regulatory "*source terms"* in design-basis accident (DBA) assessments is deeply embedded in the regulatory policy and practices of the NRC, even as the licensing process has evolved over the past 60 years. The *source term* refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release. It is based upon the concept of defense-in-depth in which power plant design, operation, siting, and emergency planning comprise independent layers of nuclear safety. This approach encourages nuclear plant designers to incorporate several lines of defense in order to maintain the effectiveness of physical barriers between radiation sources and materials from workers, members of the public and environment in operational states and, for some barriers, in accident conditions. It centers on the concept of DBAs, assessment of which aims to determine the effectiveness of each line of defense. The DBAs establish and confirm the design basis of the nuclear facility, including its safety-related structures, systems and components, and items important to safety; ensuring that the plant design meets the safety and numerical radiological criteria set forth in regulation and subsequent guidance. From this foundation, specific safety requirements have evolved through a number of criteria, procedures, and evaluations, as reflected in regulations, Regulatory Guides, standard review plans, technical specifications, license conditions, and various regulatory technical information documents.

The regulations, among others, pertaining to this issue are associated with various radiological performance criteria and include the following:

- 10 CFR 50.34(a)(1)(ii)(D), which requires an applicant for a construction permit to describe "safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur."

- 10 CFR 50.67(b), which requires an applicant seeking to revise its current accident source term, to provide "an evaluation of the consequences of applicable design-basis accidents previously analyzed in the safety analysis report."

- 10 CFR 100.11, which states when evaluating a proposed site, including deriving an exclusion area, a low population zone and population center distance, an applicant should assume a "fission produce release from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site."

Each of these regulations reference footnotes which define the source term that should be assumed for these evaluations. For example, the footnote referenced by 10 CFR 50.34(a)(1)(ii)(D) states:

> *The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.*

The reactor site criteria (10 CFR Part 100)[3] require for the purposes of licensing nuclear power plants that radionuclide releases to reactor containments associated with a "substantial meltdown" of the reactor core be postulated. The consequences of these radionuclide releases are evaluated against the radiological performance criteria associated with the regulations mentioned above, among others, for the control room and at certain offsite locations assuming that the containment remains intact and leaks at the design-basis leak rate. Radionuclides that leak from the containment are termed the "radiological release to the environment." The magnitude of the radiological release to the environment can be estimated from the containment leak rate and the radionuclide inventory suspended in the containment atmosphere as a function of time. The radionuclide inventory suspended in the containment atmosphere depends on the amount released to the containment as well as the effectiveness of natural and engineered processes that lead to radionuclide deposition within containment. The postulated radionuclide release to the containment is termed the "in Containment Source Term." The nuclear power plants currently operating in the country were licensed originally based on "in-Containment Source Terms" specified in Regulatory Guides 1.3 [USNRC,1974a] and Regulatory Guide 1.4 [USNRC,1974b],

To meet the Part 100 siting regulation, facilities were originally designed and sited with a historical source term published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, *Calculation of Distance Factors for Power and Test Reactors*. In the evaluation of the safety features of nuclear power plants, the past practice of the NRC staff has been to give no credit for any structure, system, or component that was not safety related (sometimes referred to as safety grade), assume the worst single active failure, and no credit for any non-seismic Category I equipment, components, and structures. Given this past practice, following a design-basis LOCA with no credit for non-safety-related components, and assuming the single failure of one MSIV to close, the design-basis maximum allowable leakage through the MSIVs would be the specified in the technical specification leakage limit. As mentioned, this limit is to maintain the offsite radiological consequences to within regulatory limits in the event of an accident.

Following the Three Mile Island Unit 2 meltdown, the NRC initiated a massive research effort in the area of severe accidents since the radionuclide behavior observed during the accident did not appear at all similar to the TID-14844 source term such as aerosol physics and radionuclide release and transport through the plant systems. The culmination of this work was published by the NRC in the publication, *An Assessment for Five U.S. Nuclear Power Plants (NUREG-1150)* [USNRC,1990], Research efforts focused on the development of source terms depended on accident phenomenology. From this body of research, a new set of generic "regulatory source terms" for representative BWR and PWR nuclear plants were derived and published in the NRC publication, *Accident Source Terms for Light-Water Nuclear Power Plants (NUREG-1465)* [USNRC,1995]. These revised source terms are more realistic, particularly in the areas of timing in the release of radioactivity to containment and chemical classes of radionuclides. For example, the TID-14844 source term is an instantaneous release at the beginning of the accident, while the NUREG-1465 source term is released over a two-hour period. Due to NUREG-1465 being more realistic, use of it for design basis accident radiological consequence assessments allowed removal of some systems earlier believed to be important to

[3] Applicants for a construction permit, a design certification, or a combined license that do not reference standard design certification who applied after January 10, 1997, are required by regulation to meet radiological criteria provided in 10 CFR 50.34.

safety and also permitted relaxation of requirements for other safety-related systems. In other cases, the use of the NUREG-1465 source term brought safety enhancements.

In December 1999, the NRC issued the new regulation, 10 CFR 50.67, *Alternative source term*, which provided a mechanism for licensed power reactors to replace the traditional TID-14844 accident source term used in their DBA analyses with an AST more consistent with the results published in NUREG-1150 and NUREG-1465. Regulatory guidance for the implementation of the AST is provided in RG 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*. The source term of Regulatory Guide 1.183 is an in-Containment Source Term. To date, nearly all licensees have adopted the AST as their licensing and design basis by applying the methodologies of RG 1.183.

In January 2011, Sandia National Laboratories (Sandia), reported on an updated accident source term, *Accident Source Terms for Light water Nuclear Power Plants Using High-Burnup or MOX Fuel*, (SAND2011-0128) patterned after the NUREG-1465 source term. This source term had been developed for high burnup fuel in BWRs and PWRs and for MOX fuel in a PWR with an ice-condenser containment. These source terms have been derived using nonparametric order statistics to develop distributions for the timing of radionuclide release during four accident phases and for release fractions of nine chemical classes of radionuclides as calculated with the MELCOR 1.8.5 accident analysis computer code. The accident phases are those defined in the NUREG-1465 Source Term – gap release, in-vessel release, ex-vessel release, and late in-vessel release. Important differences among the accident source terms derived here and the NUREG-1465 Source Term are not attributable to either fuel burnup or use of MOX fuel. Rather, differences among the source terms are due predominantly to improved understanding of the physics of core meltdown accidents. Heat losses from the degrading reactor core prolong the process of in-vessel release of radionuclides.

On May 13, 2020, in an NRC memo[4] reported on the applicability of source terms for accident tolerant fuel, high burn up and extended enrichment. Specifically, the Fuel and Source Term Code Development Branch (FSCB) staff has evaluated the applicability of RG 1.183, for:

- Rod-average burnups up to 68 GWd/t excluding potential impacts related to fuel fragmentation, relocation, and dispersal;
- Enrichment between 5-8 percent;
- Chromium-coated cladding (Cr-coated); and,
- Chromia-doped fuel.

For these concepts, the staff recommended using accident source terms from SAND 2011-0128 for the Containment Source Term to serve as a basis for a future RG 1.183 update. For higher burnups, the work performed and documented in SAND2011-0128 indicated little difference between the results of the low burnup and high burnup cases, even with the higher decay heat and fission product diffusivities in the high burnup case. Based on the limited impact of burnup effects between 38 GWd/t and 62 GWd/t, the staff felt it was reasonable to extrapolate the conclusion for fuel with 68 GWd/t peak

[4] See from Michael Case, Division Director for the Division of Safety Analysis, Office of Nuclear Regulatory Research, to Joseph Donoghue, Division Director for the Division of Safety Systems, Office of Nuclear Reactor Regulation, and Michael Franovich, Division Director for the Division of Risk Assessment, Office of Nuclear Reactor Regulation (ADAMS Accession Number ML20126G376)

rod-average discharge burnup. For enrichments up to 8 percent, is not expected to significantly impact accident source term release fractions or release phase durations. However, enrichment above 5 percent will impact fission product inventories used in source term calculations. For near-term ATF designs, such as chromium-coated and iron-chromium-aluminum (FeCrAl) cladding and chromia-doped fuel would have little impact on the source term.

Technical Basis

As discussed above, the NRC has provided accepted means to satisfy the siting criteria through a series of regulatory guidance. As part of this Guidance, the NRC has provided Containment Source Terms that are considered acceptable for siting. The approach to evaluating the Containment Source Terms has advanced from the initial approach.

The initial Guidance documents RG 1.3, *Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*, for BWRs and [USNRC,1974a] and RG 1.4, *Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors,* for PWRs [USNRC,1974b] were based on the TID-14844 source term [DiNunno, 1962].

TID-14844

Most of the currently operating power reactors in the USA were designed and licensed based on this "TID-14844 source term". This source term was based on results of experiments involving the heat up of irradiated fuel fragments in a furnace. Releases to the containment prescribed by the TID-14844 source term are:

Table 1 TID-14844 source term

Radioactive Element	Percent of initial core inventory released to the containment	Physical form of the released material
Noble gases (Xe, Kr)	100 %	gas
Iodine	50 %	gas
All others	1 %	aerosol particles

The source term was assumed to be instantly available in the containment. Half the iodine was assumed to deposit in route to the containment. The source term to the environment was evaluated assuming the design basis leakage rate for the containment and attenuation of the radioactive material available for release by engineered safety features (sprays, suppression pools, ice beds, etc.) of the plant.

Following the accident at Three Mile Island, it was evident that radionuclide release did not closely follow the pattern that might be expected based on the TID-14844 source term. Pressure arose from the nuclear industry for a more realistic source term. The Nuclear Regulatory Commission asked the Office of Nuclear Regulatory Research if it could define a more realistic source term and if that source term would be smaller than that used for reactor licensing. The Office of Nuclear Regulatory Research reply [USNRC,1981] was that insufficient data were available for defining a more realistic source term, but that it was possible to obtain the needed data and provide such a definition. The Office was directed by the Commission to undertake the required research.

Mechanistic Reactor Accident Source Terms

The route to a more realistic accident source term defined by the Office of Nuclear Regulatory Research was to develop a mechanistic linkage of radionuclide behavior including release from fuel and transport to the containment to reactor accident phenomena. Earlier work done as part of the Reactor Safety Study [USNRC,1975] to develop mechanistic accident source terms had been hampered by insufficient understanding of accident phenomena. Consequently, the research effort undertaken was very broad in scope. It involved the experimental investigation of core degradation, steam explosions, hydrogen combustion, fission product transport, core debris interactions with concrete and even the structural response of reactor containments. Major developments in the science of aerosol transport were necessary. Eventually, more than half a billion dollars were spent on the effort.

The research on the reactor accident source term led to the development of the Source Term Code Package (STCP) [Gieseke, 1986]. This was a suite of "stand-alone" computer codes linked together to mechanistically predict for a variety of accidents the source term to the reactor containment and the attenuation of the inventory of radionuclides in the containment as a result of natural and engineered processes. This first phase of the NRC's study of mechanistic reactor accident source terms culminated in the publication of improved source terms for use in regulatory processes, NUREG-1465, [Soffer,1995] and publication of level III analysis of accident risks at representative US nuclear power plants [USNRC,1990].

The NUREG-1465ContainmentSource terms for use in the regulatory process were developed for generic pressurized water reactors and boiling water reactors. In both, it is recognized that the release of radioactivity progressed in time. Accidents are divided into four phases: gap release, in-vessel release, ex-vessel release, and late in-vessel release. The fourth of these phases accounts for the revaporization of radionuclides deposited within the reactor during early phases of an accident. Release fractions of eight radionuclide groups are prescribed for each of these accident phases. With the exception of noble gases and iodine, all releases are presumed to be in the form of aerosol particles. Most of the iodine is presumed to be released to the containment as particulate, but 5% is taken to be gaseous and a fraction of this gaseous iodine is taken to be a volatile organic iodide.

The development of the NUREG-1465 C Source Terms involved analyzing a set of accident scenarios covering the majority of the core damage frequency for each of BWRs and PWRs and aggregating into a single table. For each scenario the plant response, the thermal-hydraulic behavior, the core degradation, along with the radionuclide release and transport to containment are evaluated. It is the timing and magnitude of the radionuclide chemical groups release to containment that constitute the source term. The Containment Source Term table values represent the 50th percentile timing, and the 70th (NUREG-1465) or 50th percentile (SAND-0128) radionuclide-chemical-group release fractions, to containment within each phase. A description of the process for developing the SAND2011-0128 source term along with the accident scenarios evaluated to both this source term and the NUREG-1465 source term are provided in Appendix D: High Burnup Containment Source Term Development.

The physical models in the STCP codes were developed and validated against the relevant experiments conducted during the post-TMI research effort. Details on the experimental and validation basis for the code can be found in the STCP manual [Gieseke, 1986]. The plant models represent the plant geometry, scenario, and response of safety features. Code output includes thermal and fluid behavior, metal oxidation, hydrogen generation and ignition, fission product release, and fission product transport and attenuation. The radionuclide releases evaluated from the several scenarios are aggregated into a representative timing, radionuclide release fraction table.

The NRC adopted the gap phase and in-vessel release phase NUREG-1465 source term as the Alternative Source Term [USNRC,2000]. These first two phases are listed in RG 1.183 Tables 1, 2, and 4.

Ongoing Source Term Research

The first phase of NRC's investigation of reactor accident source terms met the immediate regulatory needs. It was, however, well recognized that the understanding that had been developed through research was hardly complete. As nuclear power plants evolved and additional use was made of quantitative risk assessment in the regulatory process, better understanding of reactor accident source terms would be needed. The strategy for continued research involved two main thrusts:

- Development of an integrated model of severe accident progression to preserve and further develop the knowledge and understanding of severe reactor accidents gained from research.
- International collaboration on continued experimental investigation of severe reactor accident and source term phenomena.

The NRC has also transitioned to using the MELCOR consolidated severe accident code NUREG/CR-6119, Rev.2, SAND2000-2417 [Gauntt, 2000]. The MELCOR code integrates many of the features and physical models originally in the STCP but in an integrated, systems level, code. An integrated, systems level code allows the use of common libraries and databases, such as material properties and numerical methods, that can be updated for all physics packages in the code as a whole as new knowledge becomes available. An integrated code also enables the adoption of modern computer architecture and software design methods throughout the entire set of phenomena; incorporating well-established, documented, and tested numerical methods; and enhancing the existing validation basis. This approach also facilitates enhanced and integrated sensitivity and uncertainty analysis required in modern computational analysis

The NRC participated in several international collaborative research activities, including PHEBUS-FP, ARTIST, RASPLAV, MASCA, MACE, and Lower Head Failure Tests. The integrated fission product behavior tests in the PHEBUS-FP program ([Simondi-Teisseire, 2008]) were viewed as particularly crucial. Research had established that the chemical forms adopted by fission products during an accident had an important bearing on the behaviors of these fission products. Assumptions concerning assumed forms of these fission products were regularly refuted by experimental studies of various types. Prediction of chemical forms was made difficult by the diversity of possibilities in so complicated an environment as a reactor accident. Better guidance based on prototypical combinations of materials in realistic environments was needed to improve modeling capabilities. The investment in the PHEBUS-FP tests has been beneficial and the US is now participating in follow-on studies to understand better the complicated chemistries of iodine and cesium under reactor accident conditions.

Extension to High Burnup and Mixed Oxide Fuels

In order to prepare for anticipated applications for high burnup and mixed oxide fuel the NRC convened an HBU/MOX source term panel in 2002. This panel was a reconstitution of the panel that developed the source term uncertainty distributions for the NUREG-1150 study [USNRC,1990], which also served as the technical basis for NUREG-1465. In their evaluation, the panel considered the data and insight that have been generated since NUREG-1465 was published, and the physical phenomena that affect fission product release and transport mechanisms for high burnup and MOX fuels [ERI, 2002].

The panel did not consider fuel fragmentation since analyzed data did not indicate this to be a significant effect. The panel expected that high burnup could affect core degradation processes. The panel identified potential effects on fuel melting and fuel liquefaction processes: 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. They considered that degradation of high burnup fuel could involve 'fuel foaming' rather than fuel candling as observed with fuel at lower bumup levels. The panel concluded that high burnup 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. As such, the panel recommended further experimentation on high burnup fuel and cladding materials before defining a High-Burnup source term.

Since publication of the NUREG-1465 and adoption of the Alternative Source Term, research into the behavior of radionuclides under reactor accident conditions has continued, including for high burnup and mixed oxide fuels. In addition to experiments on radionuclide release from high burnup and mixed-oxide fuel in different gas environment (VERCORS [Ducros, 2001, Pontillon, 2010] and VERDON [Gallais-During, 2017]), notable undertakings include the PHÉBUS-FP project [Simondi-Teisseire, 2008] to investigate radionuclide release from degrading, irradiated, reactor fuel, transport of the released radionuclides through a simulated reactor coolant system and behavior of radioactive particles and vapors in a simulated PWR containment. There have also been studies of aerosol transport through steam generators with either wet or dry secondary sides [Güntay, 2008], iodine chemistry under accident conditions, and mitigation of aerosol production during the “ex-vessel release phase” of severe accidents.

These different research programs have led to improvement to the MELCOR code. The results of the 72 GWd/t VERCORS RT6 experiment were used to update the fission product release models for high burnup fuel in the MELCOR code [Gauntt, 2010].

The subsequently developed high burnup Containment Source Term involved the results of experiments on fission product release from high burnup fuel. These experiments did not exhibit the fragmentation observed in TH tests except following oxidation of fuel since the pressure was relieved from the samples when they were cut from the irradiated fuel rods.

The MELCOR code, updated for high burnup and MOX release behavior, was used to develop representative source terms for multiple accident scenarios covering the majority of the core damage frequency for different BWRs (SAND2007-7697 [Leonard, 2007]) and for different PWRs (SAND2008-6664 [Ashbaugh, 2010]).

The results of these analyses were aggregated into a Containment Source Term that is intended to be representative of releases from high-burnup and mixed-oxide fuels, SAND2011-0128 [Powers, 2011].

An expert panel was convened in 2011 to peer review this synthesis report and high burnup Containment source term [ERI, 2011]. The panel concluded that the “proposed source terms in [SAND2011-0128] are technically justified and appropriate” but recommended that Sandia National

Laboratories modify the characterization of early releases and provide additional documentation of the methods used for the calculations and of the accident progression results.

Consistent with the position stated in [Case, 2020], the results are considered applicable to peak rod-average burnups up to 68 GWd/t, despite the original analyses involving assumptions at a lower burnup. These analyses are considered to be applicable to higher burnups because the major impact from increased burnup, increased diffusivity, and decay heat history, are adequately accounted for. The 2011 SAND-2011-0128 [Powers, 2011] synthesis report and the contributing analyses and reports involved the use of radionuclide release diffusivities derived from the VERCORS RT-6 [Gauntt, 2010] radionuclide release experiment involving irradiated fuel with a burnup of 72 GWd/t. This expectation is further strengthened by the fact that, despite different decay power histories and substantially different diffusivities, high-burnup and low-burnup analyses produced similar Containment Source Terms (radionuclide release fractions and phase timings) [Powers, 2011].

Although the contributing analyses involved peak rod-average burnups of 62 GWd/t, the decay heat profile is not significantly affected by increasing the burnup to 68 GWd/t.

Containment Source Terms assessed for FFRD

The available Containment Source Terms are the Gap and In-Vessel releases from NUREG-1465 reproduced in RG1.183, tables 1, 2, and 4 for peak rod-average burnups up to 62GWd/t and SAND2011-0128 for peak rod-average burnups up to 62GWd/t. Table 2 and Table 3 compare the SAND2011-0128 BWR and PWR source terms with those in NUREG-1465, respectively. SAND2011-0128 compares the results of the two source terms in detail. The Containment Source Term table values represent the 50th percentile timing, and the 70th (NUREG-1465) or 50th percentile (SAND-0128) radionuclide-chemical-group release fractions, to containment, within each phase, of the considered accident scenarios. Appendix D: High Burnup Containment Source Term Development describes the process of developing a source term and the aggregation of the results of different scenarios.

These Containment Source Terms are assessed for the impact of FFRD in this report. Another way to more precisely state the question/problem that this report addresses is:

How does FFRD affect the radionuclide Release Fractions and timings of the Gap-Release and In-Vessel phases listed in these tables?

Table 2. Comparison of SAND2011-0128 BWR high burnup durations and release fractions (bold entries) with those recommended for BWRs in NUREG-1465 (parenthetical entries).

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

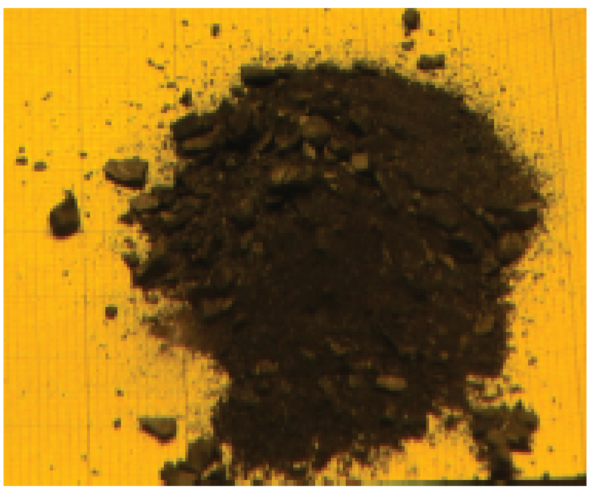
Table 3. Comparison of SAND2011-0128 PWR high burnup durations and release fractions (bold entries) with those recommended for PWRs in NUREG-1465 (parenthetical entries).

	Gap Release	In-vessel Release	Ex-vessel Release	Late In-vessel Release
Duration (hours)	**0.22** (0.5)	**4.5** (1.5)	**4.8** (2.0)	**143** (10)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr,Xe)	**0.017** (0.05)	**0.94** (0.95)	**0.011** (0)	**0.003** (0)
Halogens (Br,I)	**0.004** (0.05)	**0.37** (0.35)	**0.011** (0.25)	**0.21** (0.10)
Alkali Metals (Rb, Cs)	**0.003** (0.05)	**0.23** (0.25)	**0.02** (0.35)	**0.06** (0.10)
Alkaline Earths (Sr, Ba)	**0.0006** (0)	**0.004** (0.02)	**0.003** (0.10)	**-** (-)
Tellurium Group (Te, Se, Sb)	**0.004** (0)	**0.30** (0.05)	**0.003** (0.25)	**0.10** (0.005)
Molybdenum (Mo, Tc, Nb)	-	**0.08** (0.0025)	**0.01** (0.0025)	**0.03** (0)
Noble Metals (Ru, Pd, Rh, etc.)	-	**0.006** (0.0025)	[0.0025]	-
Lanthanides (Y, La, Sm, Pr, etc.)	-	**$1.5x10^{-7}$** ($2x10^{-4}$)	**1.3x10-5** (0.005)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	**$1.5x10^{-7}$** ($5x10^{-4}$)	**$2.4x10^{-4}$** (0.005)	-

Review of Fuel Fragmentation, Relocation, and Dispersal (FFRD)

The NRC and the international nuclear community have extensively studied fuel fragmentation, relocation, and dispersal associated with the intent to move to higher burnups. These efforts are captured in the literature; see, for example, Flanagan 2012 (NUREG-2119), Raynaud 2012 (NUREG-2121), Flanagan 2013 (NUREG-2160), Noirot 2015, OECD 2016, Capps 2020, and Capps 2021. These and other documents reviewed the regulatory history of fragmentation, reviewed the experimental database for fragmentation, and/or assessed consequences of FFRD for the design basis LOCA (10CFR50.46) event. Key results of this FFRD research are described below.

When the high-burnup fuel ruptures at temperature, fragments can be expelled from the fuel. Fuel with rod-average burnups lower than ~50 GWd/t do not experience this expulsion of fuel fragments. Fuel fragment expulsion has been observed upon rod in LOCA tests of irradiated fuel. Figure 1a shows fuel fragments collected from Studsvik test rod 193. The fuels community have characterized the transition from fuel fragments that are too large to likely escape the clad to small pulverized fragments that readily escape clad upon burst. Figure 1b shows this local-burnup/clad-burst-temperature fuel pulverization threshold developed by Turnbull that well describes fuel fragmentation behavior [Capps,2020] and [Capps,2021].



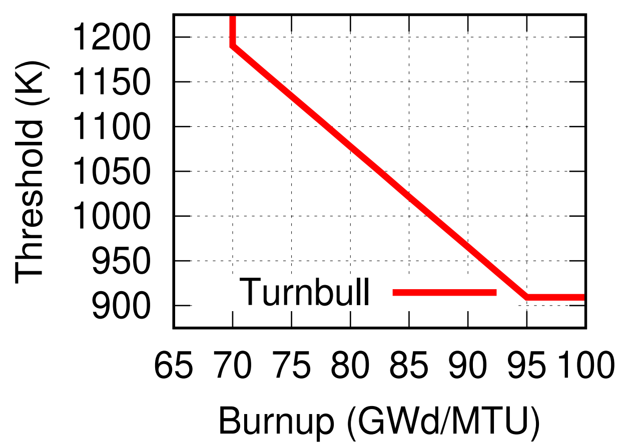


Figure 1 a."Sandlike" fuel fragments collected from Studsvik test rod 193" (From NUREG 2121 [Raynaud,2012]) , b. Turnbull burnup/rod-burst-temperature pulverization criteria, (info from [Capps, 2020])

The concern was raised that such fragments could cause problems that had not previously been considered in accident analyses. One concern is that fragments can settle, block cooling channels, and interfere with the cooling capability of the ECCS. The second is that fragments could be transported to the containment and thereby increasing the amount of radioactivity released to the environmental.

Assessing the impact of FFRD on the Containment Source Term involves an understanding of: the mechanisms of fragmentation; the fraction of fuel fragmented; the expelled fragment size distribution that governs transport behavior; the fragment power density and thermal properties; and, of the radionuclide release behavior with temperature and gas composition. Discussions of each topic are provided which brief literature reviews of earlier significant work and discussions of related factors that affect the impact of FFRD on the Containment Source Term.

In addition, an analysis is provided of the flow conditions expected during fragment expulsion and flight. This includes temperatures and thermal properties of fluids and structures that fragments are in thermal communication with by conduction, convection, and/or thermal radiation.

Fragmentation mechanisms

LWR reactor fuel contains gases at high pressure. Gas bubbles can form within grains and at grain boundaries. The gas pressure increases with burnup. The higher pressure in the grain boundaries drives fission gases to the gap/plena of fuel rods. Under accident conditions, the average fuel temperature rises which increases the gas pressure in the fuel, grain boundaries, and fuel-clad gap. Some species that are solid at operating temperatures may vaporize during accident conditions, further increasing pressure both within the intergranular gas bubbles and within the fuel-cladding gap. Large pressure differences between the fuel rod and the coolant lead to clad ballooning which removes the mechanical restraint provided by the cladding. This loss of restraint results in the formation of stresses in the fuel and can lead to fuel fragmentation.

Under accident conditions, pressures can burst the fuel clad. This results in a sudden reduction of the gas pressure in the fuel-clad gap to that of the surroundings resulting in a sudden large pressure differential between the gases in the grain boundaries and the surrounding gas. These changes in mechanical forces can cause the pellets to fragment. At low burnups, the fragments remain large and clad burst does not typically result in substantial release to the reactor coolant system. At high burnups, the fuel can fragment into very fine particles. The Turnbull threshold describes the fuel particle fragmentation size distribution as a function of both fuel burnup and temperature. At pellet-average burnups of 60 GWd/t, fine fragments generally originate from the high burnup structure near the outer radius of the pellet. At pellet-average burnup of ~78 GWd/t, the entire pellet is susceptible to fine fragmentation [Capps 2021].

Fuel fragment release and dispersal (FFRD) fraction

A crucial parameter that governs the impact of fragmentation on the Containment Source Term is the fraction of the core mass that fragments, referred to in this document as the "FFRD fraction." The impact of fragmentation on the source term is limited by the fraction that fragments and disperses. A small quantity of fragments is less likely to collect in thick layers that are difficult to cool. The expulsion of a large fraction of the core fuel as fragments would possibly result in a pile that could not be cooled and would thus heat up and release radionuclides.

Previous analyses have estimated the FFRD fraction.

The [Phillips, 2015] mobility analyses considered that between 35.0 kg and 207.3 kg of fuel fragments are dispersed depending on their chosen burnup threshold. This constitutes a very small fraction of the fuel. However, that study involved a core loading pattern with a discharge burnup of 62 GWd/t peak rod-average and used nominal best estimate conditions to calculate fuel temperatures used in rod burst predictions. (The variation in dispersed mass is due to uncertainties in the burnup threshold at which fuel is susceptible to fine fragmentation.)Phillips used the following criteria for dispersal: "The dispersed fuel masses presented in this section assume that all particles with a size below 1mm are fine particles that can and will be dispersed from the regions of the fuel rod near the rupture node if the cladding hoop strain is above 5%, which is assumed to be the threshold for fuel axial mobility and dispersal." Phillips noted that the burnup threshold at which fragmentation starts (50, 60, and 65 GWd/t) greatly affects evaluated dispersed fragmentation fraction. One of the main reasons is that many rods [with mean burnups below the assumed burnup threshold] can have sections of the rod that lies above the burnup threshold.

Large FFRD fractions could significantly impact the accident progression. FFRD fractions less than 2% (or even perhaps 4%) of the core inventory are likely too small to make a difference.[5] Lower fractions are less likely to collect to non-coolable configurations.   A FFRD fraction of 10% to 20% of core is likely to result in fragments collecting into a debris bed that may be non-coolable and thus heats up. Even for this large FFRD fraction, releases likely will not be greater than those Containment Source Terms calculated for and reported in NUREG-1465 and SAND2011-0128. As discussed above, these Containment Source Terms were developed to meet the requirements of those regulations requiring a source term being derived from such accidents having *"generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products"*. Therefore, the Containment Source Term already involves a large degree of core degradation consisting of both debris bed generation and fuel melt. A large debris bed may result in heat up of fragments in the debris bed and in release of radionuclides from the fragments in this debris bed. This will not however, increase radionuclide releases because the radionuclides released from the heated fragments in the debris bed are the same radionuclides that would have otherwise been released from the core under the standard source term. The Containment Source Term also involves the development of fuel debris bed during the later early in-vessel and ex-vessel release phases of the scenario as the core degrades. The impact of fragmentation could have a small effect on the timing of debris bed formation, boil off, and subsequent releases. Sufficient loss of fuel from the core by FFRD far in excess of current estimates could reduce the heat source to the extent that clad oxidation is substantially delayed, or potentially even prevented.

Because of the substantial impact of the FFRD fraction and concern about the potential for large values, an independent analysis was performed to estimate the FFRD fraction was made using prior characterizations of fuel fragmentation [Capps, 2021] and using a possible representative PWR core loading [Zhang, 2019]. The estimated FFRD fraction for the modeled core with a peak rod-average burnup of 76 GWd/t (47 GWd/t core average) returned a prediction of 0.0135. This estimated FFRD fraction is believed to be conservative for this particular loading pattern. Assumptions include failure of all rods in core, full pulverization of fuel locally (both axially and rod-radially) greater than 70GWd/t (lower (conservative) of Turnbull threshold) over a ballooning length of 1/7th rod length. The estimate was extended to different burnups to explore the trend. Accounting for local power and burnup can potentially result in some of the peripheral rods not failing in some scenarios. A more detailed set of assumptions and results are provided in the section below titled FFRD Fraction. The full set of assumptions and the FFRD fraction analysis are provided in

[5] Note that this statement only applies to scenarios used to develop the Containment Source Term. For dispersed fuel fractions less than 2%, mechanisms have not been identified that would significantly affect accident progression when ECCS cooling systems are already considered inoperable. This differs from the potential to affect ECCS functionality in 10CFR50.46 scenarios and analyses.

Appendix A: Estimation of fuel fragmentation fraction.

Fragment Size Distribution and Dependencies

The size and shape of a fragment, along with its density, determines how it transports and cools for given fluid conditions, including whether or not it can be lofted by gas flowing upwards at a certain velocity. These properties also determine how the temperature that particle will reach for a given power density and fluid temperature. The fragment size distribution provides information about the fraction of all fragment mass that behaves in a given manner. Similarly, as for a single fragment, the size distribution can be used to characterize the mass fraction of fragments that can be lofted by a specific gas flow or the mass fraction of fragments that heat up above a certain temperature for given thermal conditions.

The used size distribution was the same as that used by [Phillips, 2015]. This paper documented an analysis conducted by Sandia National Laboratories and NRC RES staff of mobility of fuel fragments for a 10CFR50.46 design-basis LOCA. This analysis characterized the fraction of fragments released from bursting fuel that would be mobile a specific large break LOCA accident scenario. This size distribution came from results of Studsvik experiments. This size distribution was conservatively extrapolated to sizes smaller than the smallest sieve by extending the size trend to sizes that are more relevant to the Containment Source Term analyses. This extrapolation assumes smaller sizes since smaller particles transport easier and are more likely to reach containment. In the extrapolation, a max fragment size of 5mm was arbitrarily assumed. This value does not affect the rest of the distribution but does factor into the extrapolated trends. Fragments larger than 1mm are not considered to be small enough to escape the clad at the burst location. The adjusted size distribution is shown in Table 4. Figure 2 shows the mass fraction in each adjusted bin. Figure 3 shows the corresponding CDF.

Table 4 Fragment fraction in bin, adjusted from Studsvik data. (adjusted values in blue). Data obtained from [Phillips, 2015]

Bin end size (mm)	55 GWd/t	70 GWd/t
0.016694		
0.03125	0	0.0597
0.0625	0	0.0749
0.125	0.0061	0.0899
0.25	0.0065	0.111
0.5	0.0058	0.1382
1	0.0051	0.1715
2	0.0099	0.1907
4	0.1926	0.1547
5	0.774	0.0094

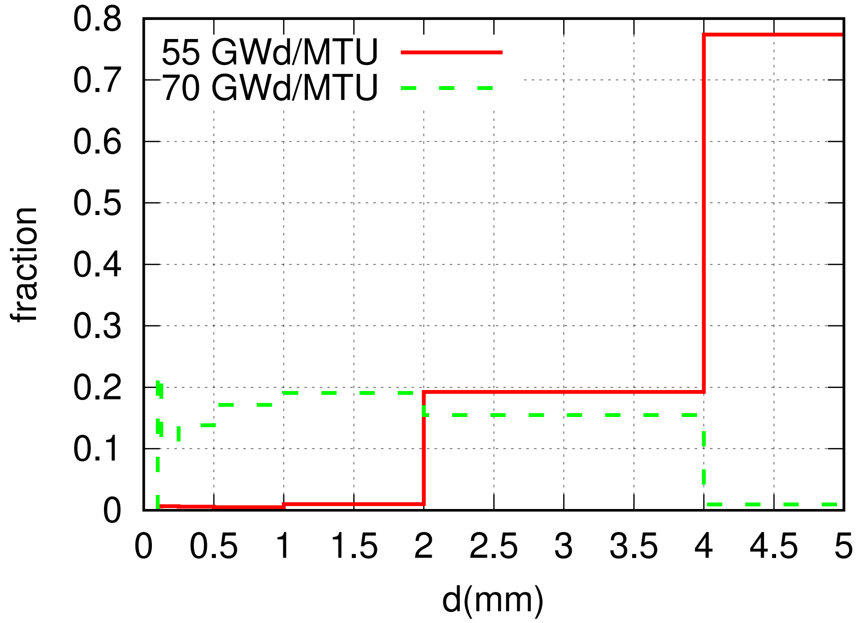


Figure 2 Mass fraction in fragment size bin

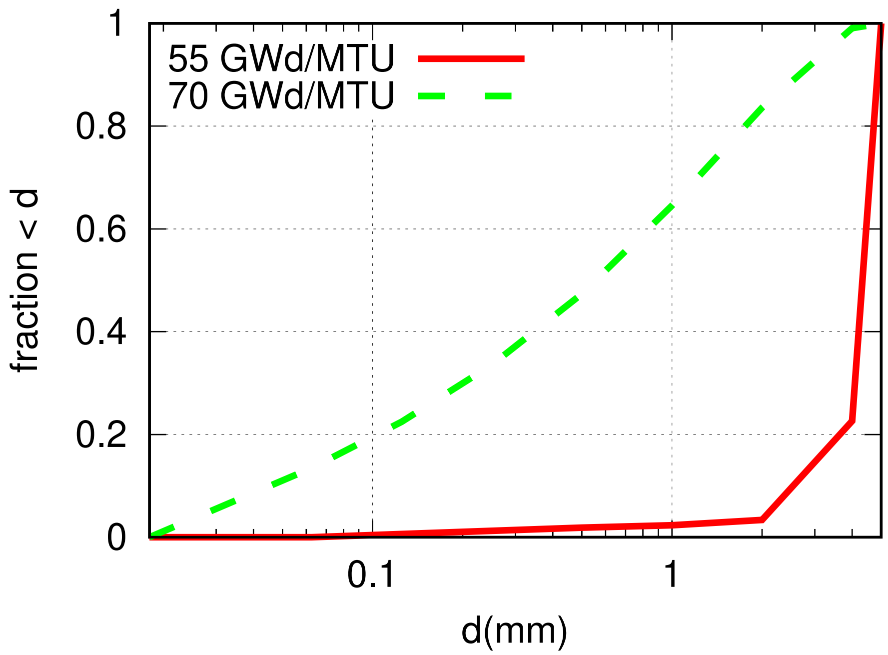


Figure 3 Fragment size CDF

NRC staff have speculated that fragments may break up upon impact with adjacent rods. No adjustment was made to account for this effect. It is considered possible that similar behavior (further fragmentation upon collision of nearby structures) may occur in experiments in which case the sieve-measured size distribution already includes the results of this effect.

Early noble gas release

While not strictly a fuel fragmentation effect, experiments show that at high burnup, noble gas releases are enhanced due to a higher fraction of fission gas being located on the grain boundaries. It has been speculated that fragmentation liberates the additional noble gases located on the grain boundaries. This would then constitute an earlier release of noble gases than that considered by the Containment Source Term described in NUREG-1465 and SAND 2011-0128.

AECL ((formerly) Atomic Energy of Canada Limited (now CNL – Canadian Nuclear Laboratories)) and CEA (Le Commissariat à l'énergie atomique) in France (Atomic Energy Commission) developed the "ADAGIO" technique and facility to characterize the distribution of noble gases, not only spatially within a pellet, but also to differentiate between noble gases located in the grain boundaries and noble gases located within the grains themselves. This technique was complementary with other approaches to characterization of the NG location with respect to grains. The facilities were improved, and the technique modified, in the GASPARD program over the course of two decades. The experimental techniques and results are described in several documents, including [Ravel, 2000], [Pontillon, 2004], [Noirot, 2014], and [Pontillon, 2018]. The research included modeling of observed behavior [Noirot, 2006], [Noirot, 2011].

The researchers characterized the releases in LOCA experiments with temperatures up to 1200 ^{o}C (~2200 ^{o}F). Table 5 shows the maximum ^{85}Kr release obtained. The release values in this table include puncture and release during heating. These values represent slow heat up with corresponding more time at temperature for releases. Faster heat up rates result in lower releases. These releases generally correspond to the ^{85}Kr content in the grain boundaries. These same grain-boundary gases are the ones that are subject to release by fragmentation. Comparison with the shorter-lived ^{133}Xe, which is primarily located within the fuel grains, allows the differentiation between grain-boundary releases and releases from within grains. Comparison of the results of the GASPARD LOCA program with those of VERCORS and VERDON Severe Accident programs, [Pontillon, 2010a] [Pontillon, 2010b] [Pontillon, 2010c][Gallais-During,2017], indicates that NG release from grains starts before the grain-boundary releases are complete. If fuel is heated rapidly, the timing of NG release from grain boundaries and grains will not be greatly affected[6]

These effects are discussed in more detail in Appendix B: Excess Early NG Release at higher burnup.

Table 5 Maximum NG LOCA Releases (Data from Pontillon, 2004 and Noirot, 2014)

Burnup (GWd/t)	Max ^{85}Kr Release up to 1200 ^{o}C
48.5	0.108
71.8	0.212
103.5	0.300 (fuel disc)

Given the biological effects of noble gases relative to that of other volatile radionuclides, and the magnitude of NG release considered in the current high-burnup source term calculations, it seems

[6] RIAs experience much faster heat up rates than LOCA scenarios and experience relatively large releases of noble gases due to fuel fragmentation. This behavior differs from the trends in LOCA tests.

unlikely, even considering early NG release, that fragmentation would result in greater consequences than the equivalent high-burnup source term without considering fragmentation.

NGs contribute most of the dose to equipment in containment over an extended period (~ 1GRad over a month) but do not contribute substantially to personnel or public doses in the DB source term analyses. This limits the impact that earlier NG release can have on the Containment Source Term.

Another factor that limits the increase in NG contribution to source term by fragmentation-related processes is that the NG release for high-burnup fuel is near total by the end of the in-vessel phase. There is nowhere for the NG source term for high burnup fuel to go but down. This can be seen in Table 6 and Figure 4. The table shows the Gap and In-Vessel releases for some of the highest-NG release HBU PWR scenarios in the SAND2011-0128 source term. The figure shows the In-Vessel PWR releases for both low burnup and high burnup cases in the SAND2011-0128 source term.

Table 6 : Some contributing numbers to NG release values in the HBU ST: (SAND2008-6664 [Ashbaugh, 2010], Input to SAND-2011-0128 [Powers, 2011])

Scenario	Gap	IV	Gap+IV
1A Surry SBO	0.016	0.978	0.994
1D Surry SBO w/RCP seal LOCA	0.021	0.977	0.998
1C Surry LLOCA ECCS Inj	0.008	0.949	0.957
4F Sequoyah LLOCA with AFW	0.017	0.945	0.962

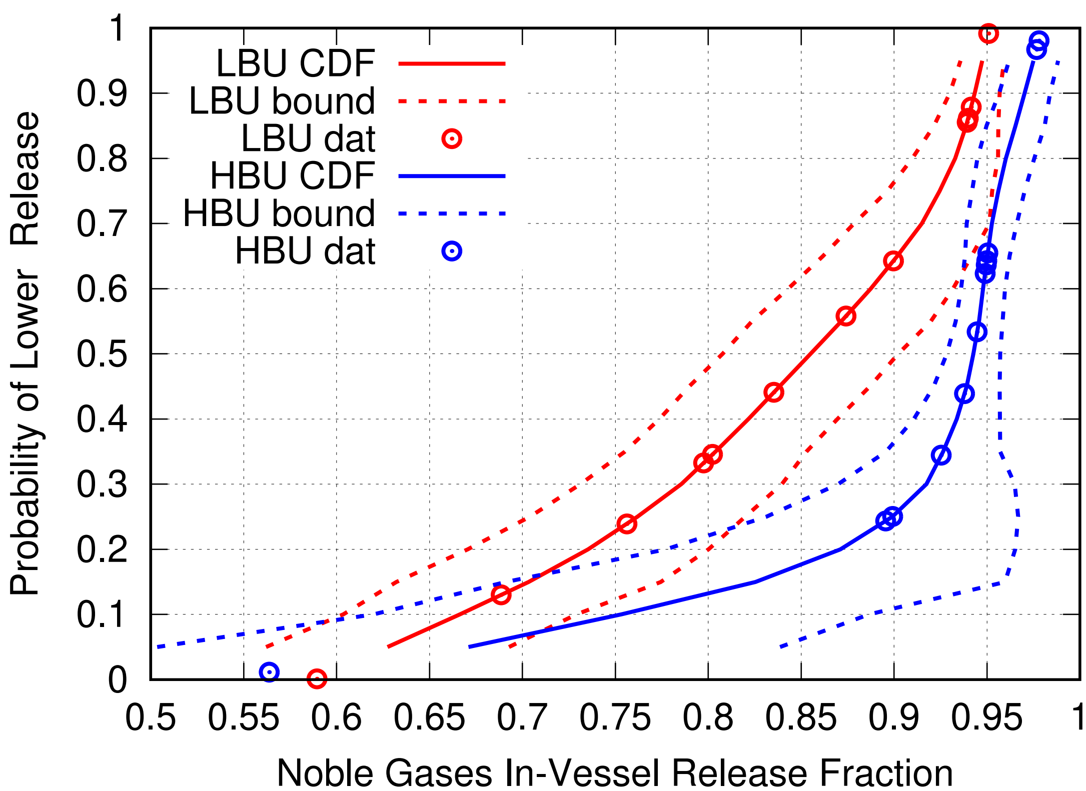


Figure 4 PWR HBU-LBU In-Vessel NG RF comparison, data from SAND2011-0128 analyses

It is expected that fragmentation will bring the NG source term down, even if all grain-boundary gases are assumed to be liberated upon fragmentation, and even if short-lived grain boundary gases are assumed to have similar fractions in the grain boundaries as ^{85}Kr (which they don't), because it is likely that, for most situations, fragments will never heat up to the extent they substantially release their in-grain noble gases.

Given that only minor changes in NG timing are expected because gap and in-vessel releases overlap in the core, in addition to overlap of NG in-grain and grain-boundary release within the fuel itself, that the biological effect of NGs are low compared to other volatile RNs, and that the fragmentation will reduce overall NG releases, scoping MELCOR calculations that involve bounding NG release evaluated by interpolating Table 5 are recommended. In reality, releases from shorter-lived NG isotopes will have substantially lower grain-boundary fractions. Furthermore, faster heat up could result in lower predicted early NG releases since there would be less time for diffusion of gases from the grains to the grain boundaries. Finally, higher system pressures may reduce fragmentation-induced early NG releases [Une et al., 2002], [Turnbull et al., 2015]. These can be accounted for if necessary. It would be preferable to use more representative NG fractions in future analyses. Substantial data and modeling exist to refine NG release prediction if needed or desired. (See [Jernkvist, 2019] for one such model for transient fission gas release that has been incorporated in Quantum Technologies AB's version of the FRAPTRAN-1.5 fuel performance code.)

A closer look at this early NG release behavior may be useful for non-LOCA scenarios in RG1.183.

Insights on fragmentation by the Accident Tolerant Fuel - Phenomena Identification and Ranking Table Process (ATF-PIRT)

In 2020, the NRC began preparing for anticipated licensing applications and commercial use of accident tolerant fuel (ATF) in the U.S. commercial nuclear power reactors. Several fuel vendors, in coordination with the Department of Energy (DOE), had announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance (e.g., fuels with longer coping times during loss of cooling conditions). Vendors had also expressed interest in increasing fuel burnup above the licensed limit (which varies by vendor, but roughly corresponds to 62 GWd/t rod-average), as well as increasing enrichment beyond 5%.

In support of preparing for reviewing ATF designs, the NRC sponsored a Phenomena Identification and Ranking Tables (PIRTs) exercise to assess severe accident implications of certain ATF concepts. The PIRT process is a systematic way of gathering information from experts on a specific concept, and ranking the importance of the information, in order to meet some decision-making objective. It has been applied to many nuclear technology issues, including nuclear analysis, to help guide research or develop regulatory requirements. The execution of the PIRT process on a specific concept is called an exercise and these exercises will vary greatly in scope and depth based on the ATF concept's departure from the current state of practice and its maturity.

In addition to evaluated ATF concepts, the PIRT panel addressed the behavior of high burnup and high assay low enriched fuel (HALEU) with less than 20 w/o% U-235 that is also of interest to the NRC. In the present context, HALEU is viewed as increased fuel enrichment for use in LWRs, and not as envisioned for non-LWRs. The LWR industry is anticipating fuel enrichments as high as 10%. Results of the ATF-PIRT are documented in [ERI,2021].

As part of the HBU/HALEU review the PIRT panel discussed the impact of fragmentation on severe accidents and the Containment Source Term.

The ATF PIRT source term panel indicated that they did not expect fragmentation to have a significant effect for severe accidents and the Containment Source Term. The analyses in this report support the expectations expressed by the ATF-PIRT panel.

Some of the points made by the panel regarding fragmentation of high-burnup fuel include:

- Fuel is expected to have a different amount of fragmentation or sintering at higher burnup, which can in turn affect how the fuel behaves during core degradation and relocation, as well as phenomena such as temperature-induced creep rupture, which can be affected by debris particle size.
- Regarding Gap Inventories/Pressure and Release at Cladding Failure: The potential for fuel fragmentation for HBU under LOCA conditions that can result in some increase in noble gases, etc. (due to particulate release to the reactor coolant system) is insignificant in terms of its impact on severe accident source term. Among the ideas that were discussed related to the short or medium terms consequences of the contemplated ~1 mm particles that could result in a heavy fission gas load to the containment. During the long-term recovery, leaching would be a potential issue. Furthermore, the panel considered it unlikely for the possibility that any fragmented particles could, via leakage into or from the ESF lines, cause this normally liquid leakage to be considered as a gaseous leakage (i.e., resulting in consequences on the environmental release of radionuclides).

- Regarding Relocation Phenomena: There is the potential for more fragmented fuel (in the absence of reflood) for HBU and for more likelihood of debris slumping with consequences for coolability inside the lower head. Phébus (lower than 35 GW/t burnup) showed fragmentation (which were large as compared to what is expected for HBU), but this fragmentation did not seem to have an impact on relocation. HBU fuel will be more fragmented than conventional fuels (without consideration of reflooding); this will affect slumping; the sintering of the pellets may (or may not) be less complete.
- Regarding aspects of in-vessel coolability:
  - The potential for embrittlement has not been examined for HBU fuel. It is expected that HBU fuels have a higher potential for fragmentation. Especially the behavior of higher Pu content needs to be explored.
  - The panel noted that there is greater potential for recriticality for HBU/HALEU fuels due to the higher enrichment, which can enhance/speed-up degradation of fuel and fission product release. It may also reduce the potential for core damage arrest. For instance, there is the potential that some part of the core may go critical while the rest of core remains sub-critical, and the part that goes critical may be mostly un-degraded, with result that the re-criticality is immediately responsible for much of the fission product release from fuel.
- The panel did not note any impacts of fragmentation for lower head and ex-vessel phenomena.
- One panelist noted that calculations in MAAP-MELCOR crosswalk [Luxat, 2014] suggested that fragments may enhance core-to hot leg heat transfer, accelerating its rupture [and thus reducing likelihood of induced SGTRs]. Two possible effects could enhance the heat transfer: direct heating from fragments settled in the RCS and enhanced steam generation from fragments that settle in water in the lower head and release their heat there. These effects could be explored with MELCOR calculations.

The panel considered the impact on severe accident source terms (e.g. Containment Source Term) of possible high burnup fuel fragmentation under the maximum hypothetical accident LOCA conditions, related to increased release of noble gases, and the potential for formation of particulates and their transport into the reactor coolant system, to be insignificant.

FFRD Impact on DBST

This section evaluates the impact of FFRD phenomena on the Containment Source Term to estimate the impact of FFRD on radiological releases to containment. The evaluation built upon considerable previous work on Severe Accidents, Source Term, and FFRD, and on insights from the recent ATF-PIRT. The evaluation focused on the release of iodine, cesium, and noble gases.

The goal of this evaluation is to evaluate whether the timing and release fractions of NUREG-1465, SAND 2011-0128, and RG 1.183 Rev. 0 Tables 1, 2, and 4Containment Source Term tables, developed with the current approach remain representative or bounding once FFRD phenomena are considered. The potential FFRD changes of concern are greater radionuclide release fractions or earlier release timing. It would also be useful to quantify the extent to which FFRD may affect the release fractions and timings if possible. The values in the source term table are not arbitrarily decided on. They are evaluated as representative releases and timings evaluated by mechanistic severe accident calculations conducted for a representative set of severe accident sequences that covers nearly the entire core damage frequency space. To understand the impact of fuel fragmentation on the Containment Source Term, it is also necessary to understand the regulatory bases, technical basis, status, and approach to developing Containment Source Terms. This is discussed in detail in The Containment Source Term section above.

To understand the impact on the source term, one must consider the threat to public health and safety that dispersed fuel fragments can pose. Fuel fragments carry with them less-volatile fission products that are not released in appreciable quantities when fuel melts. These radionuclides can potentially increase the dose from a given amount of fuel over the volatile-only release from the same amount of fuel. On the other hand, radionuclides from ingested or inhaled fragments, in addition to the fragments being far less likely to be inhaled, would absorb into the body at lower rates than from aerosols since these radionuclides would have to escape the larger fuel particles first.

Fuel fragments behave very differently to the nuclear aerosols that contribute to the normal source term and can pose a health hazard. Nearly all fuel fragments are far larger and heavier than typical nuclear aerosols produced under severe accident conditions and settle at far greater rates. As a result, fuel fragments do not transport as easily as nuclear aerosols, are less affected by flow recirculation and turbulence, and are unlikely to successfully travel offsite or be inhaled. Most fuel fragment mass lies in fragments that are far larger than those typically considered “respirable” (< 10 μm aerodynamic diameter). Therefore, for dispersed fuel fragments to significantly contribute to personnel or public dose from the Containment Source term DB analyses, the fragments must heat up to the extent that they release their volatile radionuclides. The rate of release of radionuclides, and of fuel oxidation/reduction, depends on temperature and the local gas composition. Volatile radionuclides that are released from fuel fragments that reach containment are not attenuated by the RCS in the same manner as volatile radionuclides released from the core.

If particles do not get hot, they do not release their radionuclides. An exception to this is that some radionuclides can be leached from fragments submerged in water. Radionuclides can leach out of smaller fuel fragments faster than from the sizes typically assumed from core debris resulting from a severe accident.

The evaluation of the impact of FFRD on the source term is centered around the evaluation of the circumstances for which dispersed fuel fragments could heat up and therefore release their volatile radionuclides. It should not be forgotten that a large fraction of these volatile radionuclides would have been predicted to be released and would have contributed to the Containment Source Term had the fragments remained with the rest of the fuel in the core.

The governing question that determine FFRD impact on the Containment Source Term are:

- How much fuel fragments?
- What is the fragment size distribution?
- What fragments can escape rod?
- Where do the fragments go?
- How much do they heat up?
- How much RNs are released from fragments?

A brief discussion on the interrelations of these items follows. Each element is considered individually in more detail later.

Fuel fragments can behave differently in different accident scenarios because different scenarios have different decay powers due to elapsed time between shutdown and start of releases and because the fragments can be exposed to different conditions (e.g. high pressure or low pressure, different gas velocities, different leakage pathways to containment, impact of subsequent accumulator injection or not).

The rate-limiting phenomena for radionuclide release is typically atomic diffusion within fuel grains although mass transfer can possibly limit releases. The temperature difference between fuel at the center of the core and that of dispersed fragments governs the relative impact of fragmentation. Exposed to similar gases the release rates for similar temperature profiles should generally be similar independent of location. It is expected that an analysis of the heat losses (conduction, convection, and radiation) would show that the temperatures of fuel fragments dispersed either upwards or downwards would not approach that of fuel that remains in the center of the core where clad burst and fuel fragment dispersal are more likely.

Fragments that fall downwards can fall in water and remain cool. Fragments that fall downwards and get held up on structures can both conduct- and thermally radiate to structures far cooler than in the middle of the core. Similarly, fragments that transport to reactor coolant system piping or to containment, if they remain dispersed, conduct- and thermally radiate to structures far cooler than in the core centerline. An additional effect is that dispersed fuel is not directly exposed to the heat released upon clad oxidation since the fragments have moved away from the clad.

Some radionuclides can be released at lower temperatures in reducing (hydrogen) or more oxidizing (e.g. air) conditions. Even when accounting for potential differences in releases due to exposure to different gases, it is expected that the temperature difference would dominate behavior.

Dispersed fragments in the reactor coolant system or containment see a lot of thermal mass that limits their possible temperature rise. The maximum radiative heat flux emitted by a surface is given by the Stefan-Boltzmann law, $q''=\sigma T^4$. The heat flux from a real surface is $q''=\varepsilon\sigma T^4$. In these equations, q'' represents the heat flux, σ the Stefan-Boltzmann constant (= 5.67E-8 $W.m^{-2}.K^{-4}$), T the surface temperature, and ε the emissivity. With a temperature to the fourth power dependence, the seen mass must rise in temperature. The transparency of the gas to the thermal radiation being emitted should be considered in a thermal analysis. So should the presence of aerosols carried by the gas which can both absorb and scatter the thermal radiation.

It appears that the size distributions/flow patterns are such that it is unlikely that fragments can collect in large quantities far away from the core in the scenarios used to develop the Containment Source Term, other than perhaps in the lower head, since fuel fragments are very large and dense relative to aerosols. Complicated flow channels tend to separate out different particle sizes, thus dispersing the mass rather than collecting in a single location.

Fuel fragments are far more easily removed from flow streams than the aerosols. Other things being equal, settling rates and the rate of deviation from streamlines, correspond to the diameter squared. Fuel fragments are on the order of 1000 times larger than aerosols on average so settle nearly a million times faster (the increase in settling rate with increasing diameter slows as the flow around the particle transitions to turbulent). Fuel fragments will also collide into obstacles that some aerosols miss. One must of course consider the density and shape of particles, the lower tails of both aerosol and fragment size distributions that are most likely to navigate obstacles, and the fact that aerosols can grow by agglomeration and vapor condensation. The release fraction of fuel fragments to containment would be less, and likely far less, that of radionuclides under the same conditions. One should consider different scenarios and that clad burst and fuel fragment release and dispersal could occur at different times in a scenario when the gas velocity through the core is higher. Given the disparity in removal rates, it is difficult to conceive of a scenario where fragments could transport to containment even when considering these additional effects. Given the large size of fragments, it seems possible that a significant fraction of the fuel fragment mass may not even successfully transport outside of the vessel prior to lower head rupture for most scenarios.

One effect that has previously been considered is that substantial fragmentation could affect accident progression in some cases. If a large fraction of fuel fragments and settles during a PWR station blackout scenario the additional heat to water in the lower head would enhance steam generation which results in transporting more heat from the core to the hot leg resulting in earlier hot leg creep failure. The hot leg failure would not occur in other scenarios in the development of the source term that postulate reactor coolant system failure as an initiating event. It seems that more than 2% fuel fragmentation would be necessary to significantly affect lower head boiloff timing and that the change in timing from 2% fuel fragmentation would not be significant.

Experimental results indicate that a higher proportion of long-lived noble gases reside at the grain boundaries with increasing burnup. The concern exists that grain-boundary noble gases could be liberated upon fragmentation. This behavior could result in a higher proportion of early noble gas release.

Even though a mechanism for significant release of fuel fragments to containment has not been identified, one can consider the situation where fuel fragments somehow transport into containment and somehow collect in debris bed, heat up, and release their volatile radionuclides.

Even if 2% of the fuel fragments and disperses with the 70 GWD/t fragment size distribution, most of the mass will initially settle downwards upon clad rupture except for large-break LOCA scenarios. Of the fragments that initially go upwards some fraction won't reach leave the reactor vessel. Of the fraction that leaves the vessel some will settle or get trapped before it makes it to containment. Some of the fragments that make it to containment will not make it to a sump where they could be pumped through the ESF piping. It does not seem likely that large fragments can reach the ESF piping in fractions approaching that of radionuclide aerosols released from the core.

For fragments to significantly impact releases by dry out in ESF, one must assume significant fraction of the 2% of fuel fragments somehow reaches the containment, reaches the ESF piping, collects into a debris bed, heats up and starts releasing fission products. Fragments consisting of 2% of the fuel should contain around 2% of the fission products in the core, or perhaps somewhat more because of the variation of burnup throughout the core and along both the length and radius of fuel rods.

Even if a significant fraction of fuel fragments somehow collects in a debris bed and completely released their volatiles, the around 2% of total possible release would not constitute a significant change to the 37-54% (PWR-BWR) halogens or 13-23% (BWR-PWR) alkali metals evaluated in the SAND 2011-0128 Containment Source Term which are representative values for multiple sequences. The variability in release and timing from the scenario choice far exceeds the possible change of 2% volatile release from fragments. (For example, the variability in halogen release in calculations the ranged from less than 2% to greater than 98%). The uncertainty in the median RF (which the first numbers in this paragraph represent) was around 20% of release.

If these fragments were exposed to air rather than steam, other radionuclides could be released, e.g. Ruthenium which is considered to result in similar short-term consequences as iodine and long-term consequences as Cs.

Leaching from dispersed fuel fragments to water has been discussed as a concern. Although the rate is faster than for typical fuel debris, the effect is limited by the fragment fraction.

Criticality was identified as a potential high-burnup, if not FFRD, issue, since fuel enrichment may be increased to allow for higher burnup. The fuel designed for high burnup is more susceptible when the fuel resides at burnups low enough that fragmentation does not occur.

Approach

The FFRD behavior was applied to the Containment Source Term using the following approach.

The approach is to first address the impact of the limiting case, then research and perform scoping calculations, as needed, on the contributing elements to the FFRD source term to refine the estimate of each element. Other requested "what if" scenarios were considered and analyzed.

The limiting case can be described by the following question: "What if it all fuel that fragments somehow reach containment, somehow heat up, and release all their volatiles?". The evaluation of the fraction of fuel that fragments and disperses was a necessary input to this calculation. Although wide variation in the considered fragment fraction existed, an independent assessment of the FFRD fuel fraction supported the expectation that <2% of the fuel fragments and disperses for expected fuel loadings with peak rod-average burnups in the range of 75 GWd/t.

The following elements that factor into FFRD source term were investigated in more detail:

- FFRD fraction
- Fragment size distribution
- Fragment transport
- Fragment heat up
- Radionuclide releases from fragments
- Source term timing effects
- Overall fragment behavior for different accident scenarios

These effects were explored by further literature review and scoping calculations, where applicable. Other issues affecting the FFRD source term were also explored:

- What if significantly more fuel fragments?
- Leaching
- Criticality
- ESF leakage evaporation

Based on the findings of this investigations and the insights from the recent ATF PIRT [ERI, 2021] an estimate of the expected FFRD impact on the Containment Source Term was made.

Sequence evaluations

The different elements are intended to be applied, if not to each sequence, then to representative sequence classes, to evaluate the FFRD impact on source term. The individual scenario-specific source terms can then be adjusted for FFRD with the resulting representative value extracted.

The following sequence classes were considered:

- Large break LOCA – hot leg break – uniquely easy access of fragments to containment
- Large break LOCA – other scenarios
- High-pressure boiloff scenarios (station blackout (SBO) + time shift)
- Low-pressure boiloff scenarios

For each scenario, or scenario class, the following was planned:

- Compare gas velocity at rupture to discriminate the fraction of fragments that rise or settle.
- Review the leakage pathways and assess the fraction of fragments that are likely to traverse past the most-restrictive obstacle (bends).
- Identify potential fragment collection points and assess possible debris bed depth.
- Evaluate fragment temperature accounting for decay time between shutdown.
- If fragments can get hot, evaluate radiological releases.

It was considered unlikely that analysis could be done with the allotted resources and available time since it required functional scoping calculations in essentially all elements, in addition to MELCOR code output for the sequences or sequence classes, before it could be performed.

Therefore, simplified scoping calculations were performed when useful and to the extent possible. Any analyses that were not completed by the deadline are discussed qualitatively.

Limiting FFRD Impact

The limiting case can be described by the following question: “What if it all fuel that fragments somehow reach containment, somehow heat up, and release all their volatiles?”. The evaluation of the fraction of fuel that fragments and disperses was a necessary input to this calculation. Although wide variation in the considered fragment fraction existed, an independent assessment of the FFRD fraction supported the expectation that <2% of the fuel fragments and disperses for expected fuel loadings with peak rod-average burnups in the range of 75 GWd/t.

This bounding case consists of the expected maximum fragmentation (2% of the total fuel mass in the core) and assumes that all radionuclides successfully transports to containment, collects into an uncoolable configuration, and heats up enough to release all volatile RNs.

To come up with this limiting case, one has to make conservative assumptions that are inconsistent with expectations:

- Assume significant FFRD (2%).
    - Data and bounding analyses show small FFRD fraction.
- Assume fragments reach containment
    - Not credible for most fragment mass and scenarios, even when using conservative assumptions.
- Assume fragments heat up and release all volatiles.
    - Fragments may disperse and not collect to debris beds.
    - Not enough power in dispersed fragments to heat up surroundings.
    - Fragments unlikely to heat up to extent that volatiles can be released.
- Neglect the fact that a significant fraction of the same volatiles would have been released from the fuel within the core.
    - No change in FFRD source term likely since nearly all volatiles would have been released anyways from center of core where fragments originate (FFRD source term same as SAND2011-0128 Containment Source Term)
        - This is likely to well-approximate behavior since the center of core from which fragments originate release essentially all volatiles. Some volatile radionuclides not released from peripheries and extremities.
    - Some value in-between if one were to assume only partial volatile release from center of core.

Table 7 and Table 8 show the effect on releases with this limiting case. Even with these unrealistic conservatisms, fragmentation and dispersal has only a minor effect on the Containment Source Term. A significant increase in the source term from FFRD phenomena cannot be concluded with realistic assumptions.

Table 7 PWR Gap and In-Vessel Release Fractions, pessimistic assumptions, modifications to SAND2011-0128 Containment Source Term (parentheticals correspond to 2% FFRD adjustment). No noble gas modification.

PWR	Gap Release	In Vessel Release
Noble Gases (Kr,Xe)	**0.017** (0.017)	**0.94** (0.94)
Halogens (Br,I)	**0.004** (0.004)	**0.37** (**0.39**)
Alkali Metals (Rb, Cs)	**0.003** (0.003)	**0.23** (**0.25**)

Table 8 BWR Gap and In-Vessel Release Fractions, pessimistic assumptions, modifications to SAND2011-0128 Containment Source Term (parentheticals correspond to 2% FFRD adjustment). No noble gas modification

BWR	Gap Release	In-Vessel Release
Noble Gases (Kr,Xe)	**0.008** (0.008)	**0.96** (0.96)
Halogens (Br,I)	**0.002** (0.002)	**0.47** (**0.49**)
Alkali Metals (Rb, Cs)	**0.002** (0.002)	**0.13** (**0.15**)

FFRD Fraction

As noted, an independent estimate of the FFRD fraction was made using prior characterizations of fuel fragmentation [Capps, 2021] and using a possible representative PWR core loading [Zhang, 2019]. The FFRD fraction estimate is provided in

Appendix A: Estimation of fuel fragmentation fraction. The estimate returned a prediction of 0.0135 for the FFRD fraction for a core with a peak rod-average burnup of 76 GWd/t (47 GWd/t core average). This is believed to be a conservative estimate for reasons described below. The estimate was extended to different burnups to explore the trend.

Figure 5 shows the trend in FFRD fraction with changing burnup.

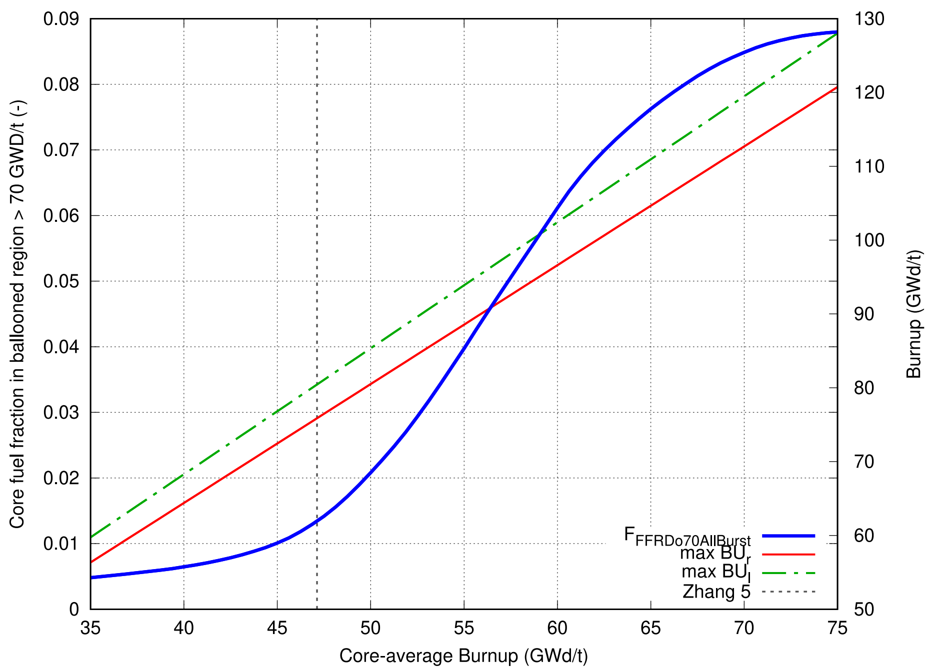


Figure 5 Bounding FFRD fraction (fraction of local (axial and rod-radial) fuel > 70GWd/t)(extrapolated far beyond realistic values to explore trends in fragmentation behavior))

It is known that this curve extends way outside its possible range of validity and beyond expected currently possible ranges for fuel. The wide range beyond the original source (Zhang) was chosen to observe the trends that would result by extrapolating the behavior. If loadings at higher burnup are similar, the trends may remain valid beyond the physically achievable burnups. If loadings at higher burnup differ substantially, they will not be applicable.

The analysis has some limitations: It considered only 1 loading and 1 rod-radial burnup distribution. These could change with burnup. It only addressed a Westinghouse 4-loop PWR. A similar BWR analysis was not conducted.

The analysis involved some significant conservatisms:

- A ballooning length equal to 1/7th the rod length (~52 cm) was assumed. The actual balloon length will likely be smaller, based on insights from tests performed at Studsvik, Halden, and ORNL, which would decrease the FFRD fraction estimated here.
- Burst was effectively assumed to be at high temperature and thus high ΔP (by assuming a 70 GWd/t Turnbull pulverization threshold). Lower burst temperatures would result in a higher burnup at which pulverization occurs and thus an even lower FFRD fraction.
- All rods are assumed to burst, including the lower power, highest burnup rods in peripheral bundles. This effect would further reduce the FFRD fraction for scenarios where peripheral

bundles don't rupture. The effect would be even greater for recovered DB LOCA scenarios for which a significant fraction of internal rods does not rupture.

The results of this analysis support prior expectations of a 2% bounding FFRD fraction. Half a percent (or even less) fragmented may be more realistic.

The model can be combined with MELCOR results to come up with scenario-specific fragmentation fractions that considers whether rods fail and, if rods fail, the fuel-rod specific FFRD fraction based on the rupture temperature and Turnbull limit.

FFRD Early NG Release

While not strictly a fuel fragmentation effect, experiments show that, at high burnup, noble gas(NG) releases are enhanced because of a higher fraction of fission gas located on the grain boundaries. It has been speculated that fragmentation liberates the additional NGs located on the grain boundaries. This would then constitute an earlier release of NGs than that considered in the current Containment Source Term. The early NG release effects are discussed in more detail in the section above titled Early noble gas release above and in the Appendix B: Excess Early NG Release at higher burnup.

Scoping MELCOR calculations that involve bounding NG release evaluated by interpolating Table 9are recommended (Note: this table reproduces Table 5 to include the numbers in the). Although the values in this table were evaluated by heating samples, they generally correspond to the fraction of NGs at the grain boundary and that would potentially be subject to release from a fuel fragmentation event. The fractions in this table only apply to long-lived ^{85}Kr. In reality, releases from shorter-lived NG isotopes with greater dose contribution will have substantially lower grain-boundary fractions. Furthermore, faster heat up would result in lower predicted NG releases. These can be accounted for if necessary. It would be preferable to use more representative NG fractions in future analyses. Substantial data and modeling exist to refine NG release prediction for best-estimate calculations if needed.

Table 9 Maximum NG LOCA Releases (Data from Pontillon, 2004 and Noirot, 2014)

Burnup (GWd/t)	Max ^{85}Kr Release up to 1200 ^{o}C
48.5	0.108
71.8	0.212
103.5	0.300 (fuel disc)

Fragment transport

Where fragments end up affect whether they will heat up to the extent that they release radionuclides. Fuel that stays in the core will release radionuclides. Dispersed fragments that deposit on surfaces where very hot gases emanating from the core pass may heat up to the extent that they release their radionuclides. Dispersed fragments in cold regions will not heat up significantly beyond ambient conditions. Dispersed fragments submerged in water will not heat up. Submerged debris beds can be typically cooled for thicknesses up to 10cm [Lipinski,1982]. Fragments that collect in dry locations likely do not need to achieve thick layers before they heat to the extent that they release volatile radionuclides.

Evaluating where fragments end up is an important part of evaluating the consequences from dispersed fragments and the impact of FFRD on the source term. The simplest analysis one can conduct is to compare the terminal velocity for the fragment size distribution with the upward gas velocity at burst to evaluate the mass fraction of fragments that are carried upwards and the mass fraction that settles. The analysis can be extended to assess the fraction of the size distribution that is carried upwards and that is likely to traverse scenario-specific obstacles to reach containment.

The nominal fragment size distributions/flow patterns are such that it seems unlikely that fragments can collect in large quantities far away from the core, other than perhaps in the lower head, since fuel fragments are very large and dense relative to aerosols. Complicated flow channels tend to separate out different particle sizes and dispersing the mass rather than collecting in a single location.

[Phillips, 2015] evaluated the mobility of fragments for a DB LOCA scenario. In this work, SNL and NRC characterized flow velocity during a large break LOCA using TRACE and characterized the critical fragment size below which fragments would be initially carried upwards following release and thus possibly transported downstream. The analysis concluded that, for this LBLOCA scenario, about 1/2 of fragment mass of the considered size distribution would be initially swept upwards. For other scenarios the gas flow rate, critical sizes, and thus the mass fraction (of the same assumed distribution) initially transported upwards would all be lower.

The terminal velocity of fragments under different conditions has been evaluated. The scenario gas velocities were not obtained so the assessment of the fragment mass fraction carried upward was not evaluated.

Idealized spherical fragment terminal velocities have been evaluated for different density particles in air at STP using the Brown correlation are shown in Figure 6. Particle densities representing water (corresponds to aerodynamic diameter for small particles), sand, an intermediate range, and that of nominal UO_2 were used. The figure also shows nominal RCS aerosol size range and the typical reported fuel fragment size range. Nearly five orders of magnitude separate the settling velocities of nominal RCS aerosols (red, 1g/cc, curve on the low range) with that of fuel fragment particles (purple line on the high end of the size range). The difference in settling velocities are so great that behavior is qualitatively different. Fragments settle at high rates in situations where nominal RCS aerosols readily transport with flows. The concept of well-mixedness relates not only to flow velocities but also to the particle settling velocity. The influence of other factors such as recirculation or the effect of turbulence act relative to the particle settling velocity.

Fragments with irregular shapes typically settle slower than that of the volume equivalent sphere. The effect is roughly bounded and the effect is unlikely to reduce settling velocity past that of the unit density particle (red curve) which effectively describes the “most transportable”, and thus conservative, assumption one can make with regards to fragments reaching containment.

Figure 7 shows a similar curve but only for UO2 and water particles densities. This curve shows three gases that roughly span the conditions in reactor operation and accidents: air at STP (same as previous plot), saturated steam at 1bar (approximately 1 atmosphere), and saturated steam at 16MPa (approximately SRV limit for PWRs). Note that the evaluation of terminal velocities did not include the critical transition. The UO_2 particle density curve exceeded the critical Reynolds number (~2×10^5; see Figure 8) for particle diameter near 1cm. Accounting for this transition would result in a step increase in settling velocity.

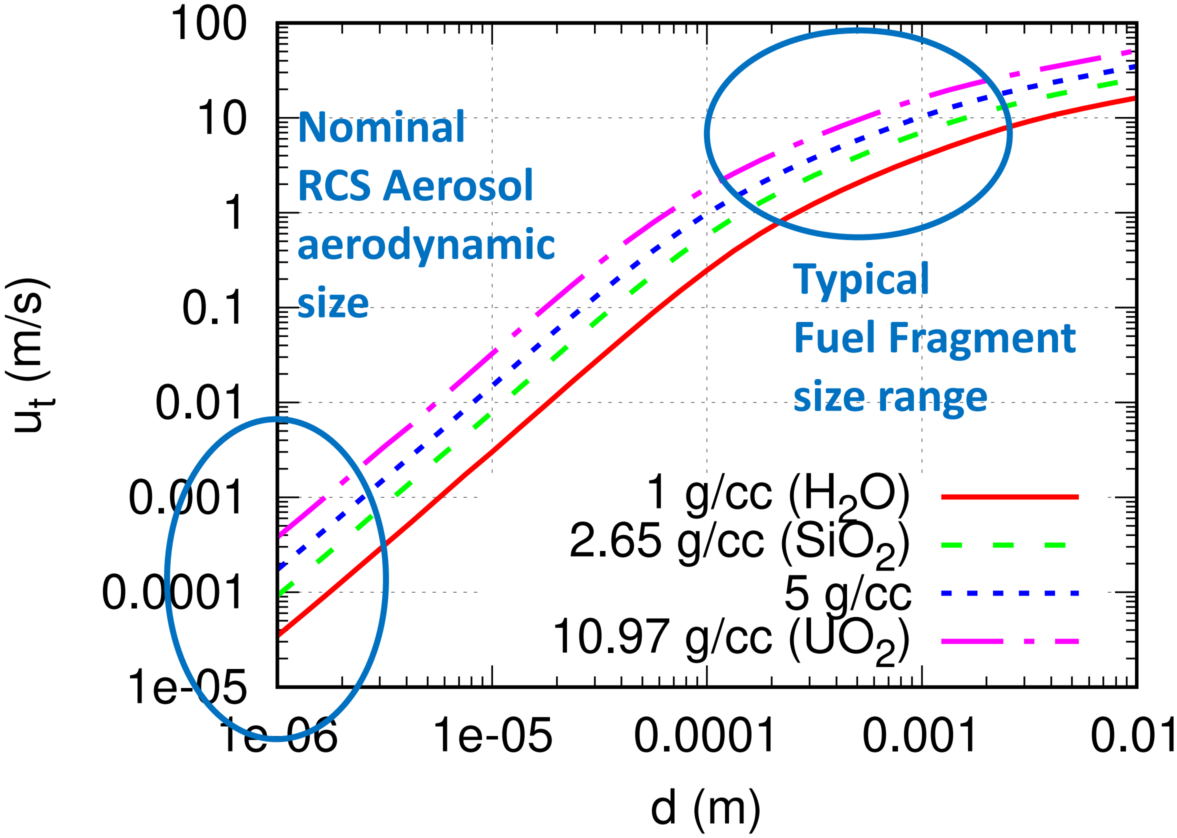


Figure 6 Settling rates of different density spheres in air at STP

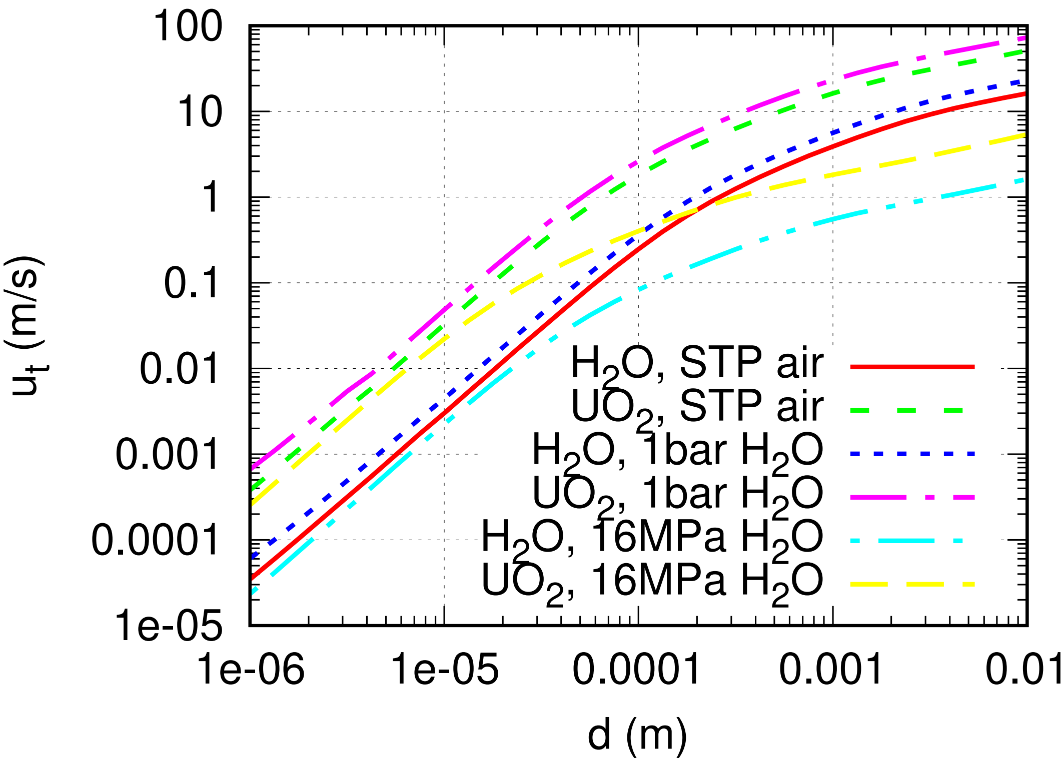


Figure 7 Settling rates of 1 g/cc (H2O) and 10.97 g/cc (UO2) particles in air at STP and in saturated steam at 1 bar and 16 MPa

These velocities were determined using an approximation of the drag curve and solving for the velocity at which the drag force matches that of gravity. The drag curve from three different sources, [Clift, 1978], [Haider, 1989], and [Brown, 2003], is shown in Figure 8.

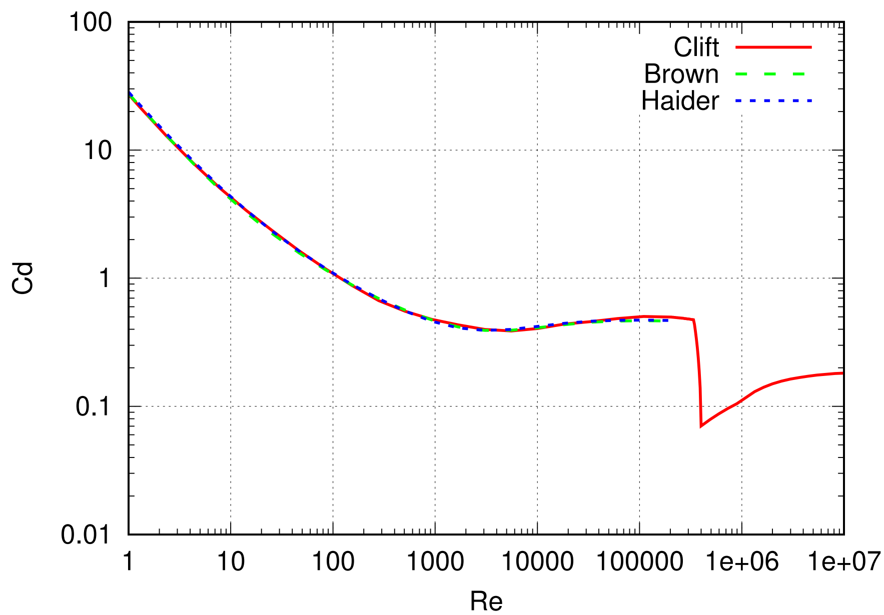


Figure 8 Nominal Sphere drag curve

Fragment heat up

Another crucial element of assessing the impact of FFRD on the source terms is evaluating the fragment temperature which is the primary factor that governs whether radionuclides can escape the fragments.

Experiments indicate that fuel that does not get hot (>~2200F,~1200C,~1500K) does not appreciably release radionuclides. This is the case whether this fuel is in a fuel rod or in a fuel fragment that has moved elsewhere.

Note that the fuel in some of the VERCORS and VERDON experiments fragmented. Oxidation of fuel does this. Some of these data reflect fragmentation conditions although the initial sample preparation (depressurizing and cutting of irradiated fuel into a sample size) avoided conditions (sudden, large ∆P) that results in the fuel fragmentation of interest in this report.

Radionuclide release from fragments

To characterize releases from fuel one could consider the model above while tracking the temperature of both intact and dispersed fuel. Although the simple model above does not account for differences in grain size, burnup/specific decay power, and gas exposure, it should give an idea of relative behavior of releases of intact fuel and fuel fragments.

A simpler question that illustrates the potential for enhanced releases is:

What fraction of fuel exceeds 1500K under intact and FFRD conditions?

This is a suitable surrogate that can indicate relative releases since fuel that does not exceed this temperature will not appreciably release radionuclides during an accident.

Evaluating the fuel temperature is a heat transfer problem. This involves consideration of the different heat sources and sinks for fuel in different configurations.

The heat sources comprise decay power in the fuel and chemical heat from clad oxidation. At low temperatures, decay heat dominates, while at high temperature clad oxidation can contribute most of the power. Fuel temperatures typically do not exceed 1500K and substantial radionuclide releases do not occur until following the heat addition by the clad oxidation event.

The heat sinks are heat transfer to other fuel, structures, fluids, containment, and the environment. The heat transfer mechanisms are conduction, convection, and thermal radiation. The conduction rate is proportional to the temperature gradient in a solid along with material properties. Convection is proportional to the temperature difference between the solid surface in addition to fluid effects. Thermal radiation is proportional to the difference of temperatures to the 4^{th} power. At temperatures near 1500K the radiative heat transfer to far cooler temperatures is tremendous. A crucial element of the effectiveness of radiative heat transfer is the temperature distribution over the solid fraction that the surface "sees" which is characterized by the term "*view factor*". The view factor or solid angle fraction to cooler surfaces governs the radiative heat loss rate.

Fuel Fragment Release and Dispersal from fuel rods

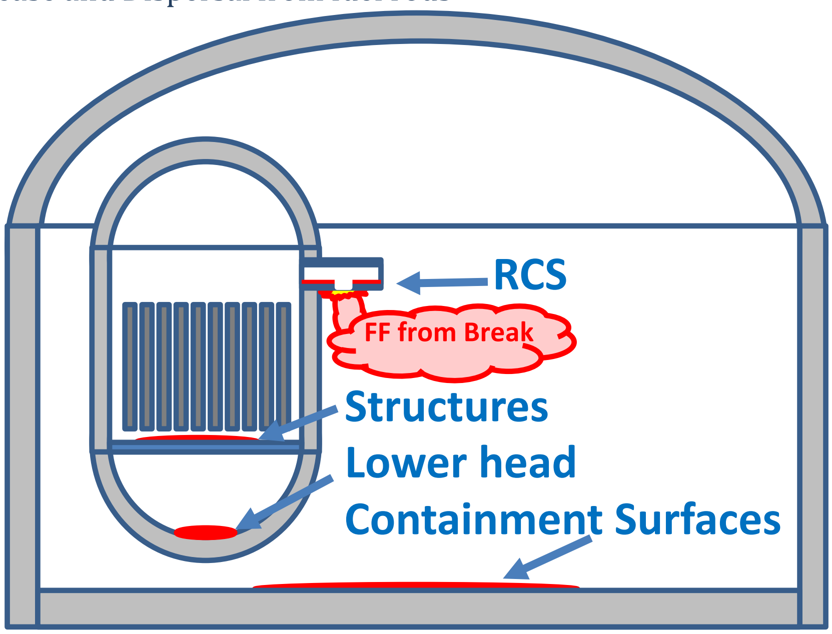


Figure 9 Considered Fuel Fragment collection locations

A small fraction of high burnup fuel is expected to be released from rods as fragments. Nominal upper bound of about 2% of fuel is assumed to fragment and disperse.

Depending on an individual fragment's size, shape, and the fluid velocity for the scenario at the time of rupture it can either settle or be carried upwards. For a size distribution of fragments and fluid velocity, some will be carried upwards and some will settle.

The considered fragment deposition locations are shown in Figure 9. Fragments that are carried upwards can follow the flow but may settle anywhere along the flow path, whether back in the reactor, in reactor coolant system piping, or in containment if fragments leave RCS through a break. Fragments that fall downwards can settle on structures in the vessel or on the lower head. Given that the sizes of fragments are substantially smaller than a fuel pellet, on the order of a millimeter or less (perhaps similar to sand grain size distribution), plugging of flow channels is not considered likely.

Comparison of fuel temperatures in FFRD and intact conditions

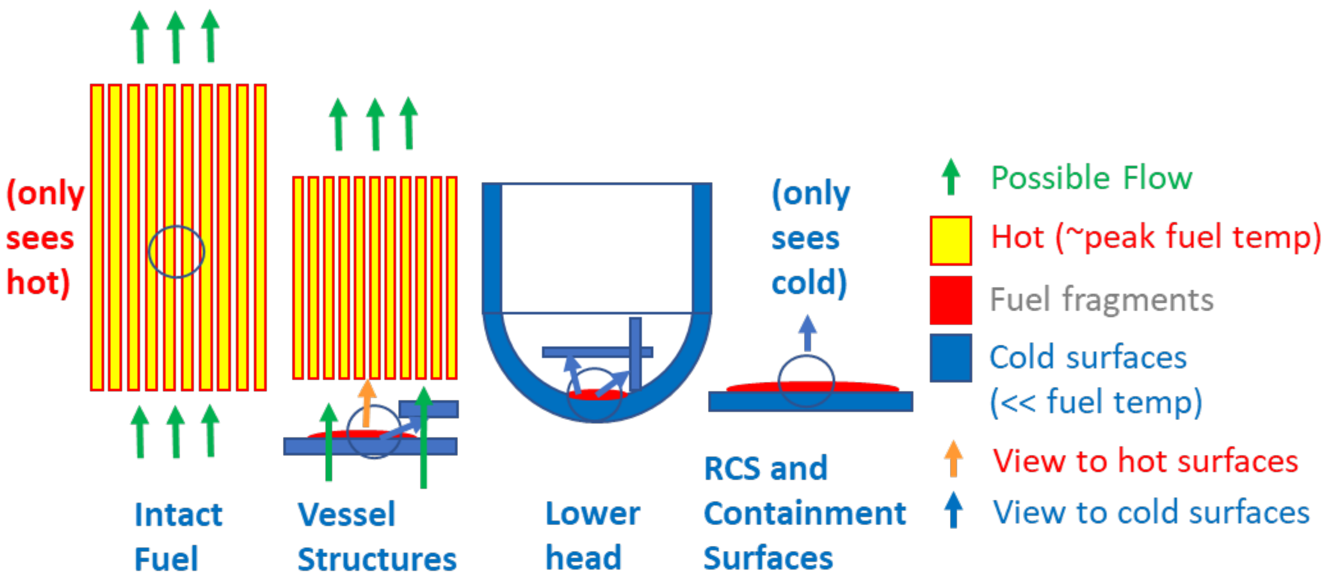


Figure 10 Heat transfer for different fuel locations

The fuel temperatures are affected by the combination of heat sources and sinks. Figure 10 depicts geometry that affect the mechanisms of heat transfer (conduction, convection, and thermal radiation). It shows hot surfaces (near peak fuel temperature and cold surfaces (substantially colder than peak fuel temperatures) that fuel is in contact with (conduction), whether substantial flows may be possible in certain scenarios (convection), and whether the fuel rod (intact) or fuel fragment surface "sees" hot or cold surfaces.

A fuel grain in fuel in an intact rod conducts radially to the rod surface. Axial conduction is typically neglected because the axial temperature gradient is far less than the radial gradient within a fuel rod. At the highest temperatures at the center of the core radiation is not efficient because hot fuel surfaces primarily see mostly hot fuel surfaces (view factor near 1) or perhaps some structural or control rod surfaces that also soon heat up to similar temperatures. Hot surfaces that only see other hot surfaces at nearly the same temperature are not appreciably cooled by thermal radiation. When decay power raises the clad temperature sufficiently high, clad oxidation contributes most of the power for a period, raising the temperature of the intact fuel faster.

Fuel fragments are exposed to a markedly different environment. Since they are distant from the clad, they are not directly exposed to the heat generated by clad oxidation. Fuel fragments that settle in the RCS, containment, and initially the lower head are in direct contact with colder surfaces (conduction) and see only cold surfaces (radiation). With the T^4 dependence for radiation heat transfer, decay heat will not raise the fragment temperature until the surrounding structures are heated unless a sufficiently thick debris bed layer develops.

Fuel fragments that settle on lower vessel structures see, not only cold structures, but also may see some of the lower regions of the fuel. These regions, though hot, are typically somewhat cooler than the center of the core. These fragments may be indirectly exposed to clad oxidation heat by thermal radiation from heated fuel. Flow that passes through the core also passes through some of the lower structures, before passing through the core, and may further cool these structures and the fuel fragments settled on them.

Even fuel fragments settled on the lower vessel structures would remain substantially cooler than fuel remaining in core, at least until fuel relocation to the region where fuel fragments have settled. Fuel fragments settled on lower vessel structures may remain cool if relocation to the lower head bypasses their location.

The depth of fuel fragment debris also affects its temperature. The lower head is the fragment accumulation location that is considered most likely to potentially result in accumulation of fragments. A single layer of submillimeter fragments with nominal specific decay power that see only cool surfaces (in the range of saturation temperature at system pressure) should remain relatively cool. Interior regions of a thicker fragment layer with the same specific decay power can increase in temperature. However, it seems unlikely the debris layer would be thick enough to heat up substantially, at least based on an assumed 2% dispersed fuel fraction. Unless the fragment fraction is substantially greater than 2%, there does not seem to exist a mechanism for substantial release of radionuclides from dispersed fuel fragments. Even for this situation, the heat up and radionuclide release from both the fragments in the lower head and in the remaining in-core fuel should remain less than and slower than that from initially intact (non-fragmented) fuel due to the dispersal of the heat source and exposure to cooler surfaces. Models for the ability to cool heat-generating debris beds already have been developed and implemented in MELCOR [Lipinski,1982].

Fragment temperature estimate

As discussed above, the heat up of fuel fragments in a few different configurations is of interest for FFRD scenarios since the fragment temperature governs radionuclide releases. These configurations of interest are:

- Falling at terminal velocity (currently evaluated for air only)
    - convection
    - convection + radiation
- On surfaces – (didn't evaluate)
    - conduction
    - conduction + radiation
- Debris beds
    - Wet – found literature (Lipinski, 1982) (debris bed depths less than 9cm likely to be wetted)
    - Dry – didn't find sufficient information

This report has only considered convective heat transfer quantitatively; other forms of heat transfer have been discussed qualitatively. The assessment used drag and velocity relations determined from the previous terminal velocity calculations. In combination with the evaluation of decay heat, the analyses of convection alone were sufficient to determine that the fragment temperature is limited by the temperature of its surroundings.

As with terminal velocity, the geometry was idealized by assuming spheres. The temperature distribution in spheres was evaluated. Another interesting parameter is the response time to sudden changes in temperature. Evaluating response time was planned but there was insufficient time to do so.

The evaluations were conducted with a nominal fragment power density of 500 W/cm^3 and nominal UO_2 properties. This power density was estimated from the loading figures in [Phillips, 2015]. Table 10shows these derived power densities.

Table 10 End-Of-Cycle Fragment Power Density Estimated using Phillips,2015 loading and nominal pellet d, UO2 ρ

		LHGR		PowDens	SpecPow
cycle	bundles	kW/ft	kW/m	kW/m^3	kW/kg
1	84	8	26.2	4.97E+05	45.3
1		7	23.0	4.35E+05	39.6
2	60	6.6	21.7	4.10E+05	37.4
2		6	19.7	3.73E+05	34.0
2	24	3.5	11.5	2.17E+05	19.8
3	1	6	19.7	3.73E+05	34.0
3	8	3.4	11.2	2.11E+05	19.3
3	16	2.1	6.9	1.30E+05	11.9

This power density is about half that achievable for fuel. The power density for high burnup fuel is generally substantially lower than the maximum so this may be about right for the power density before shutdown.

A with nominal heat transfer coefficient from [Clift,1978] of 500 W/m^{2}K was initially assumed for this analysis. This was later extended by using air correlations. Generalized correlation should be used for other gases. The air correlation can be used to check other general implemented correlations in the future.

Evaluated sphere T(d) with nominal h (500 W/m^{2}K) and high power 500 W/cm^3 (5E8 W/m^3) (corresponds to about 8 kW/ft (26 kW/m, 2.6 W/cm)). At full power (no decay) sphere temperatures can be high when considering only convection.

Figure 11 shows the temperature distribution in a 1mm sphere w 500 W/cm^3 power density and 500 W/m^{2}K heat transfer coefficient. The temperature difference between centerline and fragment surface temperature was about 2 ^{o}C. Figure 12 shows fragment surface T vs diameter in air for the same assumed power density and assumed heat transfer coefficient.

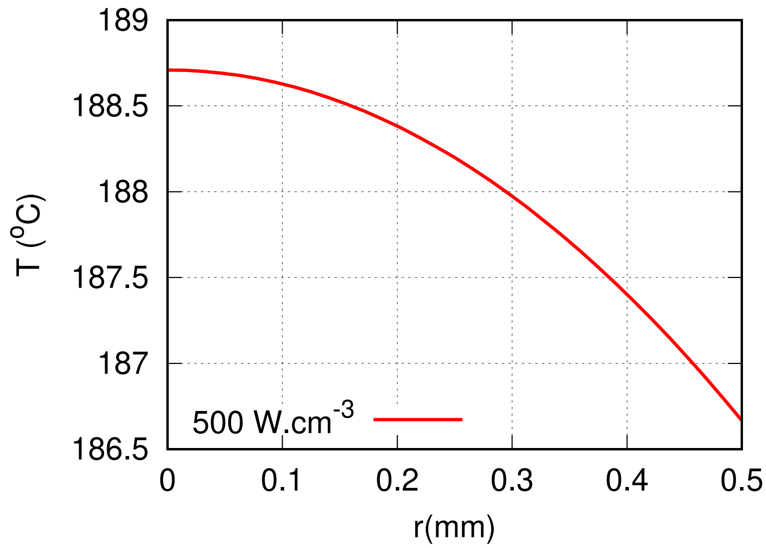


Figure 11 Temperature distribution in 1mm sphere w 500 W/cm³ power density 500 W/m²K

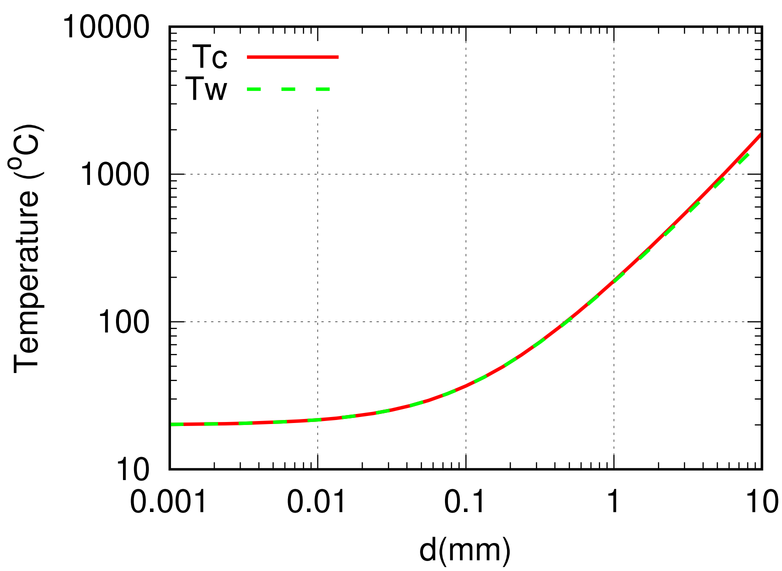


Figure 12 Fragment surface temperature vs diameter in air

Figure 13 shows the air Nusselt Number correlation recommended by Clift, 1978. Figure 14 shows the particle heat transfer coefficients evaluated using the air Nusselt number correlation and the Reynolds numbers calculated in the evaluation of the terminal velocities. This plot matches the Figure in Clift quite well.

It would be good to use the general correlations (Table 5.4 equations C-F) that would also be applicable to steam conditions. However, since the expected temperatures are so low it may not worth the effort to implement. The air correlation provides the general range of heat transfer coefficient that may be experienced.

Sphere temperatures were also evaluated using the Air Nusselt Correlation for different decay powers. Even with the +2sigma decay, peak centerline sphere temperatures of 1cm diameter were less than 200

C higher than ambient within a second of shutdown considering convection heat transfer alone. The peak temperatures are less than 10K higher than ambient for spheres less than a millimeter even a second after shutdown.

Figure 15 shows Peak T of spherical fragments in 20 ^{o}C air, assuming full power. For most scenarios, the reactor will have been shutdown for hours before rods burst and fuel fragments are expelled into the fluid so the fragments will have only a small fraction of this power density. By the time many releases start a few hours later, even peak centerline temperatures for 1mm spheres were only a few 10s of degrees K higher than ambient even when considering convection heat transfer alone.

Insufficient time was available to evaluate particle response time to perturbations. Very quick response times are expected.

Information was sought for heat up of submerged and dry debris beds. The Lipinski Dryout heat flux report can be used to estimate behavior of submerged particles. (Lipinski, 1982). Insufficient time was available to find information about the temperature of heated dry debris beds.

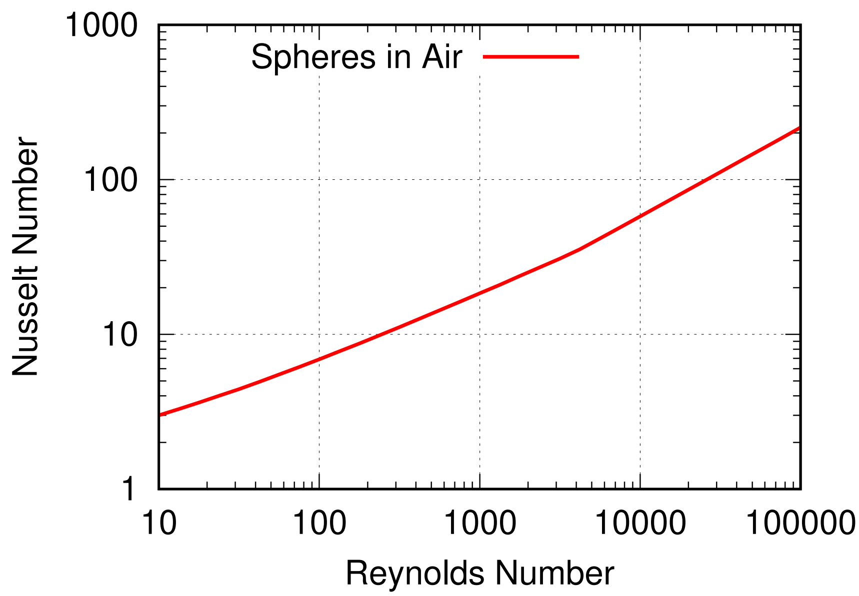


Figure 13 Sphere Nusselt number correlation for air, from Clift, 1978

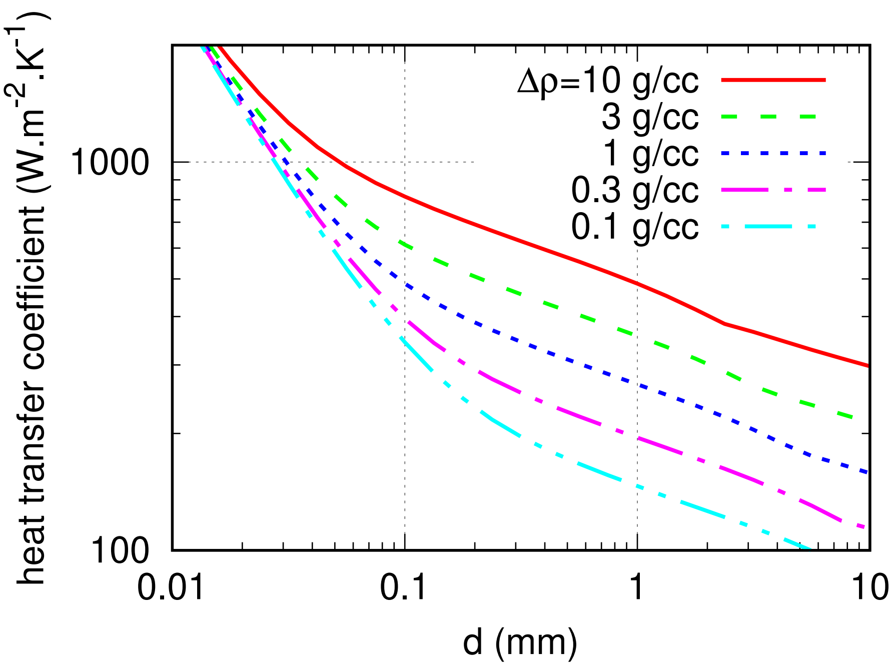


Figure 14 Convection heat transfer coefficient for particles settling at terminal velocity in air

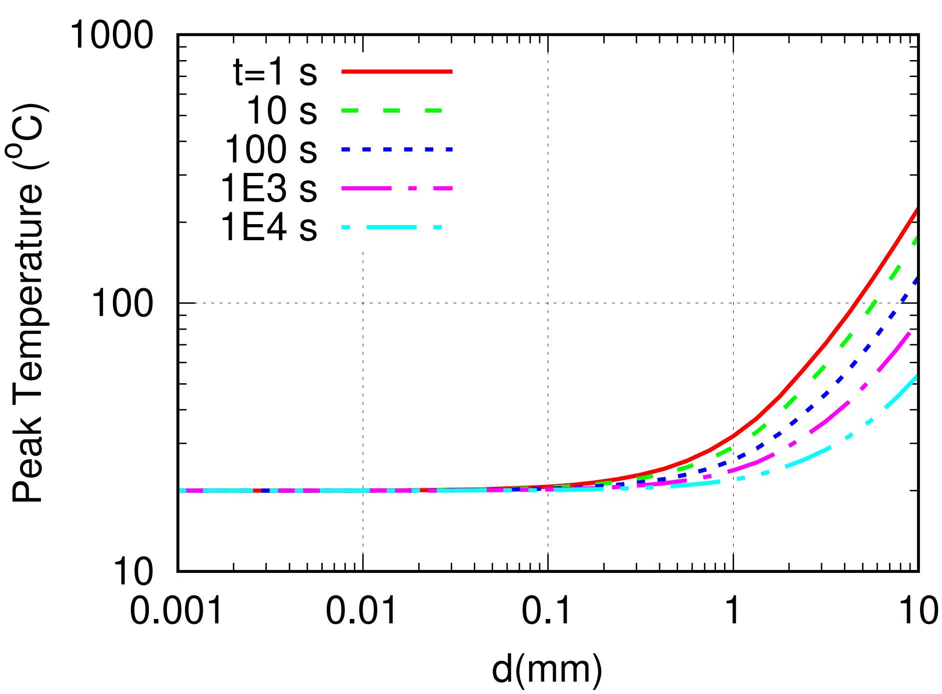


Figure 15 Peak centerline T of spherical fragments in 20 °C air, full power (before shutdown)

Figure 16 shows the initial decay curve used to estimate the decrease in power following shutdown. These curves originate from tabulated values in Research Information Letter 0202 [USNRC, 2002], Attachment 1, Appendix K Decay Heat Standards. The recommended decay power curve in the Standard Review Plan may better represent behavior for extended times [USNRC,1981b]. Figure 17 compares the decay curves from APCSB 9.2 to those in RIL 0202 over extended durations. The APCSB relation in the Standard Review Plan for an 18-month operating time, even when multiplied by 1.1, falls below the mean for the replacement ANS94 mean. The decay heat fit used in the ESF fragmentation analysis, [Metcalf, 2021], returns a higher (conservative) decay heat than the accepted +2σ ANS94 curve from 10s throughout the compared time period.

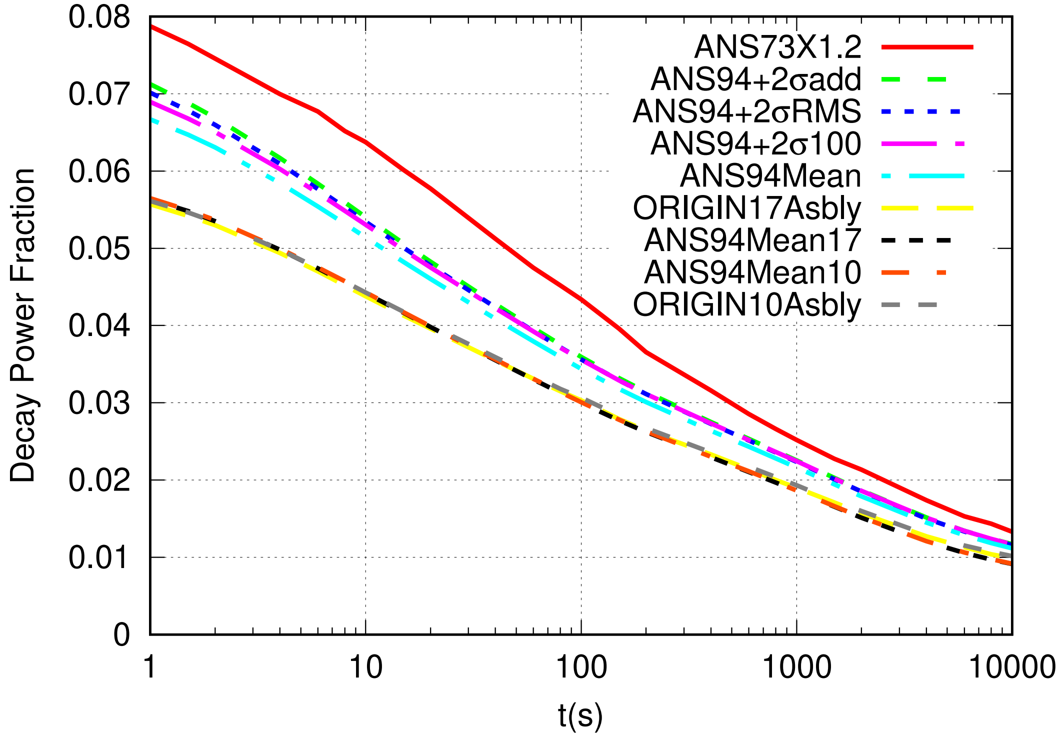


Figure 16 Different decay curves used in Appendix K

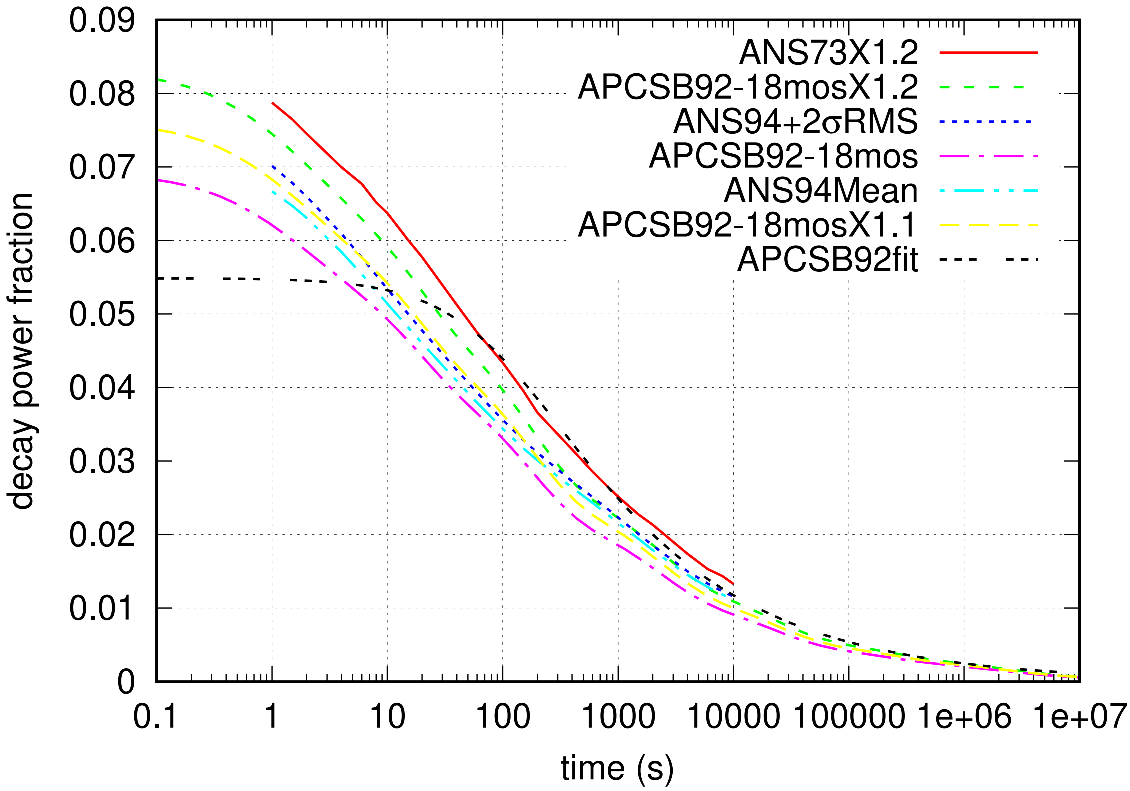


Figure 17 Decay heat curve comparison

Fragment RN release

Figure 18 shows the MELCOR model for radionuclide release from high burnup fuel based on the VERCORS experiments. Although other parameters such as grain sizes, gas composition, and transport in open porosity, can factor into release rates, in-grain diffusion is the limiting phenomena and modeling the behavior assuming a fixed grain size and accounting for the change in diffusivity with temperature should capture major trends in release behavior reasonably well. This curve could be used to estimate releases for fragments that heat up. A more involved assessment of the effects of FFRD could also consider gas composition.

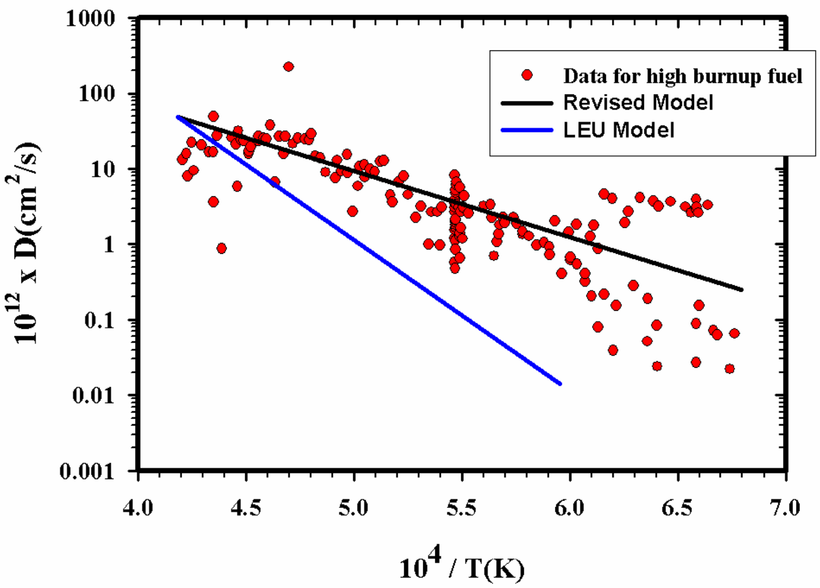


Figure 18 Cs release from high burnup fuel compared to results of VERCORS test, from [Gauntt, 2010]

FFRD Impact on timing

Fuel fragmentation could affect the accident progression. Some scenarios involve water in the lower head for some time. Fuel fragments that are submerged in water would be cooled initially to saturation temperature. If a small fraction of the fuel (~2%) fragments and is dispersed it would be difficult to justify a substantial change in source term behavior. If, on the other hand 20% of the fuel fragments and reaches the lower head the more efficient heat transfer would result in earlier boiloff of the water remaining in the lower head. This would also involve a more dispersed heat source which is generally easier to cool. Assuming 20% fuel fragmentation and dispersal leaves only 80% of the fuel in the core. One could expect that this fuel loss would be greatest near the center of the core where the fuel is more stressed. Substantial local fuel loss would result in lower temperatures in that region relative to intact fuel, not only from the lower localized decay heat, but by potentially providing a cooler region that other parts of the core can radiate to. Depending on the reactor design, scenario, and amount and location of assumed for fuel loss through fragmentation, it does not seem inconceivable that both local and overall clad oxidation would be delayed and perhaps even locally prevented, reducing the overall heat added to the core. These effects would both delay releases and reduce overall release magnitude.

Substantial fragmentation could affect accident progression in some cases. If a large fraction of fuel fragments and settles during a PWR station blackout scenario the additional heat to water in the lower head would enhance steam generation which results in transporting more heat from the core to the hot leg resulting in earlier hot leg creep failure. Fragments that get transported to the HL could release their decay heat in the vicinity of the hot leg itself. The HL failure would not occur in other scenarios in the development of the source term that postulate RCS failure as an initiating event. It seems that more than 2% fuel fragmentation would be necessary to significantly affect lower head boiloff timing and that the change in timing from 2% fuel fragmentation would not be significant. In other words, 2% fragmentation is not enough to significantly affect timing

Because of advances in SA heat transfer modeling, accident progression in current MELCOR calculations is predicted to be substantially slower than in NUREG-1465 (current RG1.183 Table 4). The difference in timing from this change is much larger than that from FFRD.

Scoping calculations in MAAP-MELCOR crosswalk [Luxat, 2014] indicate that fragments may enhance core-to hot leg heat transfer, accelerating its rupture and reducing likelihood of induced SGTRs. To explore the effect of fragments on hot leg rupture, MELCOR calculations would be needed.

Leaching

The leaching of radionuclides from submersed fragments may be an issue in the long term. Although the time scales upon which leaching acts are typically far longer that the considered duration of the design basis ST, smaller fragments have a substantially larger surface to volume ratio compared to nominal debris, so they could leach substantially faster.

The fragmentation and dispersal fraction of 2% limits the extent of leaching to less than this value since the radionuclides that would leach from fragments are the same radionuclides that would have been released from this same fuel had it remained in the core. This amount is not significant compared to other releases.

Criticality

Criticality of fragments has been listed as a concern. The paper/presentation referenced [Nakajima, 2015] indicated that criticality concerns existed primarily for fragments of fuel with burnups that don't fragment. Therefore, the criticality effect is not considered to be related to FFRD. Fuel designed for HBU will likely have higher enrichment than existing fuel designs, so it has enhanced risk of criticality when at low burnup. However, fuel at low burnup is not susceptible to fine fragmentation and dispersal.

Some additional comments of the possibility of ex-vessel criticality follow.

Radionuclide inventory generation depends on integrated time at power. The fuel sees full operating power for years building up radionuclides. Most of the volatile radionuclides have been cooked off before the fuel relocates to an ex-vessel debris bed. Ex-vessel debris can't have high critical power and thus can't build up appreciable additional radionuclides.

High system pressure and high saturation temperatures enable high power densities during operation. Another crucial required element for criticality is moderation. Ex vessel debris is not at high pressure. Because of this, once debris gets hotter than sat T, the debris loses water to evaporation and thus loses moderation. This results in a loss of power other than decay power. Low-power, low-temperature, cyclical criticality is expected if criticality can be achieved at all: water ingress, criticality, heat up, steam formation, loss of power, cooling, repeat. The temperature is limited to approximately saturation T, ~100 ^{o}C, by loss of moderator.

It seems difficult for ex-vessel criticality to contribute significant heat relative to decay heat. Without high power, ex-vessel criticality can't generate a significant quantity of RNs compared to the initial inventory at the time of the accident. Without high power, ex-vessel criticality can't generate enough heat to raise temperature sufficiently to release radionuclides in the debris. At temperatures sufficiently high to release radionuclides, liquid water cannot be present (RN Tvolatization >> Tsat water ~1atm). Therefore, at these temperatures at which RNs can be released from debris, moderation is not available, and fission is not contributing significantly to power.

If debris heats enough to release RNs, it will be due to decay heat. If enough heat is added from radioactive decay, the debris may melt, losing porosity, and thus losing the possibility of being moderated.

If that occurs there is no moderation, no criticality, no heat from fission, no additional RNs generated from fission, and thus no ex-vessel criticality source term.

Sensitivities: “What if”

A few sensitivities were considered. Some that were discussed here are the impact of higher fragmentation fraction than expected. Another is the possibility of dryout of fragment-containing ESF leakage.

Greater than expected FFRD

If current experiments and models capture behavior, an assumption 2% fragmentation is likely bounding. An independent and seemingly bounding evaluation of the FFRD fraction conducted as part of this review returned a lower FFRD fraction than the 2%. It would be good to repeat the analyses for possible BWR loadings. If substantially different fuel loadings than the one analyzed exist, then it may be useful to also check the FFRD fraction for those loadings.

The current expectation is that dispersed fuel would not heat up enough to release NGs or volatiles. The containment ST would proportionately decrease with the fragment fraction, except perhaps for some noble gases during the gap release phase of the accident.

Eventually, with enough FFRD fraction, fragments could accumulate in a sufficiently large debris bed, heat up, and release their volatile RNs. The same volatiles would be released from the fragments as would be released from the fuel if it had remained in the core. Debris beds result as part of the normal severe accident core degradation processes.

Most fragment mass is likely trapped somewhere in the vessel or elsewhere in the RCS, not in containment.

It is expected that release fractions from fragment debris beds would be similar to the nominal Containment Source Term if somehow a large (~0.2) fraction of the core fragments, since most of the volatiles would be released from the fuel in either situation. However, the release timing would likely be different between these two scenarios.

ESF Leakage – Impact if fragments escape

One specific concern that staff raised is that fragments might make it to the containment sumps, get pumped through ECCS piping and collect in a low point, and then heat up to the extent that they release radionuclides which then can be released through design basis ESF leakage to the auxiliary building, bypassing containment. Containment bypass scenarios typically result in high consequences.

J. Metcalf estimated the possibility of dryout from leaked ESF water containing fragments [Metcalf, 2021].

The assumptions in this analysis were reviewed. Some of the notable assumptions were: 2% of the fuel was assumed to be fragmented. All the fragments were assumed to end up in ESF water at 100 ^oC. The fragments were concentrated in the water. The fragments did not settle on surfaces in the water, so all fragments leaked from ESF system. The leak did not plug. Condensation to replenish puddle was neglected.

The assumptions in the analysis were generally found to be quite conservative. One exception was the assumed water temperature. At depth and in a pressurized containment, water temperatures can exceed 100 ^oC. Leaking water at higher than 100 ^oC would result in a higher fraction vaporized. Nevertheless, the analysis on the whole is considered to be conservative. The conclusion that dryout is unlikely seems reasonable.

A similar dryout analysis was independently conducted both numerically and analytically using SI units. An equation of the same form as Metcalf's was derived. The trends with the fragment concentration multiplication factor and with leakage flow rate in both the numerical and analytical evaluations matched those in Metcalf's analysis: lower leak rates and higher fragment concentrations increased the likelihood of dryout. Dryout of leaked ESF water is far less likely if less than the assumed 2% fuel reaches the coolant or leaks along with coolant.

The review can be found in **Error! Reference source not found.**.

Expected FFRD Impact on the Containment Source Term

Based on the findings of the review and scoping calculations an estimate of the Expected FFRD impact on the source term was made. The review built upon considerable previous work on Severe Accidents, Source Term, and FFRD, and on insights from the recent ATF-PIRT[ERI, 2021]. The assessed FFRD impact on the containment source was evaluated in a similar manner as for the limiting case but included the expected effects of the different elements (FFRD fraction, fragment size distribution, transport, heat up, and RN release).

In the evaluation of the expected FFRD Impact on the Containment Source Term the bounding FFRD fraction of 2% was assumed despite expectations that actual FFRD fraction may be around 0.5% (or less). A lower FFRD fraction would result approximately in a proportionately lower difference between the base and FFRD-adjusted source terms.

If one considers expected fragment dispersal, transport, heat up, and RN release, fragments don't contribute to the Containment Source Term since they don't heat up.

- Data and bounding analyses show small FFRD fraction
- Most fragment mass doesn't reach containment
- Fragments likely disperse and not collect to debris beds except perhaps in the lower head
- There is not enough power in dispersed fragments to heat up the surrounding environment, so they don't heat up and don't release radionuclides to their surroundings.

Fragments in hot regions exposed to hot gases emanating from core heat up but remain cooler than the core. Steel melts at temperatures in the range 1400-1500 ^{o}C so RCS structural failure and release of loose fragments to containment are likely before RNs can appreciably release from fragments held up in RCS.

Deviations from total release to containment (RF=1) for a radionuclide can result from incomplete release from an individual fuel pellet, from disparity in release fraction across the core, and from decontamination en route to containment. Without evaluating the localized radionuclide release and transport from individual severe accident scenarios with MELCOR, the relative influence of these effects is conservatively assumed even if the results don't seem plausible.

The impact that FFRD has on the Containment Source Term depends on the fraction of volatile radionuclides that would have been released from the fragmented fuel if it had remained in the fuel and not fragmented.

- Not all regions of the core release radionuclides similarly. Fuel in peripheral regions of the core do not heat up as much and therefore release a lower fraction of their radionuclides.
- Some volatile radionuclides deposit on surfaces en route to containment
- The non-uniform volatile release and RN retention en route to containment drive the numbers in the source term.
- Other radionuclides are less volatile, or their volatility depends on local gas conditions.
- On the expectation that fragmentation occurs in regions where volatile radionuclide release from fuel would otherwise have been nearly complete, and neglecting decontamination on transport to containment would result in a direct reduction in the Containment Source Term equal to the fraction of core that is fragmented (~2%) (RFadjusted = RF – FFRD)
- If, on the other hand, it is assumed (to facilitate the evaluation – does not represent actual behavior) that the entire core releases volatiles equally in a high fraction (~1) and the decrease from unity results from decontamination in the RCS, the corresponding RF adjustment would be (RFadjusted = RF – FFRD*RF)
- Actual behavior lies somewhere between these two limits.

Noble gases were treated differently because they do not deposit on surfaces and to account for the bounding early NG release for long-lived ^{85}Kr in experiments.

Overall, the NG release when accounting for fragmentation is proportionately lower than that of the Containment Source Term by the FFRD fraction (2%) because fragments that don't heat up retain NG's that would have been released had the fuel stayed in the core. The NGs could be released from fragments that settle once vessel dries out and the rest of the core debris relocates to the lower head and heats the fragments. For the purposes of evaluating the early gap release it was assumed that all fuel fragments had a burnup of 72GWd/t (bounding early NG fraction of 0.2). The early NG fraction was evaluated as the 0.2 of the 0.02 = 0.004. This was assigned to gap release even though MELCOR calcs indicate that most gap releases are likely to reach containment during In-Vessel phase as the transition was defined. The 2011 Containment Source Term Peer Review Committee recommended changing the criteria for source term phase transitions to be more meaningful and to better represent MELCOR results [ERI, 2011].

Note that an early NG fraction should probably be updated for all HBU fuel, but not the bounding value assumed in this report. This is to say that up to 0.2 long-lived NG release could apply to all fuel (less for shorter-lived NGs). This was not implemented here because doing so does not factor into the assessment of the change in Containment Source Term from FFRD and would dilute the findings regarding the effect. MELCOR would have to be run to evaluate the effect of overall early NG. Based on this review, only small effects on release timing are considered likely.

Although the early NG release from fragments is minimal, it will be even less for shorter-lived RN isotopes that preferentially reside within grains. Therefore, the current values are considered to be bounding.

Table 11 and Table 12 show the expected adjustments to the Containment Source Term from FFRD phenomena for PWRs and BWRs, respectively. The reduction in tables is based on the assumption of maximum RCS decontamination and thus minimal decrease in ST from FFRD effects. Assuming low importance of RCS decontamination and total localized release results in an across the board reduction of 2% for NG, I, and Cs.

Note: These tables currently reflect general expectations on median values rather the aggregated results of sequence-specific assessment of FFRD influence on source terms.

Table 11 PWR Gap and In-Vessel Release Fractions, modifications to SAND2011-0128 (parentheticals correspond to 2% FFRD adjustment)

PWR	Gap Release	In Vessel Release
Noble Gases (Kr,Xe)	**0.017** (0.021)	**0.94** (0.92)
Halogens (Br,I)	**0.004** (0.004)	**0.37** (0.362)
Alkali Metals (Rb, Cs)	**0.003** (0.003)	**0.23** (0.225)

Table 12 BWR Gap and In-Vessel Release Fractions, modifications to SAND2011-0128 (parentheticals correspond to 2% FFRD adjustment)

BWR	Gap Release	In-Vessel Release
Noble Gases (Kr,Xe)	**0.008** (0.012)	**0.96** (0.94)
Halogens (Br,I)	**0.002** (0.002)	**0.47** (0.461)
Alkali Metals (Rb, Cs)	**0.002** (0.002)	**0.13** (0.127)

If the fragmented fuel had instead remained in vessel, as most are expected to do, and if they heat up to the extent that they can release radionuclides (which is not expected unless the debris bed dries out and/or the rest of core debris collapses on it), the decontamination en route to containment would be similar. In this case, the source term values in the tables would approach the original values. Based on this assessment, FFRD does not seem to greatly impact the ST. The unadjusted Containment Source Term appears conservative with respect to accounting for FFRD phenomena.

Limitations

The following limitations were identified for this analysis:

- Insufficient time was available to combine the effects of the various calculations into an overall effect. Thus, the conclusions reflect trends and expected results if more time had been available to combine and synthesize sequence-specific effects on the source term.
- Because of the limited resources and time available for this project, the assessed impact on ST and the values in tables currently reflect general expectations applied to the median values rather the aggregated results of sequence-specific assessment of FFRD influence on source terms.
- The fragmentation fraction was not evaluated for BWRs.

- The analysis involved one significant assumption that was not explicitly verified: “Flow velocities for other scenarios are expected to be too low to substantially loft a significant fraction of expelled fragments”. This expectation was based on recollection of results during the development of the SAND2011-0128 source term and in other MELCOR calculations in comparison to the terminal velocity curve. The intent was to directly compare gas velocities at clad burst as calculated by MELCOR to the size distribution and terminal velocity distribution to ensure this expectation is accurate and that conclusions about transport in this report accurately reflect the results of MELCOR analyses. Insufficient resources and time were available to verify this expectation. Even if this expectation is not accurate, the leakage pathways to containment for most scenarios are sufficiently tortuous that nearly all fragments will be trapped before reaching containment.

- The analysis of FFRD impact assumes that the observations and models for fragmentation (which have been derived from in-pile and hot cell tests on small fuel rod segments) adequately represent prototypic fragmentation behavior.

Some of these limitations can be eliminated with additional resources and time.

Applicability of RG1.183 endorsed codes for modeling FFRD

RG1.183 explicitly endorses ORIGEN and RADTRAD. The inclusion of STCP and MELCOR-generated ST tables in the RG implicitly endorses them. These tools were developed for scenarios where fuel was considered intact up to core collapse or fuel candling/melt. The applicability of these codes is assessed.

Based on this assessment, these codes are conservative as-is, without any modification to code or usage, when considering the effects of FFRD on the Containment Source Term. Recommendations are made for code usage or model improvement when applicable.

Given the limited time available for this request, these reviews constituted of high-level reasoning about potential impacts.

Applicability of the ORIGEN module in SCALE for modeling FFRD

ORIGEN is a SCALE module for isotopic depletion, decay, decay heat, and activation calculations. ORIGEN provides the RN inventory prior to fragmentation. Assuming that fragmentation and relocation don’t occur until ballooning, FFRD does not affect the code’s use. We consider that no modifications to ORIGEN usage or code are needed. Were fuel to move during operation this would likely have to be addressed in ORIGEN.

Applicability of RADTRAD for modeling FFRD

RADTRAD evaluates the depletion of RNs in containment, the impact of natural and engineered safety features, transport of RNs, and time-dependent dose to receptors. As long as the Containment Source Term source term retains the same form, no changes are expected for RADTRAD. If, on the other hand, analyses show that the Containment Source Term must be modified for FFRD, then changes may be needed. They would most likely be implemented as boundary conditions. (E.g. assume X RNs have leached into water, or Y additional RNs in location Z). Since the Containment Source Term is expected to retain the same form, no changes are expected for RADTRAD

Applicability of MELCOR for modeling FFRD

Given that FFRD phenomena appears to reduce the source term to containment, not including FFRD effects produces a conservative source term. Because of this, the MELCOR code is considered to be applicable for conservatively accounting for FFRD effects for the purpose of developing design basis Containment Source Terms to satisfy 10CFR100.11 and for updates to Regulatory Guide 1.183.

Some improvements to consider the impacts of FFRD could be made for best-estimate calculations. Two effects of note that could be improved upon are: 1) accelerated early NG release along with adjusted NG release within grains, and 2) enhanced heat transfer to RCS structures that could accelerate structural failure and transfer of radionuclides to containment. Although these effects are not considered necessary to adequately provide a conservative Containment Source Term for the purposes of assessing adequacy of ESFs and siting, they would be useful for best-estimate calculations.

Accelerated NG release from high-burnup fuel could potentially increase the consequences over that of low-burnup fuel. It does not seem likely that overall consequences (or even perhaps NG consequences) will increase when including FFRD effects due to the fact that FFRD likely reduces overall NG release due to cooler fragments not releasing their in-grain NGs that would have been released if this fuel remained in the core. It is recommended that the NG gap fraction be updated for high burnup fuel. The acceleration of releases occurs independent of FFRD – the same grain-boundary NGs that could be released by fragmentation are those released by temperature itself in the range of clad burst temperatures – so an FFRD-specific correlation isn't considered necessary. The updated NG gap fraction can be user input or be implemented as a default. Although a best-estimate NG was not evaluated, a bounding burnup-dependent NG fraction based on data on long-lived ^{85}Kr was developed. This could be used in MELCOR sensitivities to bound the effect. It would be good if a representative best-estimate gap can be developed. The increased gap fraction may be far less important for shorter-lived NGs that contribute more to dose. Further review and analysis are necessary to derive a representative estimate of the NG gap fraction.

One of the FFRD phenomena discussed during the recent ATF-PIRT [ERI,2021] was that parametric studies conducted as part of the MAAP-MELCOR crosswalk [Luxat,2014] indicated that fuel fragmentation effects could enhance heat transfer from the core to RCS structures. This effect is especially noteworthy in PWR station blackout scenarios: the enhanced heat transfer to the hot leg could accelerate hot leg rupture resulting in earlier release of radionuclides to the containment. Accelerated hot leg rupture has a beneficial effect in that it further reduces the likelihood of an induced steam generator tube rupture that could result in a high-consequence containment bypass event. It would be useful to further explore this behavior in MELCOR calculations. Such an activity was not within the scope or time limits of this work.

It may also be interesting to use a burnup-dependent and clad-type-dependent estimate of clad rupture. The used FFRD fraction conservatively (in terms of estimating FFRD fraction, the tendency for more FFRD was considered conservative) assumed rupture at high temperature, and thus higher ΔP and assumed pulverization at lower burnups. Experiments indicate that clad rupture at lower temperatures results in a transition to pulverization at higher burnup. A more representative accounting of burst temperature could be used to estimate a lower FFRD fraction based on these results.

It may be useful to explore these effects parametrically within MELCOR in scoping analyses using control function capability. Should any of these effects prove significant, they can potentially be incorporated in a more permanent basis within MELCOR.

Summary/Conclusions

The work described in this document describes the review of the impact of fragmentation on the design-basis Containment Source Term described in RG1.183 Tables 1, 2, and 4 and on subsequent analyses to update this source term for high-burnup fuel. The review built upon considerable previous work on Severe Accidents, Source Term, and FFRD, and on insights from the recent ATF-PIRT.

The literature review and scoping calculations strongly suggest that The Containment Source Term is bounding compared to an equivalent analysis that explicitly considerers the effects of FFRD. One of the main reasons for the finding that the current source term is bounding is that, unlike other situations where fuel fragmentation effects are significant (e.g., analyses to ensure operability of ECCS (10CFR50.46)), the RG1.183 Containment Source Term already involves "substantial meltdown of the core with subsequent release of appreciable quantities of fission products." The possible release from <2% of the core fuel mass that fragments simply cannot compare to releases from the rest of the core, especially when the same radionuclides that may be released from fragments are already considered to be released in the standard Containment Source Term. Furthermore, only a small fraction of the dispersed fragments in a few scenarios can reach containment, and dispersed fragments aren't exposed to the heat from clad oxidation and are unlikely to get hot enough to release their radionuclides. If these fuel fragments had remained in the core, they would have been exposed to heat from clad oxidation in addition to decay heat and thus would have released their volatile radionuclides. It is therefore expected that fragmentation will reduce radiological releases to containment compared to the same scenario for which fuel does not appreciably fragment and disperse.

One area where there is less certainty on this effect is in the NG releases. Although FFRD does not increase NG releases overall, it results in earlier NG release upon gap rupture, especially for longer-lived NG radionuclides and slower heat up. Given that NGs are not the major contributor to dose, any increase early in the accident is expected to be far outweighed, in terms of human dose, by the reduction in releases of Cs and I in addition to the reduction in overall NG releases. Although not a primary contributor to human dose, NGs are the primary driver for equipment qualification doses. It may be useful to quantify the effect of early NG release with scoping MELCOR calculations.

Other findings and conclusions include:

- The fragment contribution to ST from recovered LBLOCAs is not significant compared to RG1.183 ST
  - ST for recovered LBLOCAS (gap + fragments for some burst rods) in 10CFR50.46 TH analysis (ECCS systems operational), even if one conservatively neglects the facts that that some fragment do not reach containment and that some cannot heat up, and that all volatiles won't release (e.g. 2% I in fragments + gap inventory of broken rods), does not compare to the unrecovered LBLOCA analysis that contributes to the 10CFR50.67 ST for which ECCS does not function and involves "substantial meltdown of the core with subsequent release of appreciable quantities of fission products" (10CFR50.67 footnote 1). The 10CFR50.67 analysis involves the release of fragments, gap, and radionuclide release from the fuel itself "in-vessel" release.
- Possible ST increase due to fragments, even when assuming unrealistically-conservative fragmentation, transport, temperature rise, and RN release, is limited for the unrecovered LBLOCA and will not significantly affect the entire source term.
  - The LBLOCA is only one of many scenarios that contribute to the 10CFR50.67 Source Term provided as Tables 1, 2, and 4 in RG1.183. The release fraction and timing values in

these tables are obtained by conducting severe-accident system code simulations (previously STCP, now MELCOR) of scenarios covering nearly the entirety of the core damage frequency, then combining the resultant releases and phase timings into distributions. The values in Tables 1, 2, and 4 in RG1.183 and is the 2011 reanalysis for high-burnup and mixed oxide fuel represent a percentile (50th percentile (median) or 70th percentile) of these distributions.

    - Fuel that is assumed to fragment no longer contributes to the releases that contribute to the current ST. The RN release from fragments often adds the contribution to ST that was removed upon fragmentation.
    - Slower scenarios provide more time for decay power to decrease before core is uncovered. Many of the scenarios contributing to the RG1.183 Containment Source Term experience several hours of time after shutdown before releases begin. This affects accident progression and fragment specific power upon release.
    - Fuel fragments transport less for other scenarios than for the LBLOCA. The LBLOCA events involve much faster blowdowns and higher gas velocities than other scenarios. Vapor velocities in other scenarios are far lower and carry far less of the fragment mass distribution.

- Simplified bounding estimates for fragment transport, and heat-up, provide estimates of deviations from the conservative assumptions.
- Some of the evaluations and scoping calculations include:
    - Independent bounding calculation indicates that less than 2% of the fuel fragments, at least for the PWR loading pattern considered here. This is consistent with previous assumptions.
    - Settling velocities were evaluated as a function of fragment size for conditions spanning those in reactor accident scenarios.
    - The fragment size distribution was conservatively extrapolated to sizes smaller than the smallest sieves. This size distribution, along with the evaluated velocities, can be compared with scenario-, location-, and time-specific gas velocities to estimate the fraction of fragment mass that transports in a given direction (up or down).
        - The comparison was not made because insufficient time was available to obtain or estimate these gas velocities.
        - The size distribution and gas velocities can also be used to estimate the extent to which particles are likely to navigate tortuous pathways to escape to containment. This was also not done.
    - Evaluations of fragment temperatures while airborne considering convection only indicated that, except for the largest fragments, temperatures would be close to that of the ambient gas not long after reactor shutdown. Considering thermal radiation would further reduce the fragment temperature. Fragment debris bed temperatures, which are also of interest, were not evaluated.
    - Radionuclide releases depend primarily on fuel temperature, time at temperature, and gas composition. Unless fuel heats up to high temperatures, most radionuclides do not escape. High temperatures are achievable only if fragments accumulate or are otherwise heated. Since the analyses of thermal conditions where releases might occur were not evaluated, neither were the releases themselves.
- Other issues include additional fragmentation upon burst, leaching, criticality, and potential for fragment breakup upon expulsion from fuel upon collisions with adjacent fuel rods.

    - The question has been raised of potential fragmentation to smaller sizes than those observed in the experiments due to postulated additional fragmentation upon impacting adjacent rods. If this behavior would occur in prototypic situations, it may have also occurred in the experiments, with the reported size distribution reflecting this additional fragmentation mechanism.
    - Leaching of RNs has been raised as a concern since they have a larger surface-to-volume ratio than, and thus would leach faster than, regular fuel debris. Even if one assumes that fuel fragments fully release RN by leaching, the magnitude of radionuclides in the expected bounding fuel fragmentation fraction does not compare to current nominal Containment Source Term releases.
    - The criticality effect is not considered to be related to FFRD. The discussed effect pertains to fuel designed for high burnup when it is at low burnup. Fuel at low burnup doesn't fragment.
- A substantial fraction of expelled fuel fragments is considered to potentially reach the containment only for the PWR Hot Leg break.
    - It is considered highly unlikely that fragments will successfully escape the RCS for PWR cold leg break, the BWR main steam line break, and the BWR recirculation line break
        - The pathways to containment are tortuous for these scenarios.
        - This assessment involves the assumption that the steam separators and dryers that are intended to capture droplets remain intact.
    - Flow velocities for other scenarios are expected to be too low to substantially loft a significant fraction of expelled fragments.
- Fuel fragments are typically much larger than typical nuclear aerosols. Fuel fragments transport less readily, fall faster, and deviate from streamlines faster, so they are more likely to hit obstacles in flow and not successfully traverse bends in flow.

- The MELCOR developed CST is considered bounding relative to those for CST effects except for perhaps some early NG release. MELCOR sensitivity calculations would be useful.

Given the results of the review and analyses the following assessment is made regarding the Applicability of RG1.183 "Endorsed" Codes. RG1.183 explicitly mentions ORIGEN and RADTRAD. MELCOR (and previously STCP) is being used to develop the Containment Source Terms like those in RG1.183. The following conclusions were made about these codes:

- ORIGEN is considered to be directly applicable as long as fragmentation doesn't affect fuel location during operation. If fragmentation results in mobile fuel this may have to be considered.
- RADTRAD is considered to be directly applicable as long as the form of the source term remains unchanged.
- MELCOR captures the major phenomena. It may be good to update the gap inventory (either default or user input) depending on burnup. It may also be of interest to track some fragments to test the effects of fragmentation on hot leg failure.

Identified limitations in the work include:

- Insufficient time was available to combine the effects of the calcs into an overall effect, the conclusions reflect trend and expected results if more time had been available to combine and synthesize sequence-specific effects on the source term.
- These assessed impact on ST and the values in tables currently reflect general expectations on median values rather the aggregated results of sequence-specific assessment of FFRD influence on source terms.
- Fragmentation was not evaluated for BWRs explicitly,

Future work (analyses that are left partly undone and could use improvement) could include:

- Updates to the evaluation:
  - Check MELCOR-evaluated flows at rod burst to verify adequacy of assumption
    - Simple assessment of fragment mass fraction up or down
  - Tie together scoping analyses to get full quantifications for each analyzed accident scenario used to develop the source term.
- Updates to MELCOR usage for best-estimate analyses

There may exist limited utility in expending considerable effort in reducing uncertainties related to FFRD that are dwarfed by other uncertainties including scenario boundary conditions, fission product speciation, and other uncertainties in core degradation behavior.

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Appendix A: Estimation of fuel fragmentation fraction

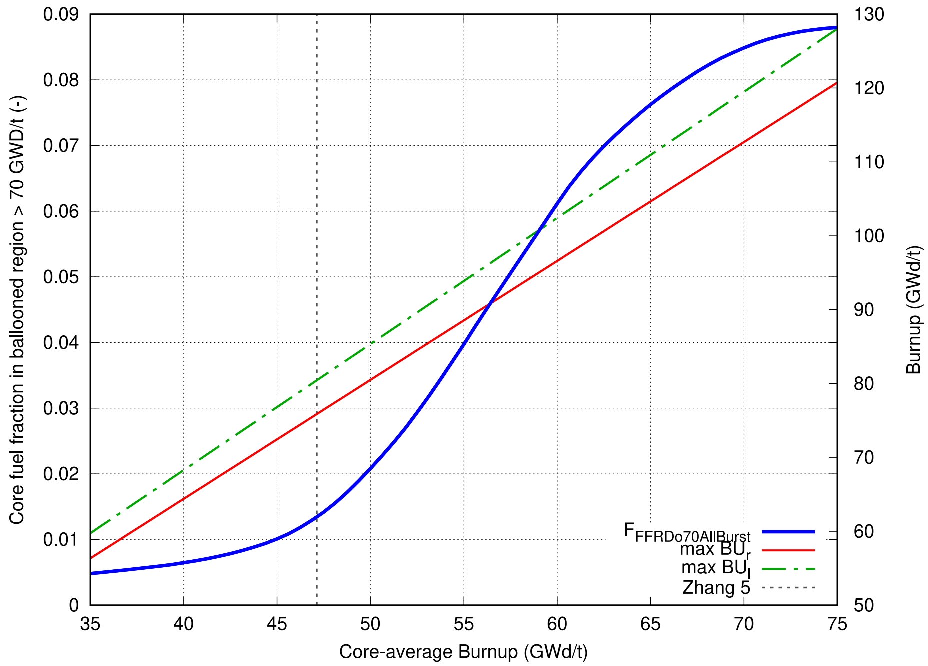

Figure 19 Bounding FFRD fraction (extrapolated far beyond realistic values to estimate fragmentation behavior)

It is known that this curve extends way outside its possible range of validity and beyond expected currently possible ranges for fuel. The wide range beyond the original source (Zhang) was chosen to observe the trends that would result by extrapolating the behavior. If loadings at higher burnup are similar, the trends may remain valid beyond the physically achievable burnups. If loadings at higher burnup differ substantially, they won't not be applicable.

This section describes the initial scoping calculations to estimate the fraction of fuel mass that becomes fragmented, the size distribution of released fragments, and the power density of released fragments. The information used included proposed high burnup loadings, data from burst experiments, and results of previous analyses and calculations. Only readily available information was used for these scoping calculations. Distributions were scaled to estimate effects at higher burnup. Where uncertainty exists, conservative assumptions were made.

The following initial assumptions were made in the initial estimation of the dispersed fragment fuel fraction. They can be updated if needed.

- Use [Capps,2021] and [Zhang,2019] (core design option 5) EOC information for bundle, rod-average, and pellet power and burnup distributions.
  - Use 1/8th rather than 1/4 core for analysis. Differences are typically on the 3d significant digit.
  - Linearly scaled (extrapolated) to estimate fragmentation at higher burnups
  - Assumption of uniform bundle and power distributions within bundle
    - à distributions of rod-averages within bundle can be determined from given parameters
      - mean = median

        - Difference from min and mean is the same as from max and mean
            - (mean - min) = (max - mean)
    - Assumption of identical peak-rod to rod-average properties within bundle
        - Peak-rod/rod-average= constant
- Use Studsvik LOCA-193 rod-radial burnup distribution (from [Capps,2020])
    - Assume all rods have this distribution until better information available. Should switch to reactor-type and fuel-type specific distributions
- Use Turnbull pulverization criteria
    - Assume unfavorable burst temperature (resources not available to look up)
    - 70 GWd taken as pulverization threshold
- Can’t calculate stress and strain so parametrically assume burst T- (not used)d
- Assume ballooning length equal to length between spacer grid
    - Assume 6 evenly spaced interior spacer grids → 1/7th of rod ballooned
- The fragment size distribution from [Phillips, 2015] was used, extrapolating distribution trends to lower sizes.
- Fragments have the average peak-pellet power density
- All fuel rods are assumed to rupture

The FFRD fraction is calculated as fraction of fuel mass over all bundles, using axial and rod-radial distributions below, > 70 GWD/t over the ballooned length. Some of the input information is provided below

The determined fraction should be bounding: it uses the max of the Turnbull criteria and assumes all rods fail using a large (1/7th of rod length) ballooned fraction.

Potential modifications:

- Evaluate independently for BWRs and PWRs
- Consider rod burst criteria with local TH to estimate individual bundle failure

FFRD pulverization threshold

The information for the FFRD pulverization threshold was obtained from [Capps, 2021].

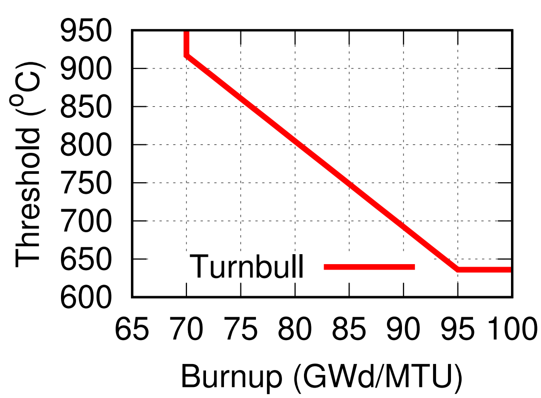

Figure 20 FFRD Turnbull pulverization threshold in °C

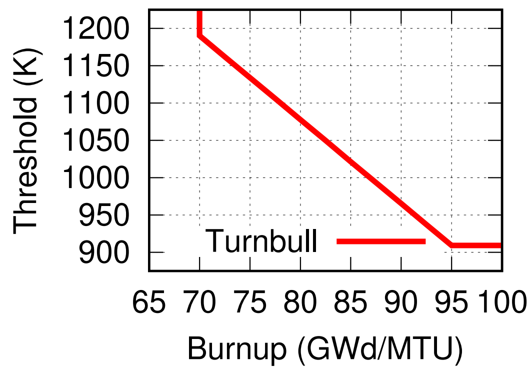

Figure 21 FFRD Turnbull pulverization threshold in K

Burst T From Capps, 2020

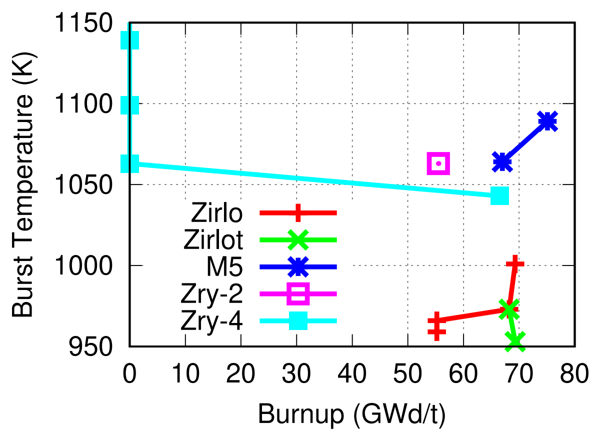

Figure 22 Clad burst temperature for selected experiments compiled by [Capps, 2020]

Capps 2021/Zhang 2019 – fuel loading:

Only 1/8th core used. Source used ¼ core. The variation was small.

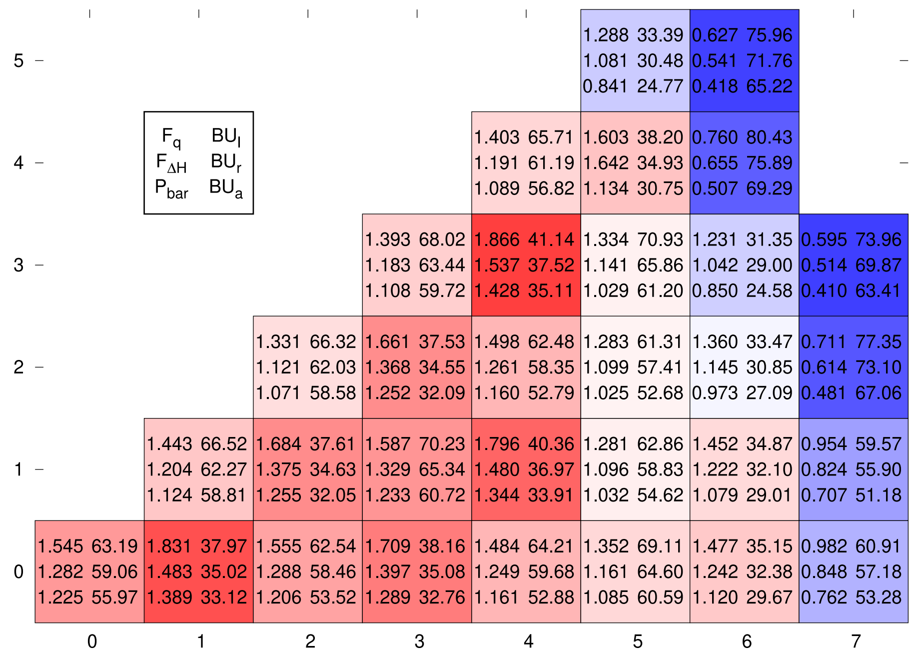

Figure 23 Power peaking and burnup distributions in Capps 2021/Zhang 2019 (colored by bundle average power)

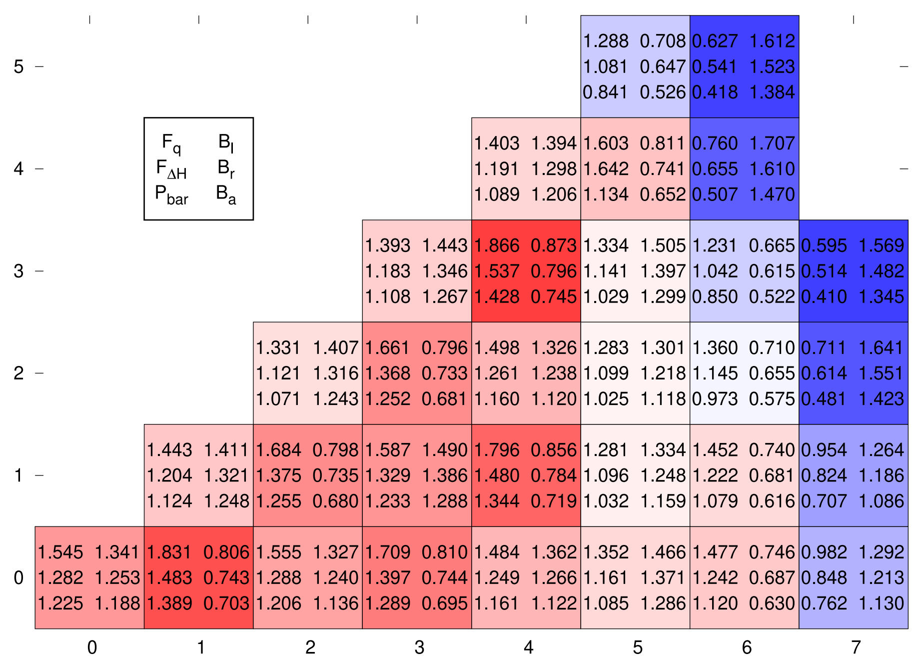

Figure 24 Normalized burnup (colored by bundle average power)

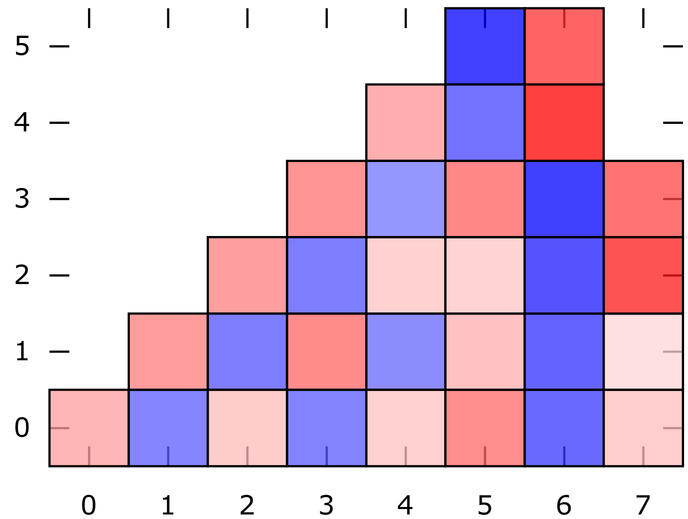

Figure 25 The same loading as previous figure but colored by burnup

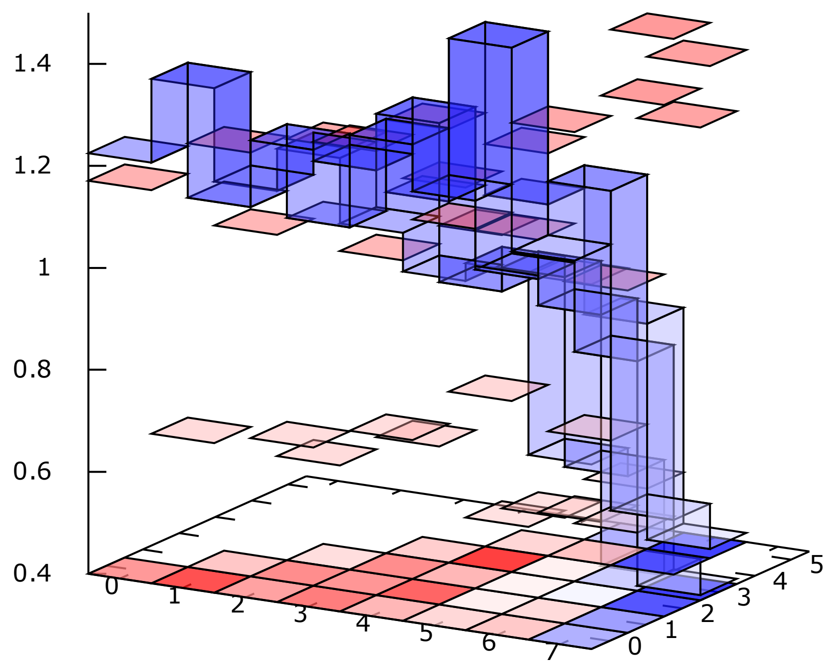

Figure 26 Relative Burnup (red) and Power (blue).

High burnup frac and high power frac do not coincide.

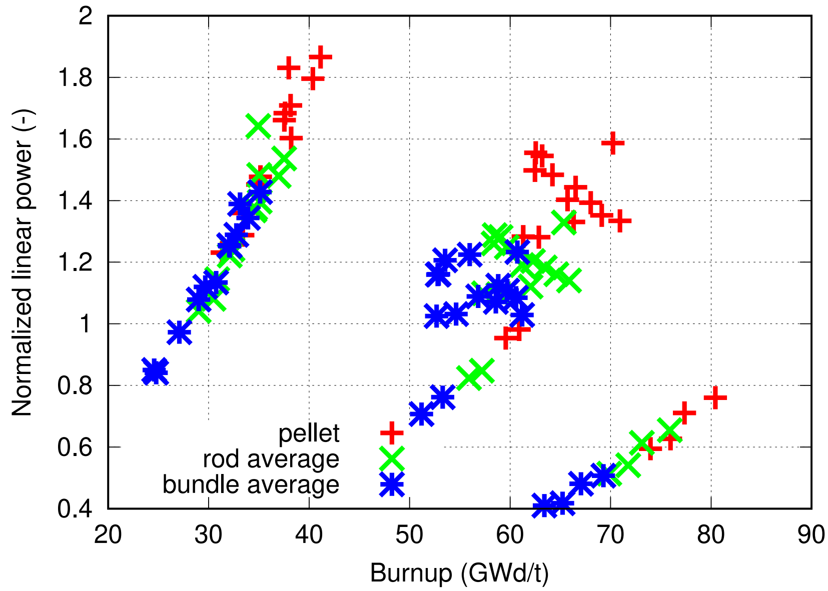

Figure 27 Burnup/power distribution

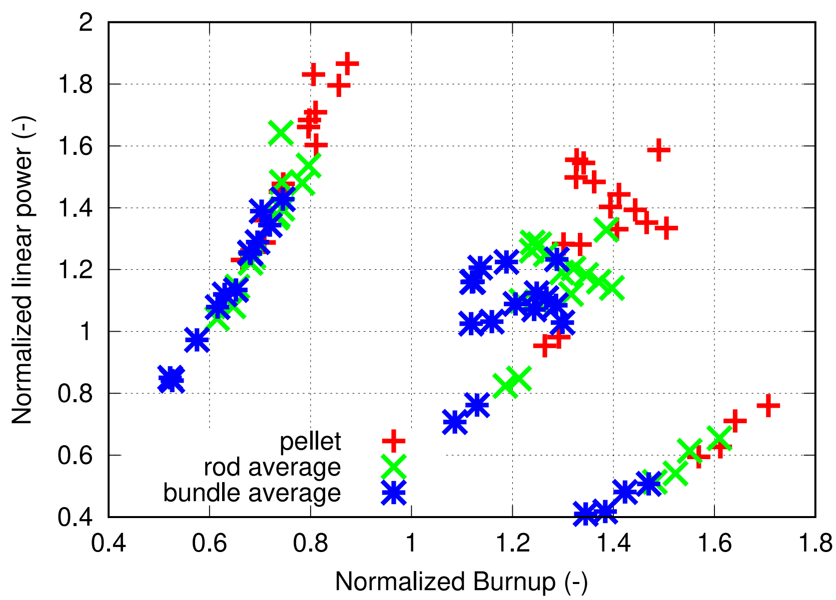

Figure 28 normalized-burnup/power distribution

Rod radial burnup: - from Capps 2020

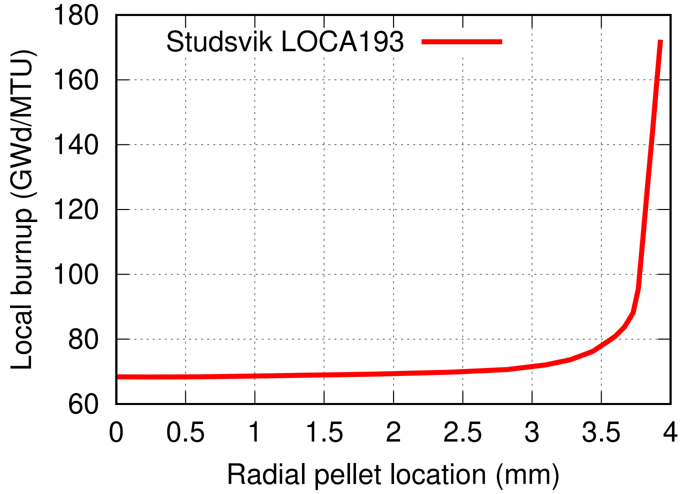

Figure 29 Rod-radial burnup profile in Studsvik LOCA193

Normalized BU rod-radial CDF:

Evaluated from rod radial-burnup:

Pellet radial relative burnup distribution derived from Studsvik LOCA193

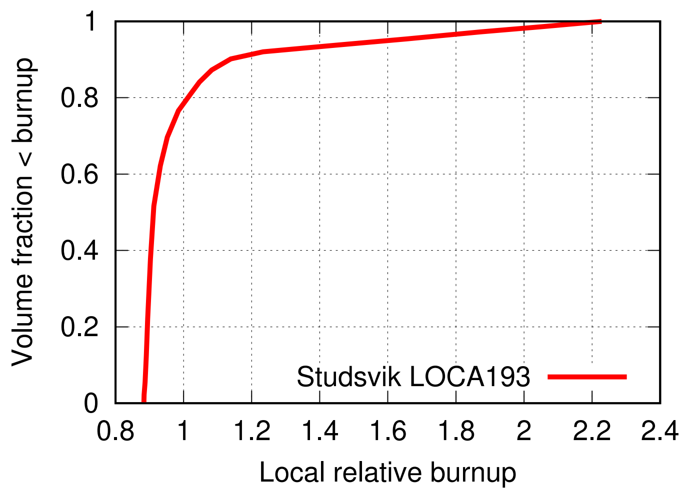

Figure 30 Used rod-radial burnup distribution

Assuming constant density (not correct), this then becomes an estimate of the mass fraction distribution for an axial rod section.

Distribution processing

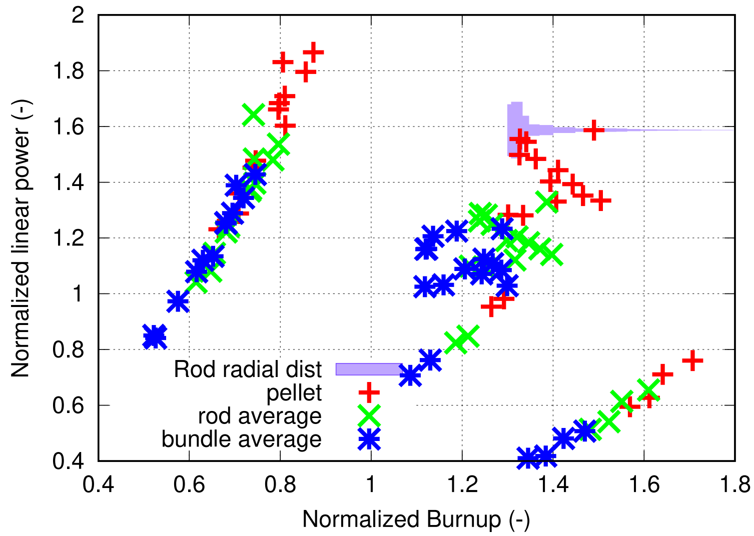

Figure 31 Rod-radial burnup distribution pdf superimposed on top right peak pellet burnup assuming all fuel in the pellet cross section has same power density (This approach is expected to be conservative)

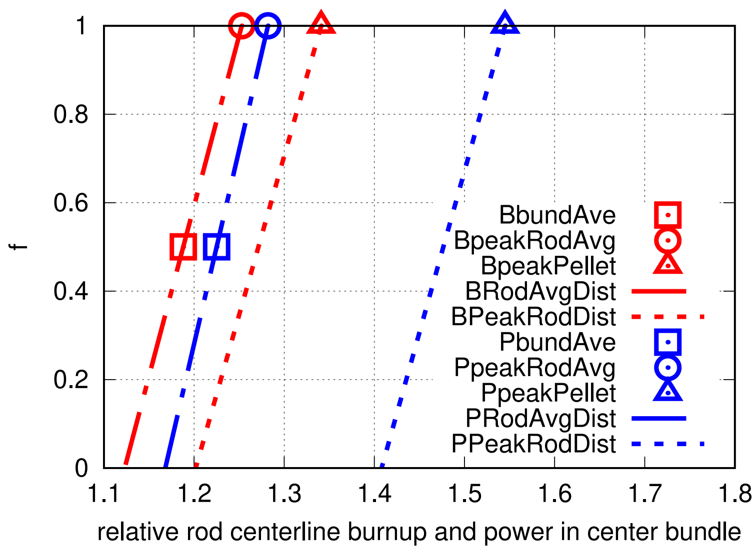

Figure 32 Derived rod-average and rod-peak power and burnup distributions within center bundle

Figure 32 shows derived rod-average and rod-peak power and burnup distributions within center bundle (using assumptions of uniform rod-average distribution within bundle and constant peak/rod-average ratio within bundle). The points are the power peaking normalized burnup of loadings provided by [Zhang, 2019] and duplicated in both [Capps, 2021] and in Figure 23. The normalized burnup values are shown in Figure 24.

The lines connecting the circles and squares represent the distribution of rod-average power and burnup values within the center bundle. The lines connecting to the triangles represent the rod-peak power and burnup values (at center of each rod) within the center bundle.

Figure 33 shows the derived rod-peak (peak-pellet) power and burnup distributions for all bundles along with the number of bundles of each type. These are the endpoints of the triangle-connected lines in Figure 32.

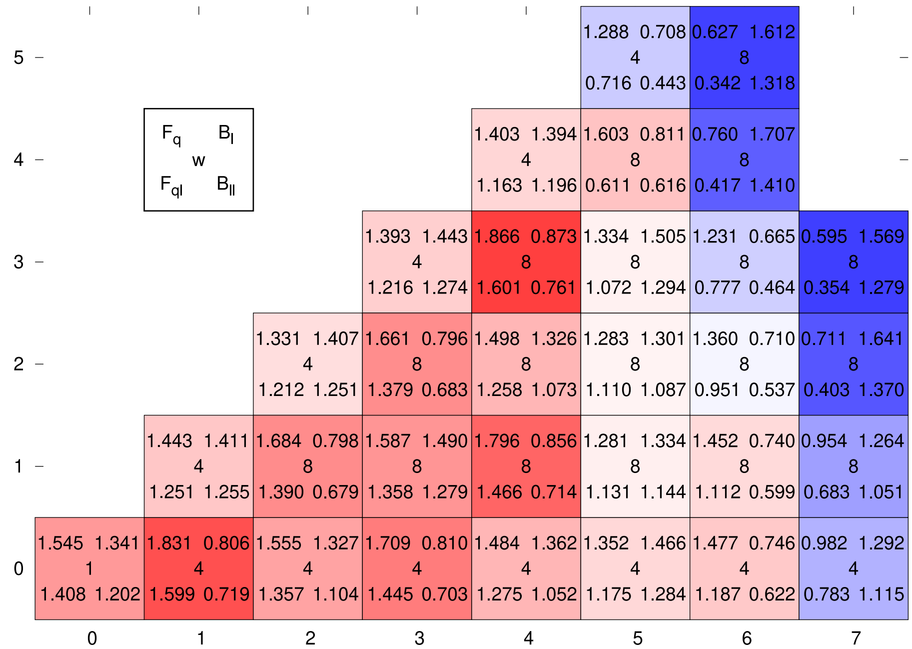

Figure 33 Peak-pellet in max rod over peak-pellet in min rod (power | burnup) along with weight (# of bundles of this type)

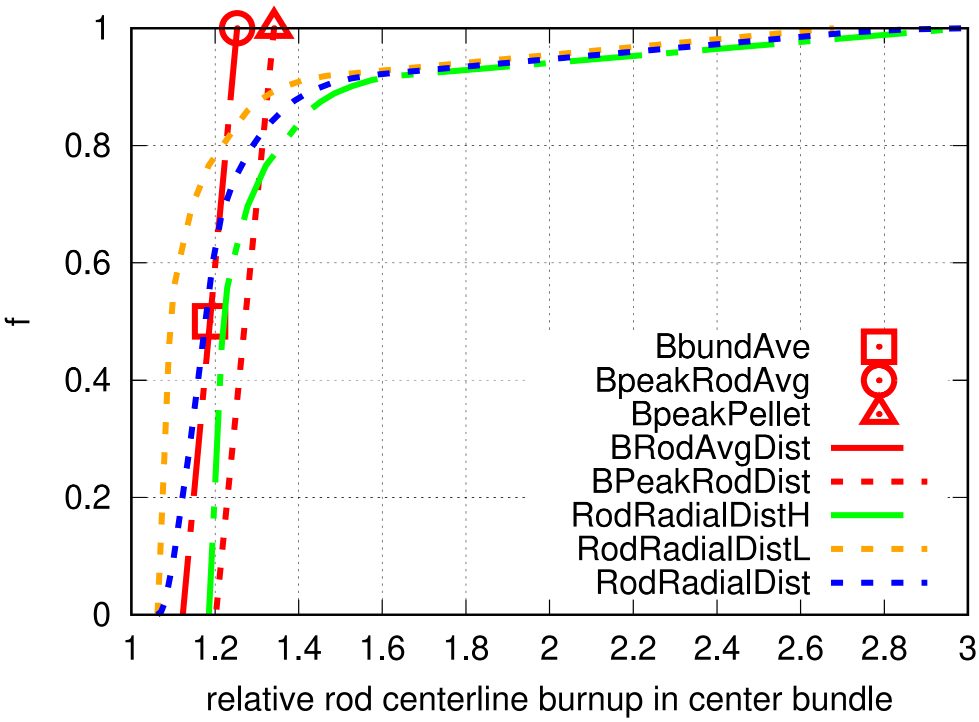

Figure 34 Application of Studsvik LOCA193 rod-radial burnup distribution to centerline burnup distributions

Figure 34 represents the application of Studsvik LOCA193 rod-radial burnup distribution to the max pellet burnup in max-burnup rod within bundle (green), max pellet burnup within min-burnup rod within bundle (orange), and over the max-pellet burnup distribution within bundle (blue).

Figure 35 shows the equivalent rod centerline burnup distributions for all bundles. This information is then processed to evaluate the fraction of fuel that fragments and disperses during an accident using 70GWd/t as the limit.

Figure 36 represents the fraction of fuel in each bundle's rod centerline with burnups greater 70GWd/t. Figure 37 is the same distribution, but colored by radial position (blue -center, red-periphery). Multiplying by the number of bundles of each type results in Figure 38. Dividing by the total number of bundles provides the core-wide fraction of each bundle. This is shown in Figure 39. Figure 40 is the sum of all bundle fractions. This represents the core-wide fraction of fuel greater 70GWd/t at the rod centers.

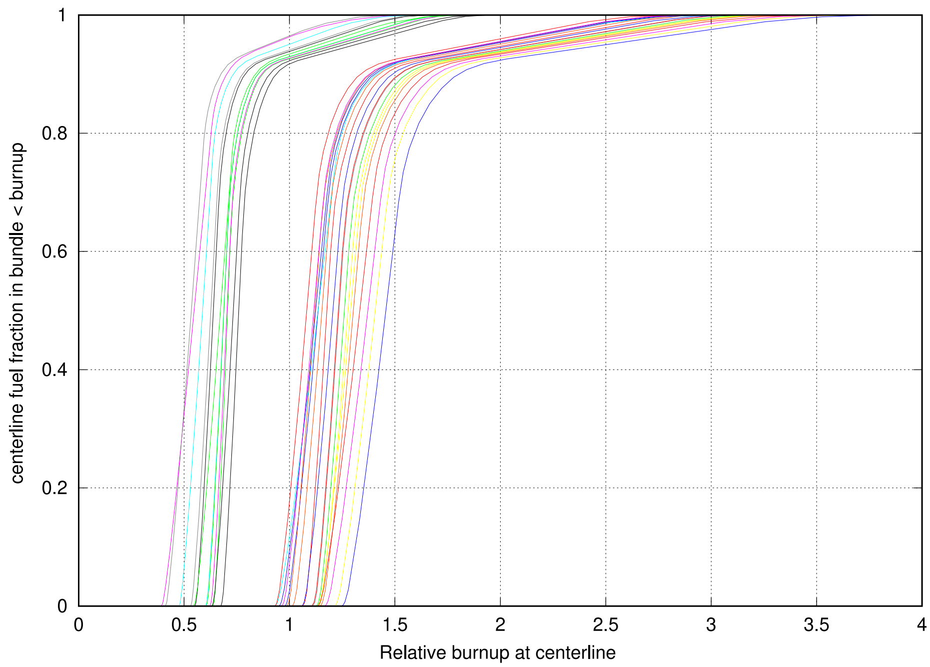

Figure 35 Rod-centerline burnup distribution for all bundles

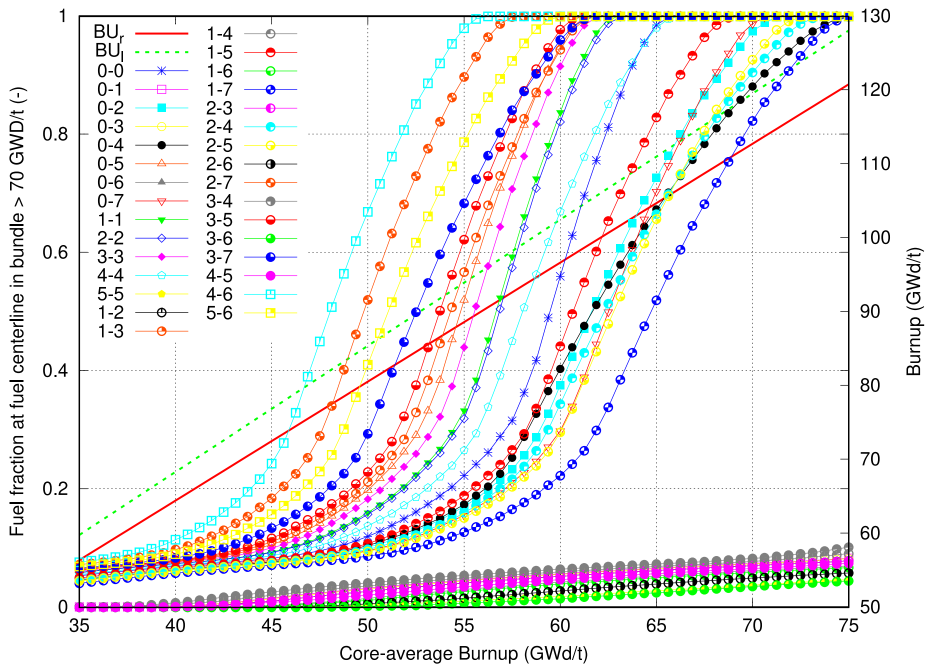

Figure 36 Fuel fraction at rod centerline > 70 GWd/t as function of core-average burnup

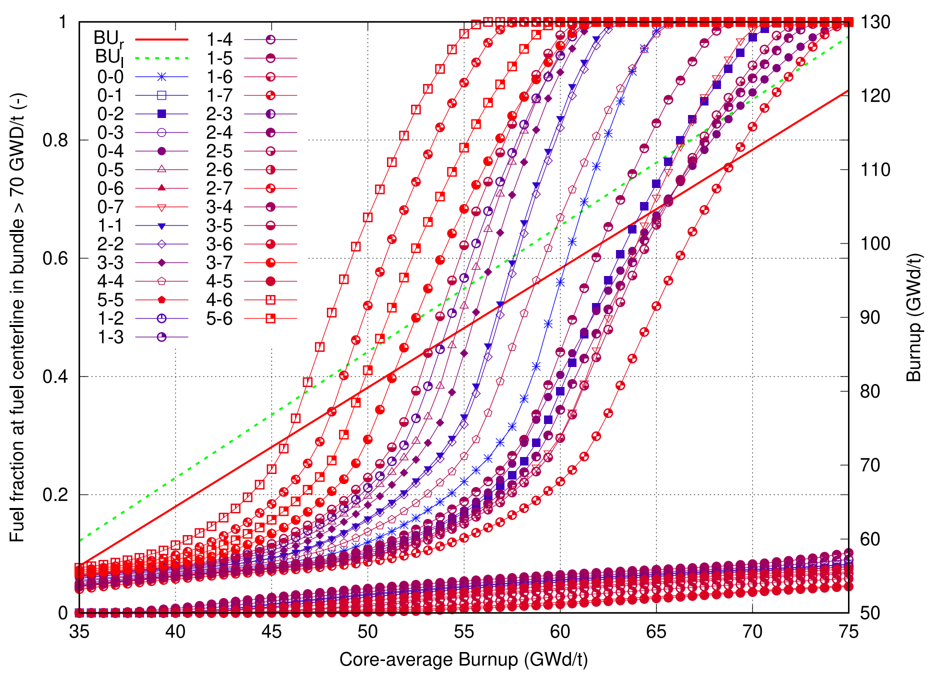

Figure 37 Fuel fraction at rod centerline > 70 GWd/t as function of core-average burnup Colored by radial location (blue - center, red-periphery)

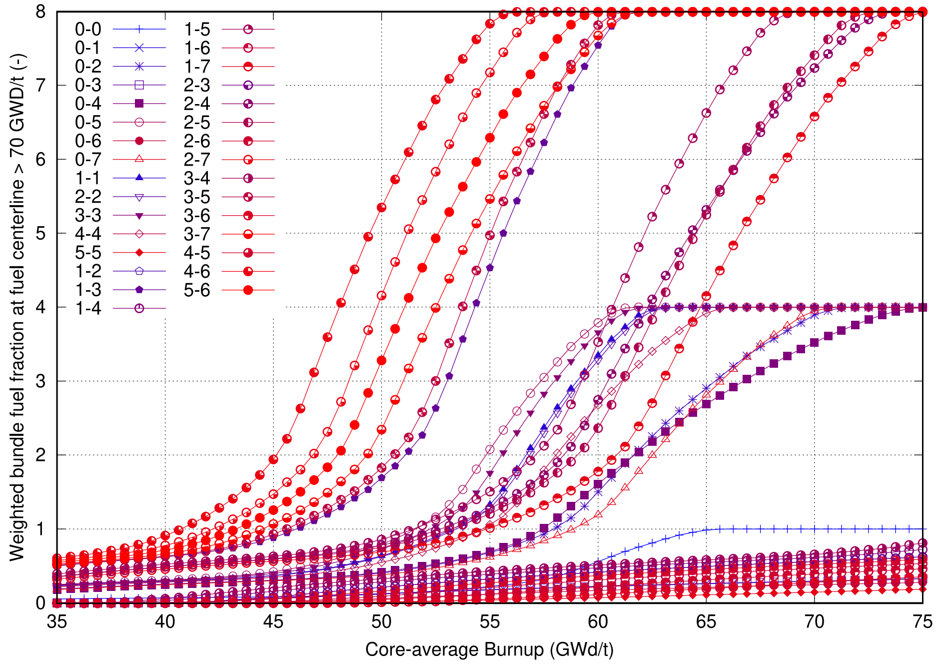

Figure 38 Fuel fraction at rod centerline > 70 GWd/t as function of core-average burnup multiplied by respective weights

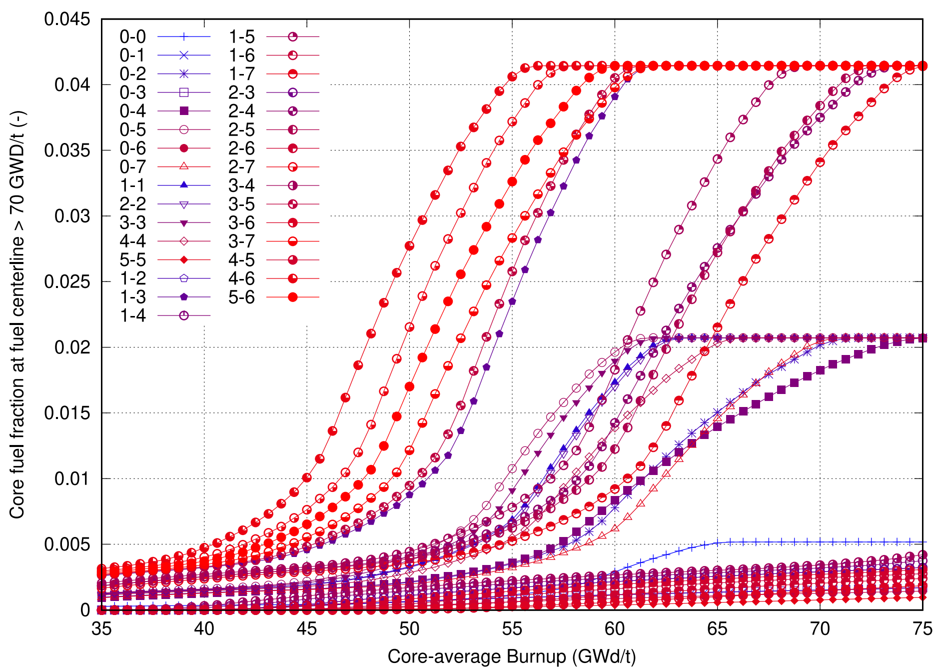

Figure 39 Fuel fraction at rod centerline > 70 GWd/t as function of core-average burnup converted to core fraction

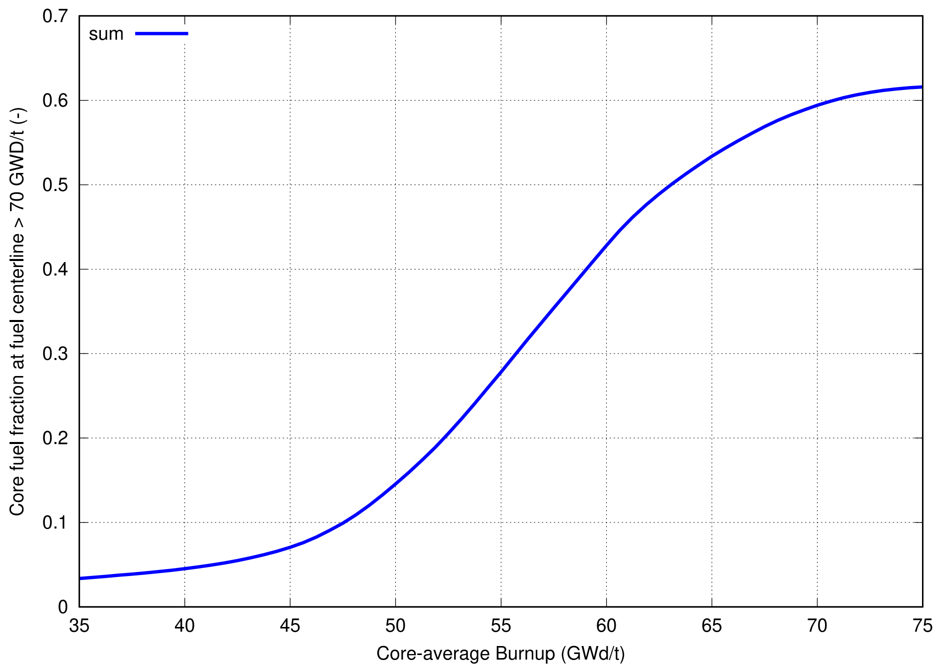

Figure 40 Sum of core fractions. This represents the core-wide fraction of fuel at rod centerline > 70 GWd/t

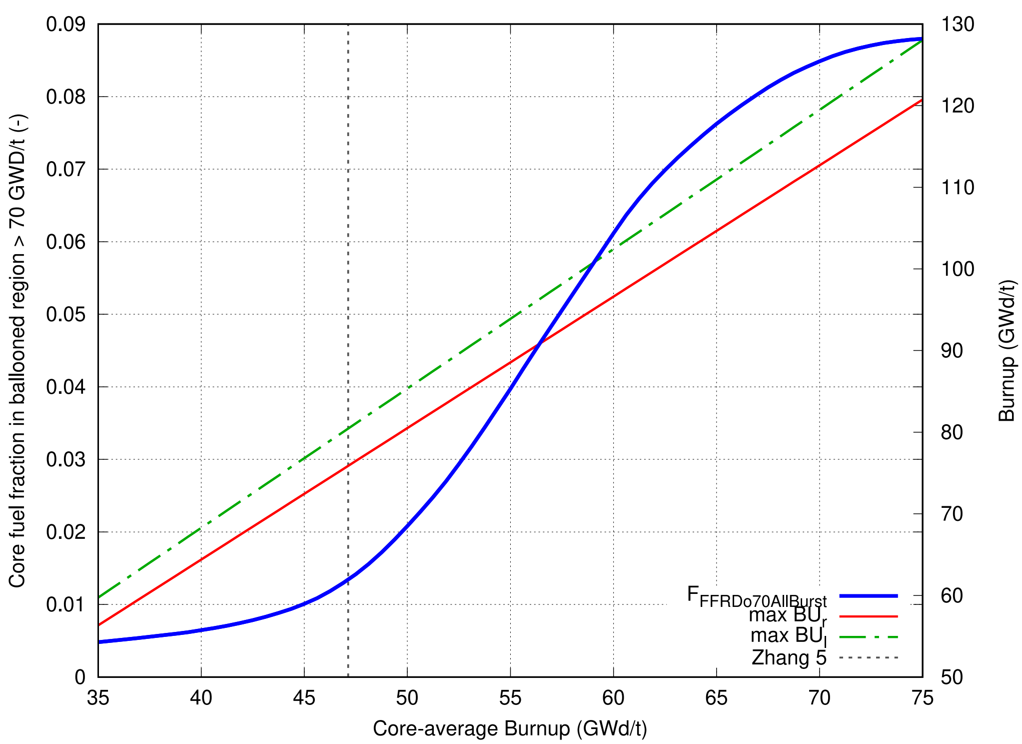

Figure 41 Fraction of fuel that is considered to fragment and disperse

Figure 41 shows the Fraction of fuel that is considered to fragment and disperse = the FFRD fraction. This is obtained by evaluating the fraction of fuel > 70 GWd/t within the ballooned region. This value is obtained by multiplying the core-wide fraction of fuel at rod centerline > 70 GWd/t by ballooned length fraction. The ballooned length fraction was assumed to be 1/7 of rod length. This evaluated FFRD fraction should be bounding since it assumes a large ballooning length and all rods assumed to fail including peripheral ones.

Other Considered Power distributions

These distributions can be used to relate the peak per rod to peak rod-average of distributions

Note: These distributions were not used in the evaluation. The peaking factors listed above were used instead.

PWR - NUREG-1754_2001

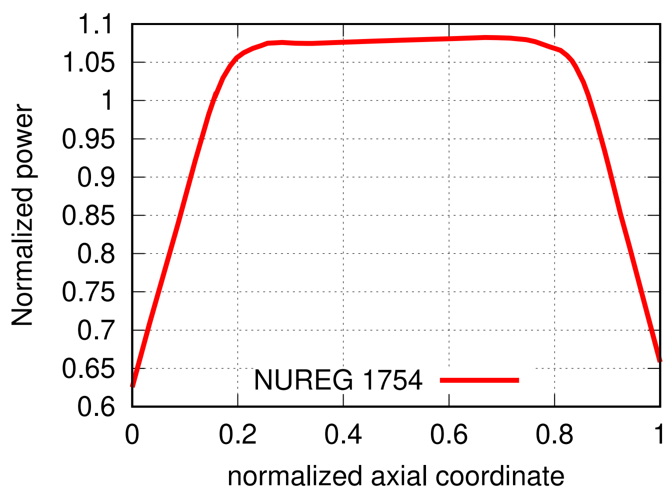

Figure 42 PWR axial power distribution from NUREG-1754

BWR - SAND2008-6664

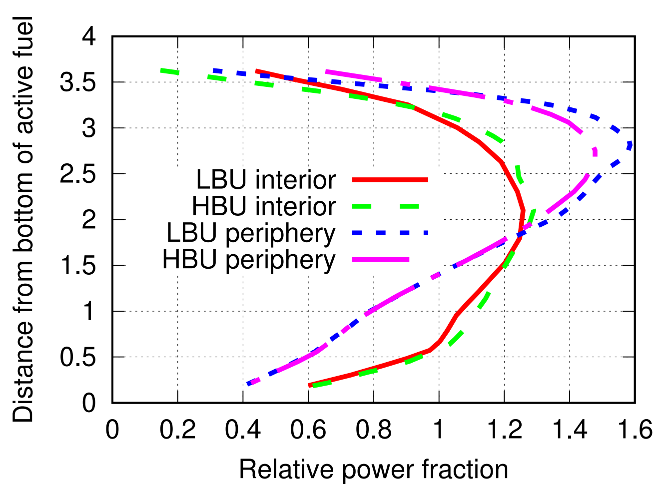

Figure 43 BWR axial power distributions from SAND2008-6664

Rod radial distribution

The same rod-radial distribution was used for both BWRs and PWRs until better information could be obtained.

Ballooning length:

Considered to be bounded by length between spacer grids

The ballooning length is assumed to be the space between 6 evenly-spaced interior spacer grids in absence of time to find available usable information. This is likely quite conservative.

There was a mention that “A significant fraction / maybe higher ~ 10” of 12 ft rod fragments.” Fragmentation experiments use shortened rods with max 50 cm rod length so it isn’t necessarily representative of prototypic behavior. Should look for a reference for this value. This corresponds to 0.0694 ballooning length fraction rather than 1/7(=0.143). Changing to this expected ballooning length would half the FFRD fraction.

Appendix B: Excess Early NG Release at higher burnup

Data on NG release from RIA experiments, LOCA experiments, and fuel characterization experiments were used to evaluate potential enhanced early noble gas release from high burnup fuel. Although the effects of fragmentation were not explicitly considered in these experiments the available information seems sufficient to conservatively estimate its contribution. For the purposes of analyzing the potential impact of fragmentation on the Containment Source Term, using bounding results for long-lived ^{85}Kr isotope was sufficient. It may be useful to consider the difference in intragranular and intergranular fractions for different noble gases for evaluating gap inventories.

Existing NG release models derived from these test programs may have already considered these effects. Detailed modeling of spatial NG behavior was found (L. Noirot, 2006 and L. Noirot, 2011) but was not reviewed in detail.

Whereas RIA and LOCA experiments measure RN release from fuel for simulated experimental scenarios, the purpose of the fuel characterization experiments (ADAGIO experiment/technique and EPMA analyses) are to identify the location of fission gases within the fuel (grain or grain boundary).

Table 13 shows Maximum Kr releases from LOCA Heat up Tests (1200 ^{o}C) (48.5, 71.8 GWd/t) from Pontillon, 2004 GASPARD tests and Noirot, 2014 103.5 GWd/t test in the MERARG facility. The LOCA tests with slower release rates and thus greater time at temperature result in greater releases for the same temperature.

Table 13 Maximum ^{85}Kr releases from LOCA Heatup Tests (1200 ^{o}C) (48.5, 71.8 GWd/t) from Pontillon, 2004 GASPARD tests and Noirot, 2014 103.5 GWd/t test in the MERARG facility.

Burnup (GWd/t)	^{85}Kr Release up to 1200 ^{o}C
48.5	0.108
71.8	0.212
103.5	0.329 (fuel disc)

Concerns have been raised that fuel fragmentation on burst could liberate NGs on grain boundaries. One may consider it unclear how much this effect could potentially enhance releases relative to those from LOCA tests which involve the heating of previously-punctured irradiated fuel samples.

It may be best to address NG releases parametrically. It is expected that a somewhat earlier NG release would not substantially change consequences from those in the nominal RG1.183 Containment Source Term (10CFR100.10), even when using bounding NG release, in terms of both early (gap) release and magnitudes.

It seems, from initial review of the experiments, that all or nearly all the intergranular releases are already considered released by 1200 ^{o}C. If this is the case the additional contribution of intergranular NG release does not exist – one can't release more of the intergranular gases if all the intergranular gases have already been released – in which case the measured NG release fractions from the experiments up to 1200 °C would still apply.

Based on this it is assumed that the NG releases are already covered in the above table.

The recommendation for a near-bounding early NG release upon rod burst, including any effects from fragmentation, would be to interpolate the above table based on local burnup. This release amount corresponds to slow heat up rates (0.2 °C/s) and long-lived NG isotopes (^{85}Kr). The early NG fraction released can be substantially lower for shorter-lived NGs and for faster heat up rates with less time at high temperature.

Sufficient data are available to develop detailed mechanistic release models. CEA has already developed NG behavior and release models based on many of these experiments [Noirot, 2006] [Noirot, 2011].

The rest of this section describes the initial potential models considered for enhanced NG release due to fragmentation and the experimental data that led to the above expectations and recommendation.

Insufficient time was available to review existing models in detail or to perform a thorough comparison of the different observations for NG release behavior (RIA tests, LOCA tests, ADAGIO).

Considered modeling

Two potential approaches to address additional early NG release were considered:

It seems like the obvious choice is frac pulverized * intergranular NG fraction F_{NG} (= fraction of intergrain NG /total NG) where both the pulverized material and intergranular fraction depend on pellet burnup and radial location within pellet and perhaps other stuff.

For the estimates of the spatial distribution of F_{NG} as a function of burnup, perhaps the work of Noirot can be used, [Noirot, 2006] [Noirot, 2011]. It seems that L. Noirot thoroughly reviewed much of this same material during model development.

As an initial bounding value for F_{NG}, the (Intergrain/total NG) measured for this analysis.

F_{NG} Ravel 2000: (0.372 center, 0.209 peripheral, 0.274 overall for 4-cycles in Gravelines reactor: ~ 48 GWd/t).

The second approach that was decided on was to use that observed to be released in LOCA tests which may be more representative of the scenarios consider in the derivation of the Containment Source Term.

[Pontillon, 2004] LOCA tests: 4 cycle (~48GWd/t) ^{85}Kr ranges from .032 to .088 (+~0.015 to 0.02 upon puncture), 6 cycle (~72GWd/t) ^{85}Kr ranges from .062 to .153 (+~0.058 prior to rupture). Longer-duration tests (lower T ramp rates) released higher NG fractions. Temperature ramp rates ranged from 0.2 to 20 °C/s. ^{85}Kr release data is also available for 103.5 GWd/t disc (0.30) [Noirot, 2014]

Sources of information on NG release during accident scenarios

RIA experiments indicate enhanced early release of noble gases upon clad burst with increasing burnup. Figure 44 shows NG releases from RIA experiments used in Guidance [Mendiola, 2009].

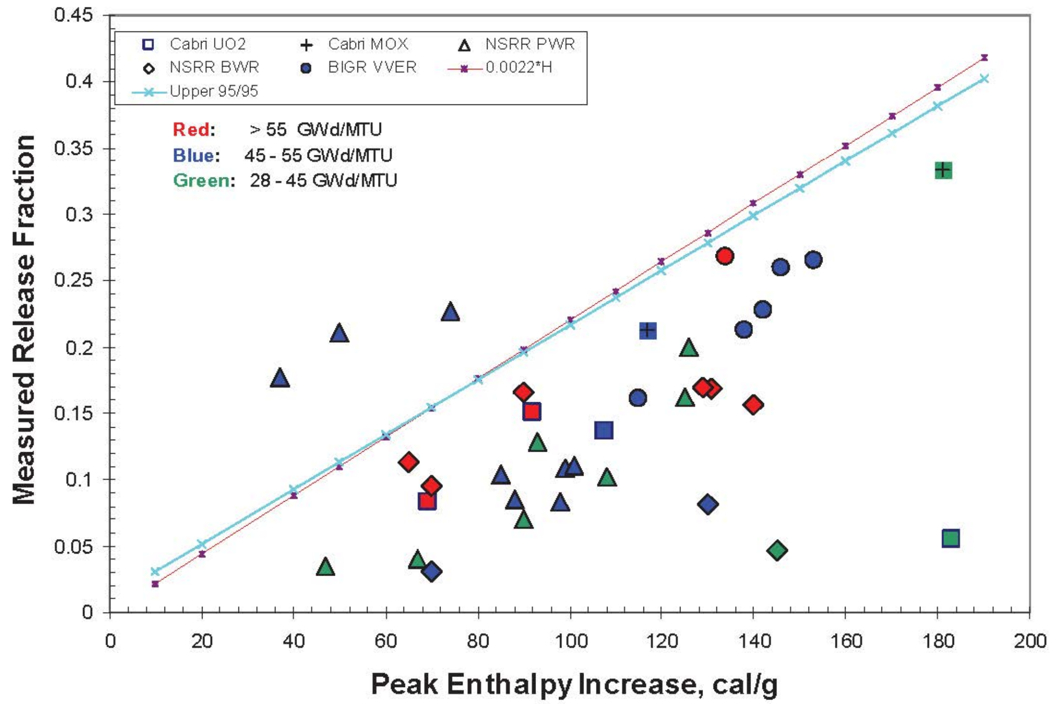

Figure 44 RIA release as f(enthalpy increase) from [Mendiola, 2009]

It has been hypothesized that the excess NGs results from enhanced release of noble gases in the grain boundaries.

ADAGIO

The ADAGIO facility and experimental technique distinguish between intragranular and intergranular gases and characterize the distribution of each radially within an irradiated fuel pellet [Ravel, 2000]. CEA built the ADAGIO facility at its Grenoble site in 1997. ADAGIO is a French acronym for "Discriminating Analysis of Accumulation of Intergranular and Retained Gas". The approach was developed and tested at the Chalk River Nuclear Laboratory (AECL). The objectives of the experimental program were extending the database to high burnup and transient conditions and informing the development of a mechanistic model. The facility includes a pyrometer, a thermocouple, and a gamma spectrometer which monitor behavior and releases.

The experiments rely on the fact that, after several months after irradiation, the only remaining NG is ^{85}Kr. This NG is located both within the grains and at the grain boundaries in bubbles and as-fabricated porosities. Reirradiation introduces short-lived ^{133}Xe in the grains that preferentially remains in the grains, especially if kept at lower temperatures. The ^{85}Kr releases can be compared against those of ^{133}Xe to determine what fraction of NGs reside in grains or in the inter-grain regions. Pellet sections are cut up to evaluate the distribution of the intra- and inter-grain NGs within the fuel. Heat treatment at 450 ^{o}C oxidizes UO_2 to fine U_3O_8 powder within an hour. The oxidation occurs preferentially at the grain boundaries with some coming from the grains. Assuming that ^{133}Xe comes only from the grains, the NG fraction at the grain boundary is fraction of ^{85}Kr released during this process – fraction of ^{133}Xe released during this process.

The results are generally consistent with those of EPMA although not a perfect match: the fraction of intergranular gas decreases with increasing radial position whereas the global retention increases. [Ravel, et al, 2000] note that EPMA can detect most of the intragranular gases except those in large bubbles but also the fraction of intergranular gases in small bubbles.

Gaspard

Pontillon describes Gaspard tests in [Pontillon, 2004]. Figure 45 and Figure 46 show integral ^{85}Kr releases from fuel in the GASPARD tests when 48.5 and 71.8 GWd/t (local) burnup fuels were heated to 1200 °C, respectively. The samples were re-irradiated prior to testing. The NG release upon puncturing was estimated to be 1.5-2% for the 4-cycle 48.5 GWd/t sample and 5.8% for the 6-cycle, 71.8 GWd/t sample. The puncture releases are not shown in the plot.

Similar releases from fuel irradiated to 103.5GWd/t in the Halden reactor can be found in [Noirot, 2014]. The ^{85}Kr release was approximately 0.30, which includes the 0.029 NG released upon puncturing (total of 0.329). This Integral ^{85}Kr release figure shows releases including the 0.029 release following puncturing. This experiment involved a fuel disk and not a fuel rod.

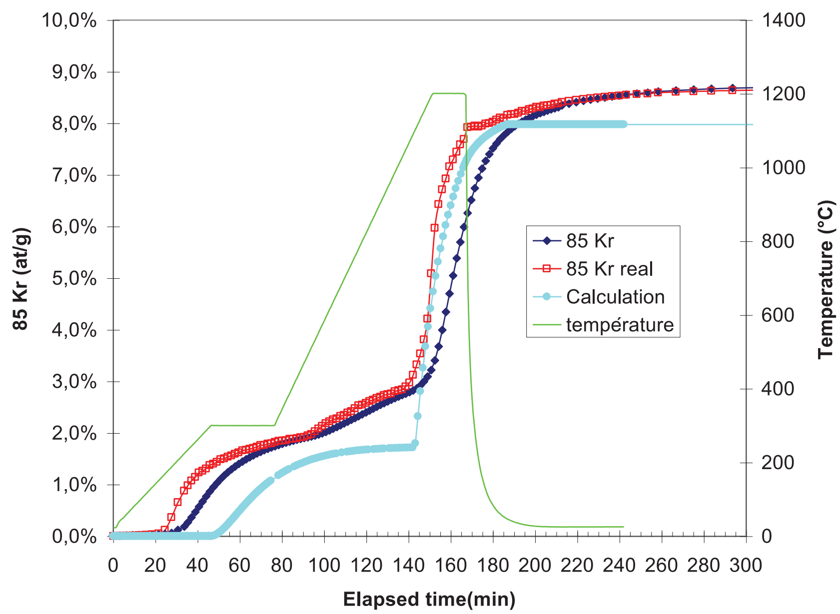

Figure 45 Integral 85Kr release, A-0, 4 cycles (48.5 GWd/t), From [Pontillon, 2004]

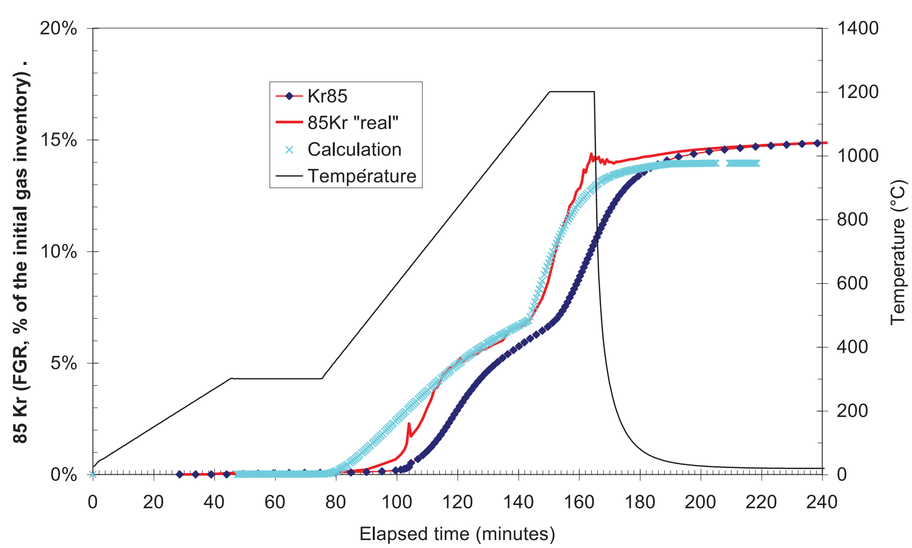

Figure 46 Integral 85Kr release, A-0, 6 cycles (71.8 GWd/t), from [Pontillon, 2004]

The NG release behavior differs depending on burnup. The release rates for the two Gaspard tests discussed above are shown in Figure 47 and Figure 48.

Based on microstructure evaluations and calculations, Pontillon et al, 2004 conclude that two different mechanisms drive the primarily grain-boundary release during the temperature ramp: a rapid growth and interlinkage of grain boundary bubbles and formation of grain edge tunnels, and a fracture of grain boundaries which allows a direct release of gases present in over-pressurized bubbles.

Pontillon characterized fragmentation based on the estimated stress in fuel itself: “Fuel fragmentation due both to the stress distribution on the pellet radius (fuel is in traction in these conditions since it is in compression at the end of the base irradiation), second to the action of the intergranular bubbles on the grain boundaries. In this case, it has been assumed that grain boundaries could be broken if the stress induced by the bubbles reaches 220 MPa.”

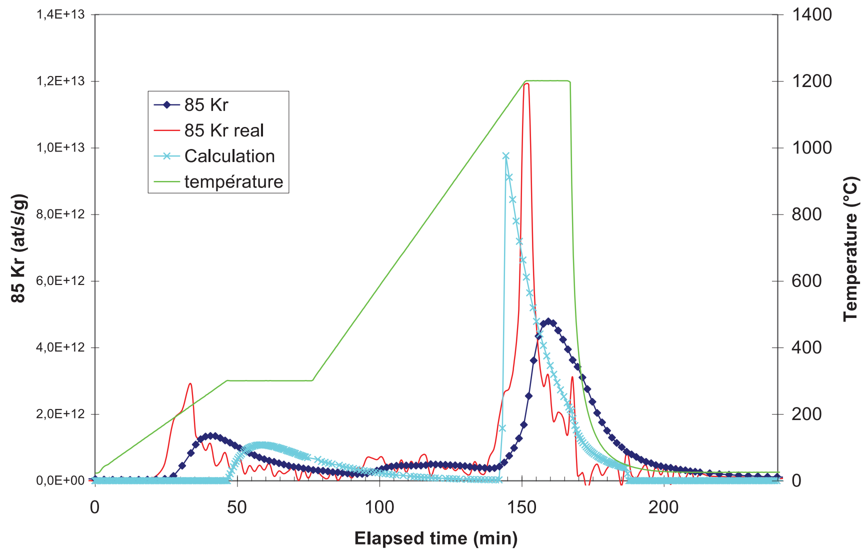

Figure 47 ^{85}Kr release, A-0, 4 cycles (48.5 GWd/t), From [Pontillon, 2004]

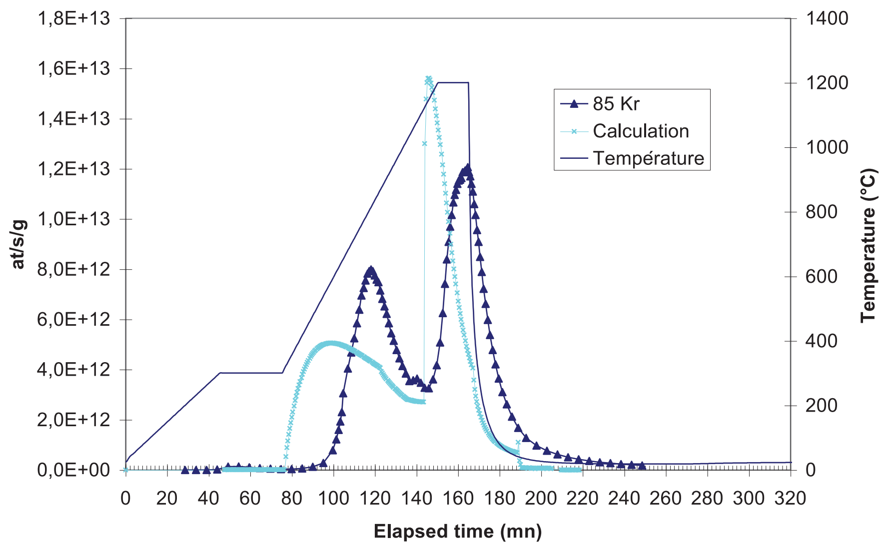

Figure 48 ^{85}Kr release, A-0, 6 cycles (71.8 GWd/t), from [Pontillon, 2004]

The importance of earlier enhanced NG gap release does not substantially change results for accident scenarios which progress to intragranular releases (generally corresponds to the in-vessel release). A few factors limit the importance of earlier NG release: 1) for biological effects, since NGs do not absorb in the body, the contribution of NGs to dose is limited, 2) even for dose to equipment in containment where NG contributes the bulk of the dose, most of the NGs are released after 1200 ^{o}C. Once the fuel clad starts oxidizing during an accident, temperatures can rise quite rapidly resulting in the intragranular releases from central regions of the core at the same time as intergranular gases are being released from more peripheral regions of the core. Some of these effects can be seen in Figure 49. This figure represents the VERCORS RT6 NG releases (from Pontillon 2018). This figure shows both the longer-lived ^{85}Kr which resides both in the grains and in the grain boundaries and the shorter-lived ^{133}Xe which resides primarily within grains. The first two ^{85}Kr peaks correspond to those in the GASPARD tests. Relevant in VERCORS RT6 results is that, as temperatures continue to rise, that NG releases continue to increase and that the intragrain NGs start to be substantially released, and that substantial intragrain releases, indicated ^{133}Xe release, start before the intergrain releases have completed.

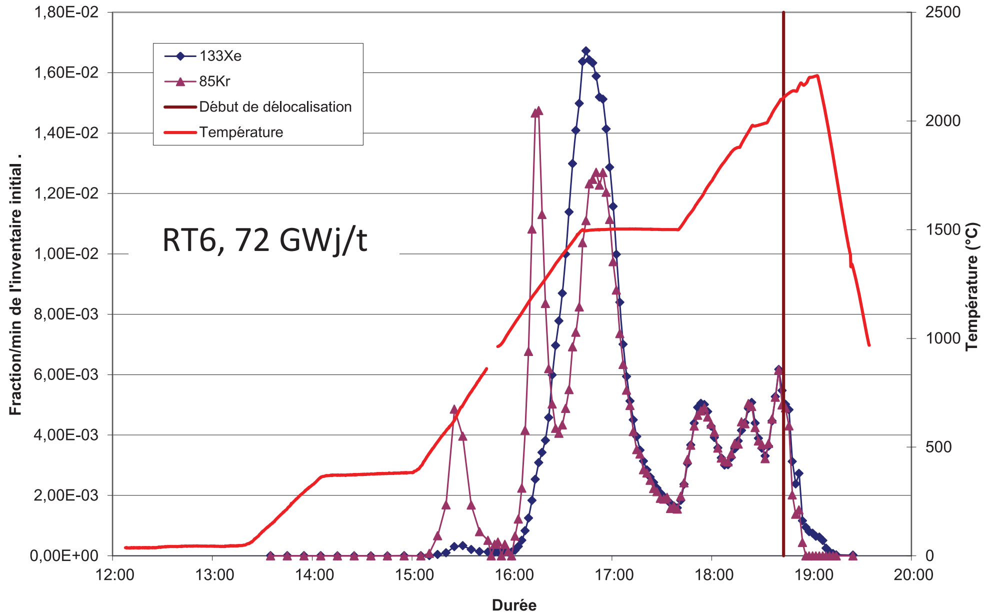

Figure 49 VERCORS-RT6 72GWd/t NG Release Rates, From [Pontillon, 2018]

Appendix C: FFRD in ESF leakage dryout calculation review

J. Metcalf's provided a scoping analysis on the potential of dryout of fragments during ESF leakage [Metcalf, 2021]. Since the document was made available in an unreviewed status, NRC conducted a brief review consistent with its scoping character. M. Salay review of assumptions and approach in J. Metcalf's proposed Analysis of the impact of FFRD on ESF leakage. The review of the assumptions in J. Metcalf's scoping analysis [Metcalf, 2021] of the impact of FFRD on ESF leakage follows. The concern was raised that, if fragment-containing water escaped the piping system, the puddle could potentially evaporate to dryness which could potentially lead to radiological releases from the fragments. Metcalf's analysis focused on whether fragment-containing leaking water would evaporate to dryness. Given that his analysis considered complete release of all expected FFRD as ESF leakage and made further significantly conservative assumptions, the analysis, and the conclusion that evaporation to dryness is unlikely, seems reasonable. It may be good to independently reproduce the calculation to verify the same results are obtained.

Review of proposed calculation assumptions

Metcalf's analysis in regular text. (***M. Salay comments in bold italics.)[<u>J. Metcalf responses bold underlined</u>]***

Question:

What is the potential impact of fuel fragmentation and dispersal on evaporating ESF leakage to dryness and thus greatly increasing iodine partitioning from that leakage?

Assumptions:

1. Core power = 3238 MWt ***(input)***
2. UO_2 mass = 216600 lbm ***(input) (~98,250kg)***
3. Initial coolant mass = 543560 lbm ***(input)***
4. Percentage of core dispersed into coolant as fuel fragments from failed pins = 2% ***(based on other sources and independent calculation, this estimate appears conservative)[<u>2% and 1 mm values were postulated as a result of a brief discussion following a much broader discussion of the ATF PIRT and the observation that particles larger than the characteristic dimension of the ESF leakage path would not be a factor outside containment. Assumptions 6, 9, and 10 are fairly arbitrary. The key parameter is the ratio of coolant to fuel mass leak rate.</u>]***
5. Fragment and minimum ESF leak path diameter = 1 mm ***(particle sizes can be smaller, not sure if it makes a difference if assume the same leakage rate as for a single 1mm leak)***
6. Single ESF leak path ***(could be more leak paths, may not make a difference)***
7. Leakage temperature = 212 F ***(pressurized water at depth can be hotter in which case some water vaporizes upon leak and less mass to evaporate to dry out)***
8. Leakage density = 59.8 lbm/ft^3 ***(~ 0.958 g/cc - OK)***
9. Hydraulic head driving leakage = 100 feet (41.5 psi) ***(Does this represent pressure in containment, depth, or a combination of both? I don't know if these conditions are typical for ESF leakage. Those who review ESF leakage calculations would have a better idea of the representativeness of this value of typical calculations.)***
10. ESF leakage pathway hydraulic loss coefficient = 3 ***(seems to be a reasonable range)***
11. Leakage latent heat of vaporization = 970.32 BTU/lbm ***(~2257.0 kJ/kg - OK)***

12. Maximum total ESF leakage = 2 gpm (0.266 lbm/sec) (Note that this maximum leakage would permit 690740 lbm of coolant to leak in 30 days, 27% more than the initial RCS coolant mass of 543560 lbm.) ***(~0.121kg/s. I don't know if this is typical of ESF leakage calculations)[This is a fairly large number, but not the largest I've ever seen. Keep in mind that a larger ESF leak rate makes more radioiodine available for release; but given the 2% limit on fuel fragmentation, it also makes evaporation to dryness more unlikely. Small leak rates in that sense are worse. BUT, if leak paths become TOO small, they will not pass the particles.]***
13. Fractional decay power = 0.27*(t+108.5)^-0.34 for t in seconds (curve fit of APCSB 9-2) ***(Not sure how the average decay power of fuel fragments compares to the average core decay power. It could be higher or lower. The average power is a reasonable first assumption.)[Agree]***
14. Fuel fragments are uniformly distributed in the initial coolant mass 543560 lbm. A decay power (i.e., fragment concentration) multiplier is incorporated to evaluate skewed distribution (i.e., fuel fragment concentrations greater than uniform). The maximum value for the multiplier is 10 (max concentration = 10 x uniform). The minimum value is assumed to be unity (corresponding to the uniform distribution in the initial coolant mass of 543560 lbm). No values less than unity are included, although they are certainly possible and even likely as fragments deposit within the RCS and are unavailable to be leaked. ***(This depends on geometry of which I'm not very familiar. If leak location (ESF room) is not at a low point in the piping, it seems that assuming evenly distributed fragments (neglecting settling) is conservative. On the other hand, if ESF rooms are at the low point, (or the lowest point), fragments could accumulate. Ten seems like a reasonable starting guess.)[Agree. ESF leakage can originate in the RCS or from the sump. There is also the question of flow rates and entrainment of the fragments (Kutateladze Number) to move the fragments to the leakage site.]***
15. Decay power of the RG 1.183-activity release in the leaked water is neglected. (Justification: By t = 2 hours, the maximum decay power of the RG 1.183 releases - as derived from licensing basis HNO_3 production for G = 0.007 molecule/100 eV - is less than 2% of the full decay power. By t = 8 hours, it is only 0.7% of the full decay power. Moreover, it takes time for the activity to be released and to be deposited in the coolant water whereas the release of the fragmented fuel contribution is assumed to be instantaneous.) ***(The time period over which leakage occurs and over which dryness is evaluated (~30 days) is much greater than the release period. I'm not sure if decay from others sources would be negligible compared to the fragment heat.)[The question of the decay power of the entrained fragments vs. the decay power of the dissolved or suspended fission product release is one that should be confirmed. But I think that one will find that over time, 2% of the core will greatly exceed the decay power of just the RG 1.183 released FPs and actinides.]***

Approach:

1. Establish a lower threshold leakage by calculating leakage from a 1 mm diameter leak path. Smaller leak paths would not pass the 1 mm fuel fragment.
2. The decay power in the leaked coolant is calculated from the expression given in Assumption 13. The Assumption 13 value is multiplied by 0.02 times the fraction of 543560 lbm leaked up to any point in time to represent the decay power of the fragments leaked with the coolant. Per Assumption 14, a concentration multiplier between unity and 10 is used to evaluate non-uniform concentration of fuel fragments in the total leakage.
3. The differential equation that must be solved to determine the time-dependent leaked mass is as follows (for the ESF leak rate, $\dot{m}$, in lbm/sec, m in lbm, and t in seconds):

$$\frac{dm}{dt} = \dot{m} - \frac{0.02 * 0.27 * M * \dot{m} * t * \ 3238\ MWt * 948\,\frac{BTU}{sec - MWt}}{543560\ lbm * 970.32\,\frac{BTU}{lbm}} * (t + 108.5)^{-0.34}$$

where M is the multiplier used to evaluate the effect of non-uniform distribution of the fuel fragments in the coolant mass. When M*ṁ*t exceeds 543560 lbm, the fraction M*ṁ*t/543560 is set to unity so that the decay power in the accumulated leakage does not exceed the maximum value of 2% of the instantaneous value of the Assumption 13 curve fit. ***("Leaked mass" is leaked water mass – evaporated water mass where the evaporation rate is evaluated as the decay power within the fragments. It neglects additional evaporation if relative humidity is low (OK since room with hot water spraying into it and a hot puddle likely has a high relative humidity) and also conservatively neglects replenishment of water by condensation on walls.)[Agree]***

Solving the DE for the appropriate range of total ESF leakage rates and decay power multipliers will allow one to determine if the accumulated leakage mass, m, ever becomes zero indicating "evaporation-to-dryness". ***(A scenario that might lead to dryness could be if a leak becomes blocked after some loss.)[Yes, but the radioiodine inventory available for release would also be capped. Interesting trade-off.]***

Summary of Comments on Calculation

Metcalf concludes that:

> It seems to me unlikely that a sufficient number of particles could be deposited in ESF leakage to bringabout evaporation-to-dryness. Leak paths would have to be few and relatively large and particle concentrations in the leakage would have to be well above uniform to bring that phenomenon about.

The following summarizes some of the main comments in the review:

A higher initial water temperature is possible at depth. More water would vaporize upon leaking at higher temperatures. Despite this, the calculation seems quite conservative.

Some water vapor would condense on surfaces and likely replenish puddle. The analysis conservatively neglects condensation.

The analysis assumes all fragment decay power reaches ESF water and leaks. Depending on piping geometry, this could be very conservative.

The decay power fit was checked against APCSB 9-2 (NUREG-0800 Standard Review Plan (Formerly NUREG-75/087)). As can be seen in Figure 17*, the fit predicts greater decay than ANS94+2σRMS from 10 to 10000s and greater decay than APCSBX1.2 from 10,000 to 1E7s. The fit is conservative compared to the recommended approaches.*

The higher assumed fragment to water concentration ratio (10 x high relative to uniformly distributed in coolant) results in less water to evaporate than if the leaked fragments were assumed to be evenly distributed in water.

Not sure if it's OK to neglect nominal decay power from leakage of substances other than fragments. The argument may have been that it's negligible. I don't know if it is. ***[Agree - should be confirmed]***

Worst case to potentially evaporate to dryness would be if the leak stops early. It would be interesting to evaluate how particles would distribute and how hot they would get if the puddle did dry out. ***[Or what the effect would be if fragments were trapped in the leak path.]***

The assumptions, approach, analysis, and conclusions seem reasonable. The analysis involves several conservatisms and seems to be quite conservative overall. If ESF water temperature is higher than 100 °C (212 °F), if significant additional heat is added by nominal RG1.183 release, and/or if concentrations can go higher and then leakage stops, dryness may be predicted for bounding calculations – but likely not if condensation (heat transfer from walls and ceilings) is considered. Effect of replenishment by condensation would be limited if room is ventilated or open to the environment.

A similar dryout analysis was independently conducted both numerically and analytically using SI units. An equation of the same form as Metcalf's was derived. The trends with the fragment concentration multiplication factor and with leakage flow rate in both the numerical and analytical evaluations matched those in Metcalf's analysis: lower leak rates and higher fragment concentrations increased the likelihood of dryout. Dryout of leaked ESF water is far less likely if less than the assumed 2% fuel reaches the coolant or leaks along with coolant.

[Lower leak rates mean that less radioiodine is available for release. Also, if lower leak rates mean smaller leak paths, then the 2% fragment release begins to look less likely as only smaller fragments can pass through with the coolant.]

Additional potential checks of the ESF analysis

The following additional checks of these analyses could be conducted: ***[Good.]***

- *Check to see if fragmented regions have higher or lower power than average. (maybe not necessary unless close to a limit. This doesn't seem likely to change results significantly.)*
- *Impact of a higher initial water temperature (> 100 °C).****[If ESF areas are at atmospheric pressure, won't flashing just occur and lower the temperature? Remember, flashing fractions are considered in radioiodine release analyses.]***
- *Impact of distributed nominal RG1.183 decay heat. (How does it compare to the fragment heat? May be small compared to fragment heat, especially with the multiplier. One of the things of interest is how FFRD may affect radiological release. If the effect of FFRD is much smaller than for nominal RG1.183 ESF leakage calculations, it may not be necessary to consider it.)*
- *Check impact of cessation of leakage. If flow stops, will water dry out?*

Appendix D: High Burnup Containment Source Term Development

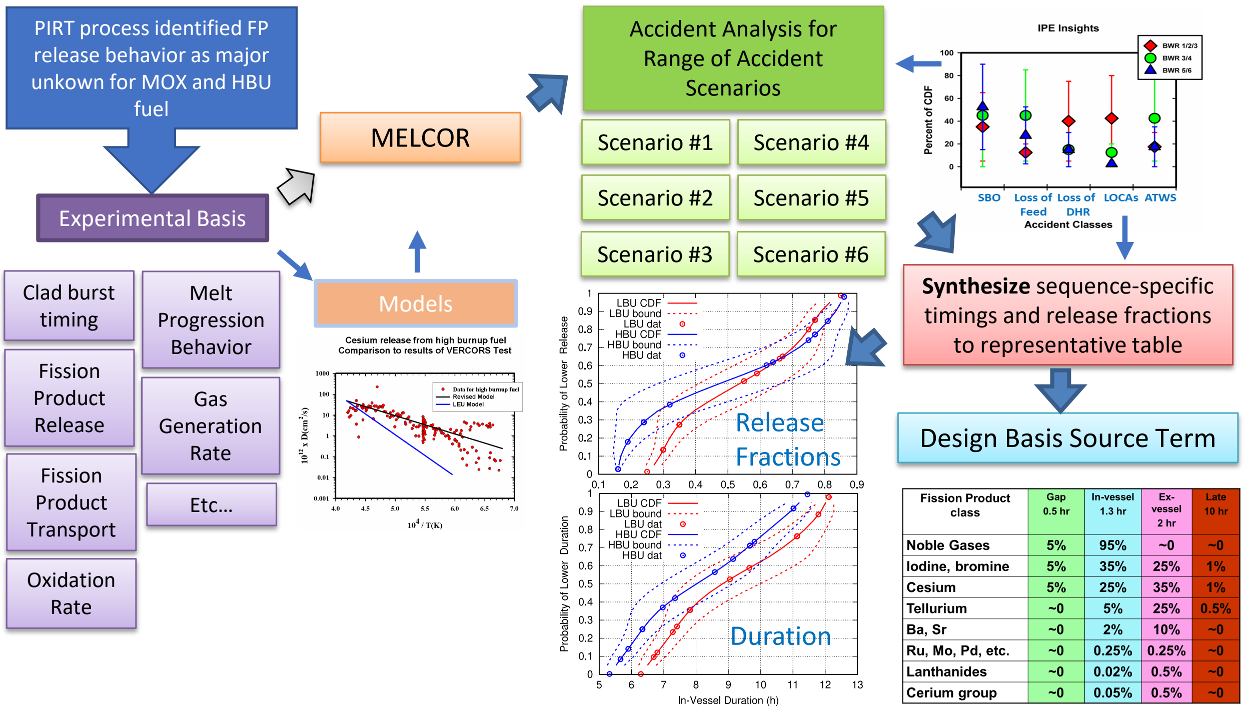

Figure 50 Development Process for the Containment Source Term defined in the SAND2011-0128 report

The development of the 2011 High-Burnup and MOX source term followed the same general approach as that for the development of NUREG-1465. **Error! Reference source not found.** Depicts the general process of developing a Containment Source Term using mechanistic codes. The process involves:

- Conduct Expert Elicitation – PIRT Process
- Evaluate acceptability of existing models and defining data needs
- Experimentation to characterize relevant phenomena
- Develop physical models
- Add physical models to MELCOR and validate
- Use risk analyses and Individual Plant Examinations to define a set of representative sequences that cover CDF
- Evaluate sequence set in MELCOR and extract radionuclide-specific release behavior to containment
- Synthesize the results into representative table of accident-phase timings and release fractions: Containment Source Term

The NUREG-1465 source term was developed by analyzing several accident sequences using the Source Term Code Package (STCP), which is a series of mechanistic codes linked together, and aggregating results into a simplified representative source term to containment that can be used with simpler codes or even hand calculations to evaluate design-basis doses for siting and control-room habitability. This source term can also be used for the environmental qualification of equipment in containment. These codes were developed following the Three Mile Island accident for the purpose of calculating more representative severe-accident and radionuclide release behavior.

The source term prior to NUREG-1465, TID-14844, was based, not on code analyses, but on the results of heating irradiated fuel in a furnace with the release being assumed to instantly be available to containment while assuming some retention in the RCS. Although release magnitudes did not differ tremendously, the NUREG-1465 source term providing relief in the timing of the release.

Sandia performed a reanalysis of source terms for high-burnup and MOX fuel using the MELCOR code. This work is summarized in [Powers, 2011]. The objective of this source term reanalysis was similar to that of NUREG-1465: Define a "representative" source term to the containment for the evaluation of defense-in-depth capabilities with the following properties:

- must be characteristic of accidents involving melting of a substantial fraction of the reactor core,
- should not be deliberately bounding, and
- should be generically applicable to the type of reactor (BWR or PWR).

As in the development of the NUREG-1465 it was not considered necessary for the source term to represent a self-consistent accident sequence.

The motivation for the BWR reanalysis was that NUREG-1465 was based on analyses of plants with fuel used to burnups < 40 GWd/t. Most plants at the time take fuel to > 50 GWd/t. The applicable regulatory limit is 62 GWd/t. Rim effect and changes in pellet/clad interactions are found for burnups in excess of about 45 GWd/t. This reanalysis evaluated the Containment Source Term for higher burnups.

For this work Sandia used a similar approach and sequence evaluations to that used in NUREG-1465. These sequences used in this reanalysis are shown below. Since the specific purpose of these analyses was to develop source terms for containment, they did not include bypass events. The starting plant decks used for the source-term analysis were the same ones used for the SOARCA effort.

Table 14. Accidents considered in development of source terms for high burnup fuels in BWRs using the MELCOR code (from Powers, et al., "Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel", Sandia National Laboratories, SAND2011-0128,2011)

#	Containment	Accident Initiator	Other Failures	Containment Failure
1	Mark I	Short-term station blackout	No coolant injection; no DC power; low vessel pressure	Early liner melt
2	Mark I	Short-term station blackout	No coolant injection; no DC power; low vessel pressure	Early liner melt; high vessel pressure
3	Mark I	Short-term station blackout	No coolant injection; no DC power; low vessel pressure	Late head flange leakage
4	Mark I	Short-term station blackout	No coolant injection; no DC power; high vessel pressure	Early liner melt

5	Mark I	Long-term station blackout	RCIC operates; DC power lost after 8 hours; stuck open safety relief valve	Early liner melt
6	Mark I	Long-term station blackout	RCIC operates; DC power lost after 8 hours; stuck open safety relief valve	Late head flange leakage
7	Mark I	Long-term station blackout	RCIC operates; DC power lost after 8 hours; stuck open safety relief valve	Late over-pressure of torus
8	Mark I	Small break; relief valve 'tee'	No coolant injection; low vessel pressure	Early head flange leakage
9	Mark I	Recirculation suction line break	No coolant injection; low vessel pressure	Early dry-well liner melt
10	Mark III	Short-term station blackout	No DC power; no coolant injection; stuck open safety relief valve	Early; H_2 burn at vessel breach
11	Mark III	Short-term station blackout	No DC power; no coolant injection; high vessel pressure	Early; H_2 burn at vessel breach
12	Mark III	Short-term station blackout	No DC power; no coolant injection; stuck open safety relief valve	Late over pressure
13	Mark III	Long-term station blackout	RCIC operates; loss of DC power at 8 hours; stuck open safety relief valve	Early; H_2 burn at vessel breach
14	Mark III	Long-term station blackout	RCIC operates; loss of DC power at 8 hours; stuck open safety relief valve	Late over pressure
15	Mark III	ATWS	All coolant injection fails following containment failure	Prior to core damage
16	Mark III	Recirculation suction line break	No coolant injection; low vessel pressure	Late overpressure

These sequences were chosen so as to cover a large fraction of the core damage frequency. Within the range of results found in IPE studies for the various types of plants accidents of the types of interest make similar contributions to the overall core damage frequency. This can be seen in **Error! Reference source not found.** and in **Error! Reference source not found.**.

Figure 51 Proportion of IPE core-damage frequency of the simulations considered in the MELCOR re-evaluation of the Containment Source Term for high-burnup and mixed-oxide fuels

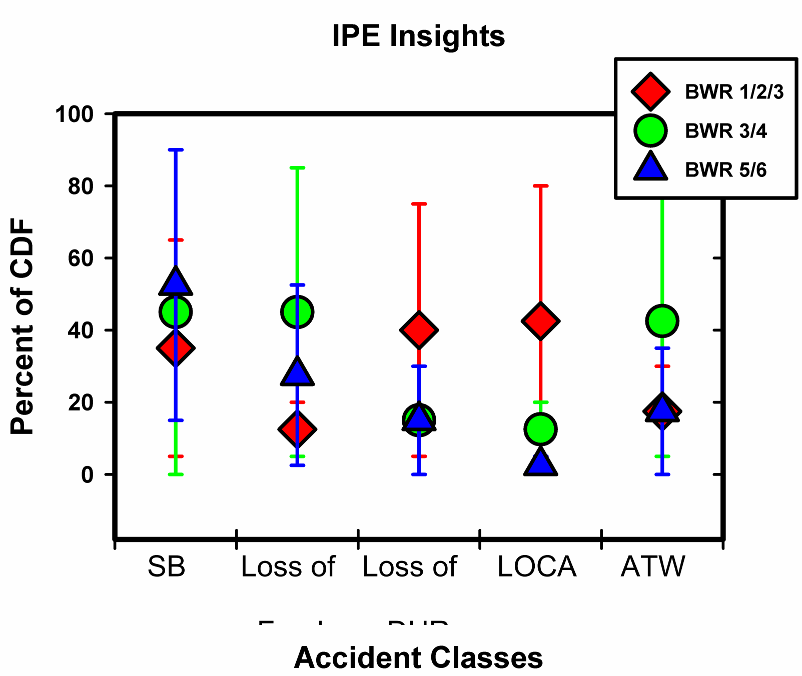

Table 15 Sequence IPE core damage frequencies for two reactors

Peach Bottom

CDF = 5.34×10^{-5}

Initiator	CDF	%
LOCA	0.44×10^{-6}	10
ATWS	0.32×10^{-6}	8
Loss of offsite power	0.49×10^{-6}	11
Transient	3.02×10^{-6}	70
Coverage = 98%		

Grand Gulf

CDF = 1.72×10^{-5}

Initiator	CDF	%
LOCA	0.38×10^{-6}	2
ATWS	0.05×10^{-6}	0.3
Station Blackout	7.45×10^{-6}	43
Transient	9.34×10^{-6}	54
Coverage = 99%		

Table 16. Accidents considered in the development of the source term for high burnup fuel in PWRs using the MELCOR code (from Powers, et al., "Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel", Sandia National Laboratories, SAND2011-0128,2011)

#	Containment	Accident Initiator	Other Failures
1	Subatmospheric	Station blackout	No ECCS; No auxiliary feedwater; induced RCP seal LOCA
2	Subatmospheric	Small break	No ECCS; Auxiliary feedwater available; late containment failure
3	Subatmospheric	Large break	ECCS injection
4	Subatmospheric	Station blackout	No ECCS; No auxiliary feedwater
5	Subatmospheric	Small break	No ECCS; auxiliary feedwater available; early containment failure
6	Ice condenser	RCP seal failure	No ECCS; auxiliary feedwater available; reactor cavity flooding
7	Ice condenser	RCP seal failure	No ECCS; auxiliary feedwater available
8	Ice condenser	RCP seal failure	ECCS injects; auxiliary feedwater available
9	Ice condenser	Station blackout	No ECCS
10	Ice condenser	Station blackout	No ECCS; no auxiliary feedwater
11	Ice condenser	Large break	No ECCS; auxiliary feedwater available
12	Ice condenser	Small break	No ECCS; no auxiliary feedwater

For the development of the NUREG-1465 BWR source term BNL analyzed many sequences using the STCP code package [Nourbakhsh, 1993]. The decks used for the development of the NUREG-1465 source term were the same as those used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants". The accident sequences simulated in this analysis are listed.

BWR Mark I (Peach Bottom) accident sequences analyzed during the development of the NUREG-1465 source term

- TC1 - An anticipated transient without scram accompanied by the failure to achieve early power reduction but successful depressurization of the primary system.
- TC2 - A variation of TC1 sequence, the failure to scram is accompanied by failure to achieve early power reduction and the failure to achieve emergency depressurization.
- TC3 - A variation of TC2 with containment venting in the wetwell gas space.
- TB1 - Loss of all offsite and onsite AC power accompanied by loss of all active engineered safety features except the steam powered emergency core cooling systems. The latter, however, would fail when the station batteries are depleted (6 hours after start of accident).
- TB2 - A variation of TB1 with containment failure due to rapid pressurization following failure of the reactor vessel.
- S2E1 - A small break (2" in diameter) LOCA accompanied by the complete failure of the emergency core cooling systems. For the purpose of this analysis the Automatic Depressurization System (ADS) was not actuated.
- S2E2 - A variation of S2E1 assuming a basaltic concrete composition.
- V - A rupture in the low-pressure emergency core cooling system piping in the reactor building outside the primary containment envelope.
- TBUX - A station blackout initiated by a loss of all DC power. The operators are assumed to be unable to depressurize the reactor vessel because DC power is unavailable.

BWR Mark II (LaSalle) accident sequence analyzed during the development of the NUREG-1465 source term

- TB - A station blackout accident with late containment failure mode.

BWR Mark III (Grand Gulf) accident sequences analyzed during the development of the NUREG-1465 source term

- TC - An anticipated transient without scram. The containment was assumed to fall by over-pressurization prior to core melting due to elevated power input to the suppression pool; containment failure was assumed to lead to failure of emergency core cooling system pumps.
- TB1 - Loss of all AC power accompanied by loss of all active engineered safety features with the exception of the steam-turbine driven emergency core cooling systems. The latter, however, would fail when the station batteries are depleted (6 hours after start of accident).
- TB2 - A variation of TB1 with containment failure due to hydrogen burn following failure of the reactor vessel.
- TBS - Loss of AC power accompanied by loss of all active engineered safety features. However, the operator was assumed to successfully depressurize the primary system.
- TBR - A variation of TBS except that electric power is reestablished through and thus the sprays in containment operate.

Subatmospheric PWR (Surry) accident sequences analyzed during the development of the NUREG-1465 source term

- AG - hot leg LOCA, no containment heat removal systems
- TMLB' – Loss of Offsite Power, no Power Conversion System and no Auxiliary Feedwater
- V - Interfacing system LOCA
- S3B – Station Blackout with RCP seal LOCA
- S2D-d – Small Break LOCA, no ECCS and H2 combustion
- S2D-b - Small Break LOCA with 6" hole in containment

Large Dry PWR (Zion) accident sequences analyzed during the development of the NUREG-1465 source term

- S2DCR - LOCA (2"), no ECCS no Containment Spray Recirculation System
- S2DCF1 - LOCA RCP seal, no ECCS, no containment sprays, no coolers-H2 burn or DCH fails containment
- S2DCF2 - S2DCF1 except late H2 or overpressure failure of containment
- TMLU - Transient, no Power Conversion System, no ECCS, no Auxiliary feed water-DCH fails containment

Large Dry PWR (Oconee 3) accident sequences analyzed during the development of the NUREG-1465 source term

- TMLB' – Station Blackout, no active ESF systems
- S1DCF - LOCA (3"), no ESF systems

Ice Condenser PWR (Sequoyah) accident sequences analyzed during the development of the NUREG-1465 source term

- S3HF1 - LOCA RCP seal, no ECCS, no Containment Spray Recirculation System with reactor cavity flooded
- S3HF2 - S3HF1 with hot leg induced LOCA
- 3HF3 - S3HF1 with dry reactor cavity M3B LOCA (3") with Station Blackout
- TBA - SBO induces hot leg LOCA-hydrogen burn fails containment
- ACD LOCA (hot leg), no ECCS no Containment Spray
- S3B1 – Station Blackout delayed 4 RCP seal failures, only steam driven Auxiliary Feedwater operates
- S3HF - LOCA (RCP seal), no ECCS, no Containment Spray Recirculation System
- S3H - LOCA (RCP seal) no ECC recirculation

Processing of results

The MELCOR HBU-MOX reanalysis were processed into a representative source term in a manner similar to that performed during the development of the NUREG-1465 source term. The processing of MELCOR results involved 50th percentile values for release fractions. The NRC chose 70th percentile release fractions to develop NUREG-1465.

Code output includes pressures, temperatures, fluid levels, equipment states, core degradation and radionuclides release and transport behavior.

Figure 52, Figure 53, and Figure 54 show select output for the Surry and Sequoyah station blackout (SBO) scenarios used for the SAND2011-0128 source term. These figures provide information about the type of information and type of output that is processed in the development of the Containment Source Terms.

In addition to evaluating Thermal Hydraulic behavior including fluid properties and temperatures, MELCOR evaluates clad oxidation, core degradation, fission product release and transport as vapors and aerosols including revaporization of deposits, component failure, combustible gas combustion, and ex-vessel behavior including core-concrete interactions (CCI) and radionuclide release during CCI.

v shows the MELCOR calculated reactor water level with event annotations, for one of the Surry SBO scenarios that contributed to the SAND2011-0128 source term. Figure 53 shows the clad temperature for the same sequence. Heat up rates increase dramatically once clad oxidation begins. One can see in this figure that high clad temperatures (and thus releases) don't start for several hours.

Figure 54 shows the disposition of CsI in the core for a Sequoyah station blackout sequence. This sequence proceeds faster than the Surry one due to a different assumption of auxiliary feedwater availability. This figure shows the amount of Cs and I released from fuel as CsI, the fraction in the vessel, in the RCS, in containment, and the environment. MELCOR tracks RNs in different chemical groups.

These figures originate from SAND2008-6664 PWR [Ashbaugh, 2008]. The sequences described therein serves as input for the SAND2011-0128 source term.

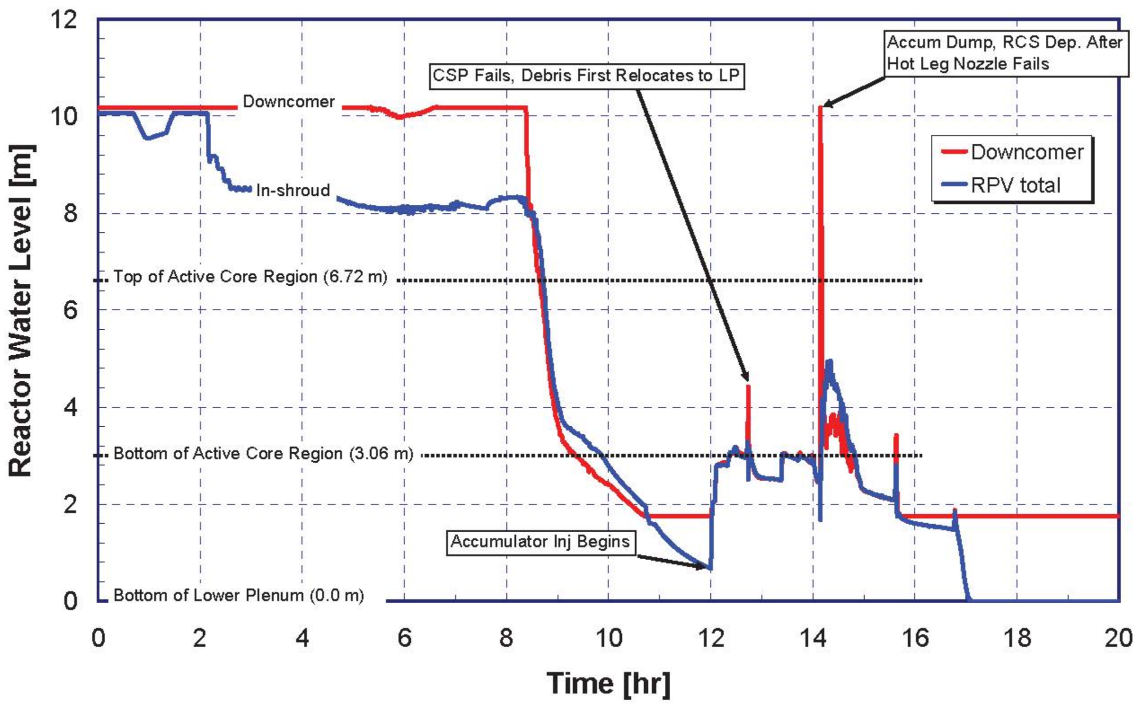

Figure 52 Surry SBO Reactor Water level, with event annotations. From [Ashbaugh, 2008]

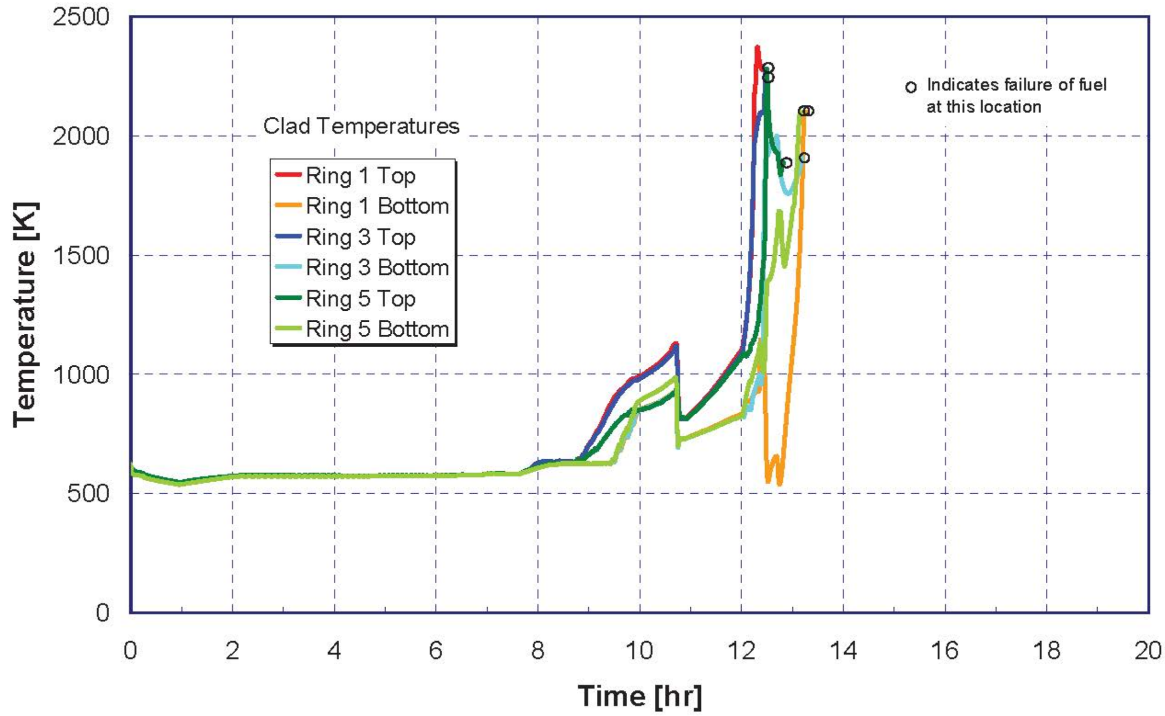

Figure 53 Surry Station Blackout Clad T. From [Ashbaugh, 2008]

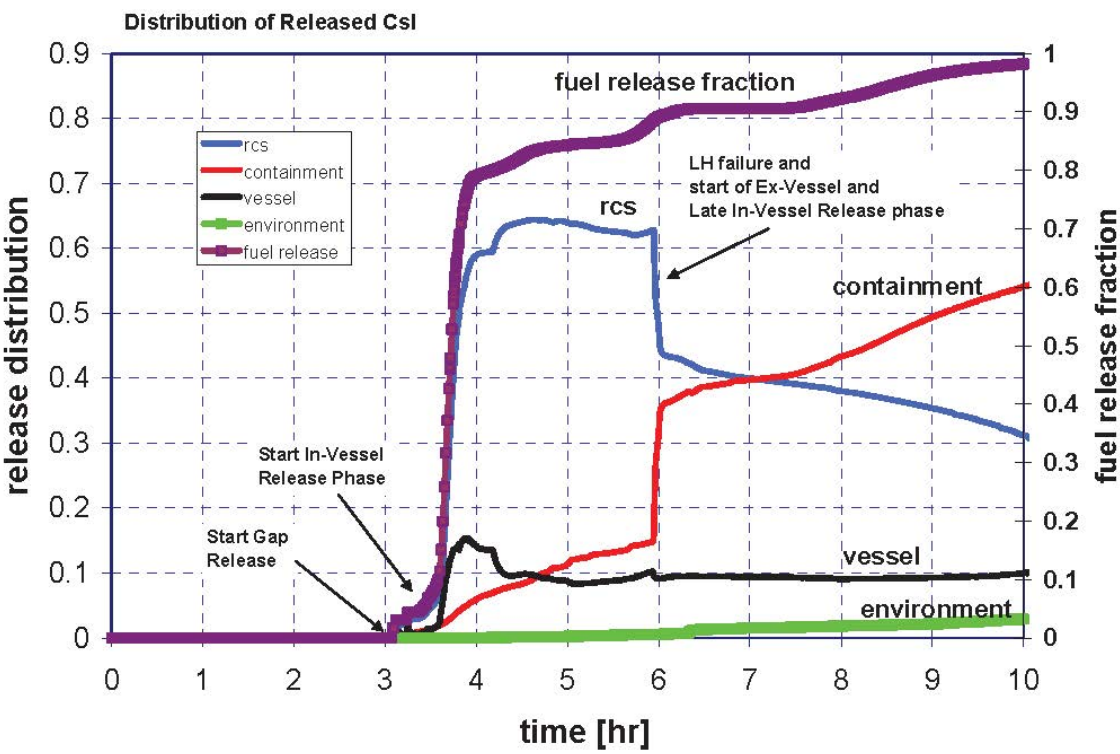

Figure 54 Distribution of CsI and Cs2MoO4 in the Sequoyah analysis. From [Ashbaugh, 2008]

The radionuclide release to Containment for each radionuclide chemical group for each sequence in each plant are synthesized into a single table. Only a high-level summary is provided here. [Powers, 2011] provides details on the approach.

The results of the HBU-MOX reanalysis (radionuclide group release histories) were split into the different release phases in the same manner as was done for NUREG-1465. Differences in the results required some assumptions to be made to fit the results in the same phases.

The results were processed as follows: Determine criteria for accident phase splitting, determine phase timings for each accident sequence based on the selected criteria, generate phase-timing distributions for each phase, generate RN-class-specific release fractions for each phase, and select the chosen percentile from each distribution to come up with a representative source term table similar to the NUREG-1465 source-term tables and therefore also the AST/ RG1.183 LOCA tables. The resultant source term tables covering PWRs and BWRs using low-burnup, high-burnup, and mixed-oxide fuels can be found in the Sandia report. The resultant BWR source term table including source terms for high-burnup and low-burnup fuel from this report is shown in Table 2. The corresponding PWR source term table is shown in Table 3.

As an example of the distributions developed for In-Vessel Release Duration and Halogen release fraction from the MELCOR results for a PWRs, from fwhich the Containment Source Term tables are derived, are shown in Figure 55 and Figure 56 below (data from SAND2011-0128).

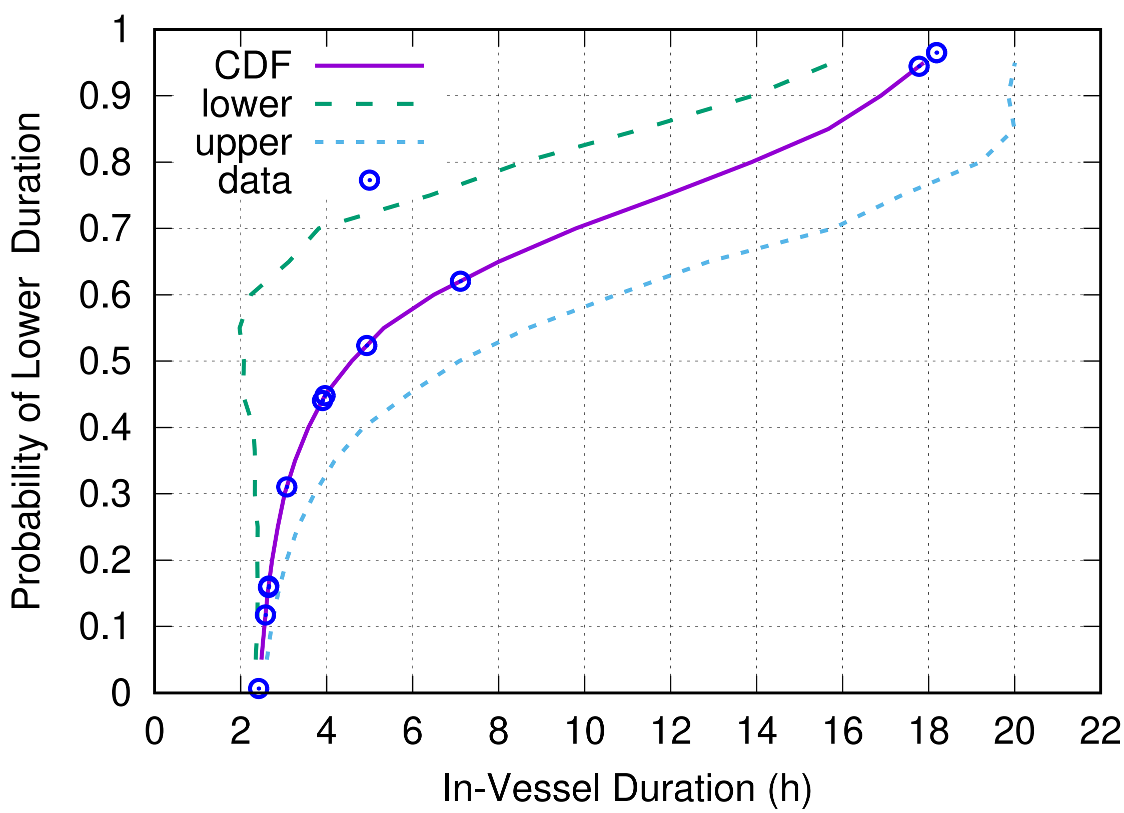

Figure 55 Example distribution of the duration of the in-vessel release phase of accidents in two PWRs using high burnup fuel. The plot shows the results of accident analyses, the cumulative distribution function, and error bounds of one standard deviation. Data from [Powers, 2011]

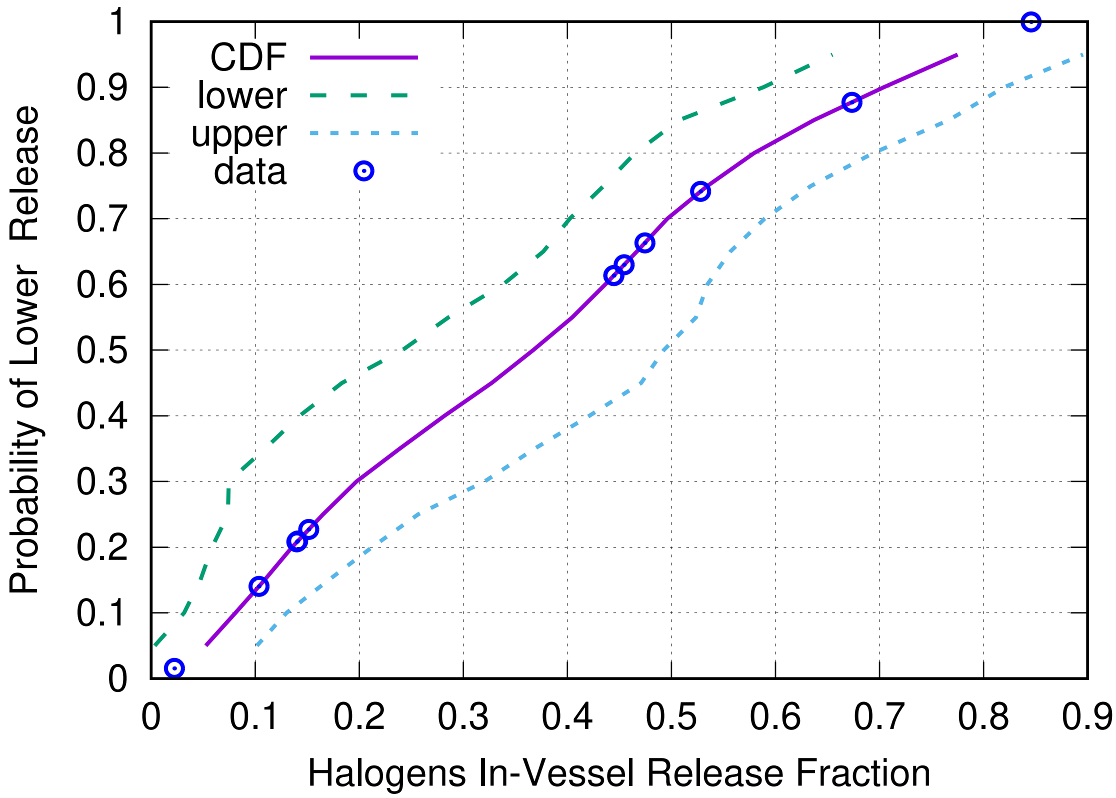

Figure 56 : Example distribution of the halogen fraction released to containment during the in-vessel release phase of accidents in two PWRs using high burnup fuel. The plot shows the results of accident analyses, the cumulative distribution function, and error bounds of one standard deviation. Data from [Powers, 2011].

The median timings and the median RFs (SAND2011-0128) or 70th percentile RFs (NUREG-1465) on these distributions are extracted to form the Alternative Source Term tables, RG1.183 Tables 1, 2, and 4 and their equivalent in SAND2011-0128.