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Accident Source Terms for Boiling Water Reactors with High Burnup Cores Calculated Using MELCOR 1.8.5

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Abstract

The primary objective of this report is to provide the technical basis for development of recommendations for updates to the NUREG-1465 Source Term for BWRs that will extend its applicability to accidents involving high burnup (HBU) cores. However, a secondary objective is to re-examine the fundamental characteristics of the prescription for fission product release to containment described by NUREG-1465. This secondary objective is motivated by an interest to understand the extent to which research into the release and behaviors of radionuclides under accident conditions has altered best-estimate calculations of the integral response of BWRs to severe core damage sequences and the resulting radiological source terms to containment. This report, therefore, documents specific results of fission product source term analyses that will form the basis for the HBU supplement to NUREG-1465. However, commentary is also provided on observed differences between the composite results of the source term calculations performed here and those reflected NUREG-1465 itself.

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Abbreviations

ADS	Automatic Depressurization System
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CF	Containment Failure
CFR	Code of Federal Regulations
DW	Drywell
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
GE	General Electric
HBU	High Burnup
HP	High Pressure
HPCI	High Pressure Coolant Injection (system)
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems Loss of Coolant Accident
LBU	Low Burnup
LLOCA	Large-break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LT-SBO	Long-Term Station Blackout
LWR	Light Water Reactor
MCCI	Molten Core-Concrete Interaction
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ORNL	Oak Ridge National Laboratory
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling (system)
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RHR	Residual Heat Removal (system)
RPF	Relative Power Fraction
SLOCA	Small-break Loss of Coolant Accident
SBO	Station Blackout
SOSRV	Stuck-Open Safety/Relief Valve
ST-SBO	Short-Term Station Blackout
STCP	Source Term Code Package
TAF	Top of Active Fuel
TID	Technical Information Document
TMI	Three Mile Island
VF	Vessel Failure

1.0 Introduction and Background

1.1 Use of Regulatory Source Terms

Estimation of the consequences of postulated accidental releases of radioactive materials is mandated in the U.S. Nuclear Regulatory Commission (NRC) policies and practices by 10 CFR Part 100 [Ref.1]. The NRC's reactor siting criteria have required, for licensing purposes, that applicants consider accidental fission product releases resulting from a "substantial meltdown" of the reactor core into the reactor containment. The applicant must assess the potential radiological consequences of this event assuming that the containment remains intact though it leaks at its maximum allowable rate. The radioactive material that leaks from the containment is called the "radiological release to the environment." This release of radioactive material is obtained from the containment leak rate and the inventory of radioactive material suspended in the containment atmosphere as a function of time.

The radioactive material suspended in the containment can be in the form of gases or aerosol particles. Together these suspended radioactive materials are referred to as the "in-containment accident source term." The suspended inventory of radioactive materials will be a function of time. It will depend on the rates of radioactive material releases from the core as well as the performance of engineered safety features such as containment sprays, as well as natural processes that remove radioactive vapors and aerosols from the containment atmosphere.

Most currently operating plants were licensed and operated originally based on the specifications of the release from the core found in Regulatory Guides 1.3 and 1.4 [Refs.2,3]. These specifications were derived from the 1962 report TID-14844 [Ref.4], which described fission product release based on very early studies involving heated, irradiated, UO_2 pellets. The derived source term was composed of 100% of the fuel inventory of noble gases and 50% of the fuel inventory of iodine (half of which was assumed to deposit very rapidly on surfaces). Regulatory Guides 1.3 and 1.4 specified that this source term be instantaneously available in the reactor containment. Furthermore, the regulatory guides specified that 91% of the iodine be present in the form of molecular iodine (I_2), 5% as particulate iodine (such as CsI) and 4% as organic iodine vapor (such as $CH_3I(g)$). These assumptions concerning the timing and chemical form of the source term have affected the design of engineered safety features and required closure times for containment isolation valves.

Use of the postulated accident source term has not been confined to evaluations of site suitability and the designs of engineered safety systems such as sprays and filtration systems. The regulatory applications of the source term have included evaluations of the post-accident environment for qualification of safety-related equipment, post-accident control room habitability requirements, and post-accident sampling systems and accessibility.

Following the reactor accident at Three Mile Island (TMI), the NRC sponsored an extensive research report to better understand the physical and chemical processes associated with accident involving "substantial meltdown" of the core including the releases of radionuclides and the transport of these radionuclides from the point of release to the containment. These studies

showed that releases and transport of radionuclides depended to a significant extent on the details of phenomena involved in the accident [Ref.5]. These studies showed that many more radionuclides could be involved in the source term than had previously been considered. Much of the radionuclide release was in the form of aerosol particulate and substantial retention of these particulate could occur by natural processes along the tortuous pathway from the point of release to the containment.

Based on the extensive understanding developed in the research, NRC developed an alternative accident source term [Ref.6]. This alternative is often called the "NUREG-1465 Source Term." The NUREG-1465 Source Term considers both the timing and the chemical composition of the source term in a great deal more detail than past studies. Releases from the degrading reactor fuel are divided into five phases, as shown in Figure 1.

Five Severe Accident Release Phases as Defined in NUREG-1465

Coolant Activity Release	Begins with a postulated pipe rupture Ends when first fuel rod fails
Gap Activity Release	Begins when fuel cladding failure commences Ends when fuel pellet bulk temperature sufficiently high such that fuel cannot retain fission products
Early In-Vessel Release	Begins at the end of the gap release phase (fuel cannot retain fission products) Ends when the reactor vessel lower head fails
Ex-Vessel Release	Begins when molten core debris exits the reactor vessel Ends when debris cooled sufficiently such that significant fission products releases stop
Late In-Vessel Release	Begins when the reactor vessel lower head fails No definition provided – infer that definition is analogous to end of ex-vessel release phase

Figure 1 NUREG-1465 Severe Accident Release Phases

Each of these phases has a specified duration and involves the release of specified fractions of the radionuclide inventory. Because of differences in accident progression in plants of different design, different specifications are provided for PWRs and BWRs. The specifications themselves were derived from the results of many accident sequences for a variety of representative plants using the Source Term Code Package (STCP) and early versions of the MELCOR accident analysis code.

The coolant activity release is the expulsion of radioactive coolant into the containment that occurs early in an accident before fuel significantly overheats. The gap release phase occurs once fuel is no longer covered by coolant and begins to overheat. It is expected that the zirconium alloy cladding on the fuel will expand and rupture venting radionuclides that have accumulated in the fuel-cladding gap and in the near surface interstices of the fuel. If the accident cannot be arrested at this point, then the fuel continues to heat and radionuclides diffuse from the fuel and vaporize. The heatup of the fuel may be augmented significantly by the exothermic reaction of steam with the zirconium alloy cladding. Eventually, residual metal

cladding will melt and begin dissolving fuel. This dissolution will further affect radionuclide release.

Radionuclides vaporized from the fuel will pass out of the core region into cooler parts of the reactor coolant system (RCS). The vapors will condense and form aerosol particles. Both aerosol particles and vapors have opportunities to deposit on surfaces along this flow path. The NUREG-1465 Source Term specifies the net effect of release and successful passage of radionuclides through the RCS to the containment.

The ex-vessel accident release phase occurs when relocated fuel and clad penetrates the reactor vessel and cascades into the reactor cavity. The processes contributing to the ex-vessel release include the potentially pressurized expulsion of melt from the vessel and the subsequent interactions of the core debris with concrete. Pressurized expulsion of core debris from the reactor vessel can occur only if the vessel remains pressurized throughout the degradation process. At the time the NUREG-1465 Source Term was developed, it was thought that for many risk important accidents, especially at pressurized water reactors, vessel pressurization could be maintained throughout the degradation process. Releases associated with core debris interactions with concrete depend significantly on the amounts of metallic zirconium still present in the core debris, and to a lesser extent, on the nature of concrete used in the construction of the nuclear power plant.

Late in-vessel release occurs because substantial amounts of radioactive material released during the core degradation process are retained on surfaces within the RCS. The continued radioactive decay of these retained materials causes the surfaces to heat. Eventually, temperatures are sufficiently high that considerable revaporization of deposited radionuclides into the natural circulation of gases through the ruptured RCS can occur. The revaporization from surfaces is slow and occurs over a protracted period. It sustains the period over which there is substantial inventory of radioactive material suspended in the reactor containment atmosphere.

The NUREG-1465 Source Term groups radionuclides released during accidents into eight groups based on the similarities of chemistry. These groups are shown in Table 1. The fractional releases of the initial core inventories of these groups for accidents at BWRs are shown in Table 2. Release rates in each of the phases are assumed to be constant over the specified durations.

Table 1 NUREG-1465 Radionuclide Groups

Radionuclide Group	Title	Elements in Group
1	Noble Gases	Xe, Kr
2	Halogens	I, Br
3	Alkali Metals	Cs, Rb
4	Tellurium Group	Te, Sb, Se
5	Barium, Strontium Group	Ba, Sr
6	Noble Metals	Ru, Rh, Pd, Mo, Tc, Co

7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8	Cerium Group	Ce, Pu, Np

Table 2 NUREG-1465 Source Term to Containment for BWRs*

	Gap Release***	In-vessel	Ex-vessel	Late In-vessel
Duration (hours)	0.5	1.5	3.0	10.0
Noble Gases**	0.05	0.95	0	0
Halogens	0.05	0.25	0.30	0.01
Alkali Metals	0.05	0.20	0.35	0.01
Tellurium Group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Lanthanides	0	0.0002	0.005	0
Cerium Group	0	0.0005	0.005	0

* Values shown are fractions of initial core inventory
**See Table 1 for a listing of the elements in each group
***Gap release is 3% if long term fuel cooling is maintained

The NUREG-1465 Source Term does not go to great lengths to specify the chemical or physical forms of most of the released radionuclides. It does, of course, assume that noble gases (Xe and Kr) are gases. Most other radionuclides are assumed to be in particulate form by the time they reach the reactor containment. The size distribution of the particulate, shape factors, densities and the like are not specified. Iodine is assumed to be predominantly (95%) in the form of aerosol particulate. Still, 5% of the iodine released to the containment is taken to be in gaseous form.

The behavior of radionuclides in the containment will be affected by nonradioactive materials, which are also involved in core degradation. The behavior of fission product aerosol is especially affected by the nonradioactive aerosol produced in the accident. This nonradioactive aerosol can include control rod materials, alloying agents from the cladding and other structural materials. Additionally, it can include nonradioactive aerosol produced during the interactions of core debris with structural concrete. The NUREG-1465 Source Term does not attempt to estimate the nonradioactive materials released to the containment, but it does caution that the nonradioactive materials need to be taken into account in estimating the time dependent concentration of radioactive materials in the containment atmosphere.

The authors of the NUREG-1465 Source Term were cautious about its applicability. They restricted application of the source term to hypothesized accidents at currently operating light water reactors. They encouraged designers of advanced reactors to use similar methods to develop source terms applicable to their novel designs. The authors also restricted application of the source term to reactions using low-enrichment uranium dioxide fuel taken to burnups typical of the time when the source term was developed. This burnup was usually less than 40 MWd/kgU.

The NUREG-1465 Source Term is now recognized in the regulatory process as an acceptable alternative to either replace or supplement the source term used in the original licensing of a currently operating nuclear power plant [Ref.7].

1.2 Research Insights Since Publication of NUREG-1465

Research into the release and behaviors of radionuclides under accident conditions continued after the publication of the NUREG-1465 Source Term. Three developments are noteworthy. The first is that the STCP consisting of a suite of phenomenological codes has been replaced by integrated accident analysis codes. These codes have both refined numerics and refined fidelity to accident phenomena. The MELCOR code used here [Ref.8] is a noteworthy example.

A second development has been the continued experimental investigation of accident phenomena. The most important of the continued studies is the PHÉBUS-FP project carried out by an international consortium at the Cadarache Centre in France. These experiments involved bundles of one meter long rods of irradiated fuel heated neutronically in steam through the point of fuel liquefaction and relocation. Radionuclides released during core degradation were allowed to transport through a model of the RCS that included a representation of a steam generator tube. Released radionuclides that successfully negotiated passage through this model of the RCS escaped into a model of a reactor containment. Five data sets suitable for validation of accident analysis models have been produced in this program. A benchmark study of the predictions of integrated accident analysis computer codes and one of these data sets has been published [Ref.9].

A third important development has been the refocus of attention on the behavior of iodine under accident conditions. The PHÉBUS-FP tests have confirmed that some fraction of the iodine will be released to the containment from the RCS as gaseous iodine. The precise fraction has not been established but is not believed to be inconsistent with the 5% assumed in the NUREG-1465 Source Term. The rest of the iodine is released as particulate, though the chemical form adopted by iodine in the particulate may not be entirely CsI as had been assumed in the past. The subsequent behavior of iodine in the containment is proving complicated. A number of irradiated tests in the RTF facility [Ref.10] as well as laboratory tests have been conducted and have supported the development of mechanistic models of iodine chemistry in the containment that are still being researched.

1.3 Evolution of the Nuclear Industry Since Publication of NUREG-1465

The nuclear power industry has evolved since the publication of the NUREG-1465 Source Term. Two aspects of this evolution are pertinent to the development of a high burnup (HBU) supplement to the NUREG-1465 Source Term. The first has been the extension of the licenses of many of the currently operating plants for an additional 20 years of power operation. License extension is expected to affect most of the currently operating plants. License extension has made it economically feasible for plants to revise their licensing bases. In doing so, plants can take advantage of the revised accident source term articulated in NUREG-1465. Many have

chosen to do so. By far, the most common change made in licensing bases is to revise the source term timing based on NUREG-1465.

In light of the increasing use of the NUREG-1465 Source Term, the importance of its applicability to current-generation reactor operation should be highlighted. The second important aspect of industry evolution is directly related to its applicability. Reactors are using fuels to higher burnups than was foreseen when analyses were performed that form the bases of NUREG-1465. At that time, end of life fuel burnups were usually less than 40 MWd/kgU. Today, fuels are being taken to higher burnups approaching the regulatory limit, which is a maximum rod burnup of 62 MWd/kgU. This corresponds to approximately 59 MWd/kgU on an assembly average basis. This increase in burnup was accomplished, in part with significant advances in fuel design features. Important aspects of these advances include:

- Significant changes in the mechanical design of fuel bundles have been made to increase neutronic performance without adversely impacting bundle heat transfer. Many operating BWRs in the U.S. began operation with fuel bundles consisting of a square array of 8x8 rods, some of which were perforated rods filled with water. Over the past 20 years, this configuration has given way to 9x9 and now 10x10 rod configurations. The configuration of water rods within the bundles has also changed from two or more small rods distributed within the fuel rod matrix to a relatively large water rods or channel in the center of the bundle. These differences result in relatively small changes in the overall mass of UO_2 in the core (± 2%), but significant changes in bundle heat transfer surface area (e.g., an increase of approx. 20% from a typical 8x8 to a typical 10x10 configuration).
- Power-shaping has been refined through the use of variable enrichment and burnable poison (gadolinia) concentration and recently through the use of partial-length fuel rods in BWR fuel bundles. Increasing gadolinia concentration and/or reducing enrichment in the lower region of BWR fuel bundles tends to flatten the axial power shape because the axial power distribution tends to be skewed toward the bottom of the core where moderator void content is lower.
- HBU core designs have higher bundle-average U-235 enrichment than low burnup (LBU) designs.

Differences in these design features associated with a typical LBU versus HBU core have been incorporated into the MELCOR core models for the calculations described here, as well as the ORIGEN decay heat and fission product inventory calculations, as discussed in Section 2.3.

2.0 Objectives and Scope

2.1 General

The objective of this report is to provide the technical basis for development of recommendations for updates to the NUREG-1465 Source Term for BWRs that will extend its applicability to accidents involving HBU cores. The same philosophy and general approach used to develop the NUREG-1465 Source Term is applied here in the development of the HBU supplement. That is, results obtained with mechanistic accident analysis computer codes are used to capture the major relevant insights on the phenomenology of radionuclide release and transport during accidents involving substantial meltdown of the core. This report documents the results of the accident analyses that will form the basis for the HBU supplement to NUREG-1465. However, commentary on applicability of results to all postulated severe accidents for all BWR types, as well as final recommendations for the HBU supplement to NUREG-1465 will be provided elsewhere.

The accidents selected for the analysis include those that have been determined in Probabilistic Risk Assessment (PRA) to be significant contributors to the estimated frequency of core damage events in PWR or BWR nuclear plants, as discussed in Section 3. There is not, however, an intent to produce a "bounding" source term. The reliance on mechanistic analyses of risk important accidents has been adopted to present a more realistic portrayal of the amounts of radioactive material present in containment for use in regulatory processes that entail consideration of a substantial core meltdown. It should be noted as well, that current consensus is that Large-break LOCA (LLOCA) accidents are not risk-significant for BWRs. However, calculations for LLOCA sequences have been performed and will be factored into the overall development of the HBU supplemental source term. The use of the LLOCA calculations should extend the applicability of the source term to design basis accident analysis and provide some measure of conservatism to the severe accident source terms (primarily in terms of timing).

The supplemental source term for accidents involving HBU cores will be cast in a form similar to that adopted for the NUREG-1465 Source Term. For consistency with existing regulatory guides, the HBU source term will necessarily be expressed in terms of times and rates of appearance of radionuclides into the containment, the types and quantities of species released and other important attributes (e.g., the chemical forms of iodine). Releases to the environment from the containment are not considered here.

The intent of this work is to define the changes in the NUREG-1465 Source Term caused by the extension of fuels in LWRs from the 40 MWd/kgU cited in NUREG-1465 to the regulatory limit of 59 MWd/kgU. There has been, however, a continuing evolution and refinement of accident modeling as discussed in Section 1.2. Some of these developments have been discussed by an expert opinion elicitation on the subject [Ref.11]. In fact, insights from the expert elicitation are applied in the selection of accident sequences that form the basis for the HBU supplement, as described in Section 4.

2.2 The Reactors

The NUREG-1465 Source Term was developed for generic applicability, separately, to PWRs and BWRs of the types currently operating in the U.S. The HBU supplement will continue the practice of development of a generic source term (for the two general classes of plants) based on a representative sample of mechanistic calculations. While the rationale for the selection of plant models and accident sequences is deferred to Section 4, the plant models used are described below. The BWR reactors selected to form the basis for the HBU supplement to the NUREG-1465 Source Term are Peach Bottom Atomic Power Station and the Grand Gulf Nuclear Station.

Peach Bottom

The Peach Bottom Atomic Power Station consists of two operating BWR units located southeastern Pennsylvania, on the western shore of Conowingo Pond. Peach Bottom Unit 2 and Unit 3 began commercial operation in 1974. The site is also the location of a small, decommissioned high temperature gas cooled reactor (Unit 1). Peach Bottom Units 2 and 3 are each a General Electric nuclear steam supply system (NSSS) of the BWR/4 product line housed within a Mark I containment. A photo of the Peach Bottom site is shown in Figure 2. The Peach Bottom reactors were originally licensed to operate at a rated power of 3293 MWth. Applications for power uprates were approved in 1994[1] and 2002, which raised their rated power to the current value of 3514 MWth.

Figure 2 Peach Bottom Atomic Power Station

The Peach Bottom MELCOR model used for the HBU Source Term analyses has been used extensively by various U.S. NRC research programs, and is currently maintained by Sandia National Laboratories as a state-of-the-art MELCOR model representing BWR/4 Mark I reactor/containment systems [Ref. 12]. A brief description of the Peach Bottom MELCOR model can be found in Appendix B.

[1] The initial uprate application for Unit 2 was approved in 1994; the parallel uprate application for Unit 3 was approved in 1995.

Grand Gulf

The Grand Gulf Nuclear Station consists of one operating BWR unit located on the eastern shore of the Mississippi River approximately 25 miles south of Vicksburg, MS. Grand Gulf Unit 1 began commercial operation in 1985. Grand Gulf is a General Electric NSSS of the BWR/6 product line housed within a Mark III containment. A photo of the Grand Gulf site is shown in Figure 3. Grand Gulf was originally licensed to operate with a rated power of 3833 MWth. A power uprate in 2002 increased the rated power to 3898 MWth.

Figure 3 Grand Gulf Nuclear Station

The Grand Gulf MELCOR model used for the HBU Source Term analyses was developed from a model originally used by the NRC for BWR loss of coolant analyses [Refs.13, 14]. The model was subsequently updated and expanded to address a wider spectrum of postulated transient events in a BWR/6 Mark III plant, including the analysis discussed here [Ref. 15]. A brief description of the Grand Gulf MELCOR model can be found in Appendix B.

2.3 The Cores

The design characteristics of fuel assemblies manufactured for modern reactor operating cycles were updated for the four plants modeled in this study. Previous MELCOR models had been based on older design information, none of which represented a current operating configuration. Fuel assembly and core design information was also obtained for recent cycles, and the MELCOR COR model was updated to reflect current assembly geometry and mass composition. Further, fuel loading patterns and assembly power and burnup histories were obtained in order to calculate a reasonably accurate spatial distribution of fission product inventories and associated decay heat. While the bulk of the MELCOR model descriptions are included in Appendix B, rigorous representation of plant- and cycle-specific core and assembly geometries and the level of detail with which fission product inventories and the associated decay heat are modeled, is a first-of-a-kind MELCOR application. Therefore, specifics regarding these portions of the MELCOR model for the BWRs are provided in the following paragraphs.

To discern whether observed differences between the NUREG-1465 Source Term and the HBU supplement are entirely due to changes in core configuration and burnup levels (as opposed to

advancements in the state-of-the-art in severe accident modeling), a LBU version of each plant MELCOR model was also created. These models were based on older, but retrievable, information on assembly/bundle design, core loading, and power/burnup histories for early operating cycles in the reference plants. Again, specifics are provided below.

In BWRs, the mechanical design of an assembly used in early low burnup core designs differs substantially from the mechanical design of assemblies designed for longer operating cycles and higher discharge burnup. Therefore, MELCOR COR input representing fuel assembly geometry, fuel mass, cladding mass, control poison mass, nose piece and upper/lower tie plates material and mass and other important physical characteristics differed in the LBU and HBU calculations. In both cases, input data were derived from vendor-specific design information for a typical assembly used during reactor operation at each of the two levels of discharge burnup of interest to this study. The fuel designs that formed the basis for the MELCOR COR models for the two BWR plants are listed in Table 3. Assembly design data are not included in this report because some data were obtained from proprietary fuel vendor documents. However, comparisons of general characteristics of evolving BWR fuel designs are compared in the relevant plant Final Safety Analysis Reports (FSAR) [Refs. 16, 17].

Table 3 Core Assembly Designs Represented in BWR MELCOR Models – part 1

Plant	Core Model	Fuel Assembly Design
Peach Bottom	LBU	GE 8x8
	HBU	GE14C (10x10)
Grand Gulf	LBU	GE 8x8
	HBU	Atrium-10 (10x10)

In addition to fuel mechanical design information, data describing core axial and radial power profiles, as well as end-of-cycle fission product mass inventories and decay heat information were required. Core power profiles and other data needed to calculated fission product inventories and decay heat histories were developed from plant-specific nuclear design reports obtained from the licensees. Core design and performance information was examined for three consecutive cycles, during early and recent periods of reactor operation. Cycle-specific data for only one of the two plants (Grand Gulf) was available at the time ORIGEN calculations were performed to characterize end-of-cycle fission product inventories and post shutdown decay heat histories. ORIGEN calculations for both plants were, therefore, based on reactor physics data obtained from Grand Gulf[2]. However, unique characteristics of Peach Bottom were taken into account by using plant-specific information on fuel design (e.g., slightly smaller core size and the use of GE14 fuel in Peach Bottom rather than Atrium-10 fuel in Grand Gulf for the HBU analysis).

[2] Core performance data was obtained from Peach Bottom after the ORIGEN calculations were completed [Ref. 18]. A direct comparison of key ORIGEN input parameters, such as axial/radial power profiles, confirmed sufficient similarity to the Grand Gulf data used in these calculations.

Axial and radial power profiles for LBU and HBU core designs were developed by examining data obtained from the licensee for three consecutive early cycles and three recent cycles [Ref. 19]. Specifically, the LBU conditions were derived from data for Grand Gulf cycles 3, 4 and 5. The core-average discharge burnup for these cycles spanned a range from 26 MWd/kgU to 38 MWd/kgU. Data from cycles 11, 12 and 13 were then used to derive properties of an HBU cycle[3]. The maximum assembly-average discharge burnup from these cycles was relatively constant at 45 to 48 MWd/kgU. These power profiles were used in the ORIGEN calculations of fission product mass inventories and decay heat, and in the MELCOR analyses of fuel damage progression.

Representative end-of-cycle radial power distributions were used to define the radial boundaries for the spatial nodalization used in the MELCOR model. The relationship between actual radial power distribution and the MELCOR radial core nodalization is shown in Figure 4 for Peach Bottom and Figure 5 for Grand Gulf. The radial boundaries of the MELCOR core model enclose radial 'rings' of assemblies with a similar relative power fraction (RPF). The average RPFs for each ring in the LBU and HBU models are listed in Table 4.

Table 4 Core Assembly Designs Represented in BWR MELCOR Models – part 2

	Number of Assemblies in Ring		Relative Power Fraction	
Ring	**Peach Bottom**	**Grand Gulf**	**LBU**	**HBU**
1	112	112	1.200	1.207
2	160	160	1.212	1.243
3	200	220	1.181	1.174
4	168	156	0.910	0.827
5	124	152	0.447	0.369
Total	764	800		

The axial power profiles for the LBU and HBU operating cycles are compared in Figure 6. In both cases, the axial power profile within the central four rings of the core can reasonably be represented by a single curve. The fifth (peripheral) ring, however, differs sufficiently from the center to warrant its own profile. At all locations, the LBU and HBU power profiles are very similar.

[3] Average fuel discharge exposures for these cycles are well below the regulatory limit of 62 MWd/kgU. However, as noted in the discussion of axial and radial power profiles, very little difference in key calculation input parameters were noted between the LBU (early cycle) and HBU (later cycle) data from the licensee. The ORIGEN calculations extrapolated the actual exposure history for these cycles out to the regulatory limit to properly calculate end-of-cycle fission product mass inventories and post-scram decay heat levels.

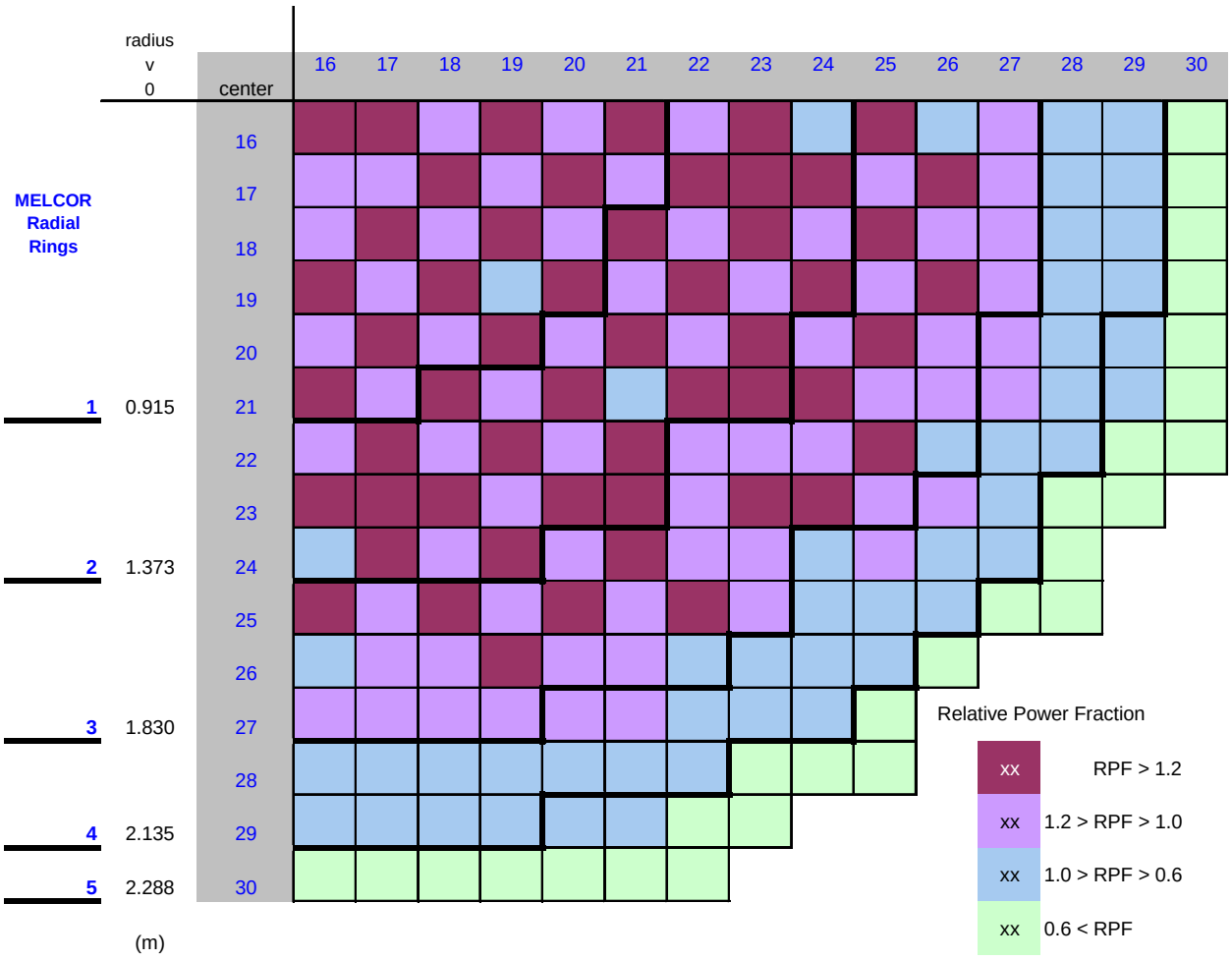

Figure 4 Radial Power Distribution and MELCOR Nodalization for Peach Bottom

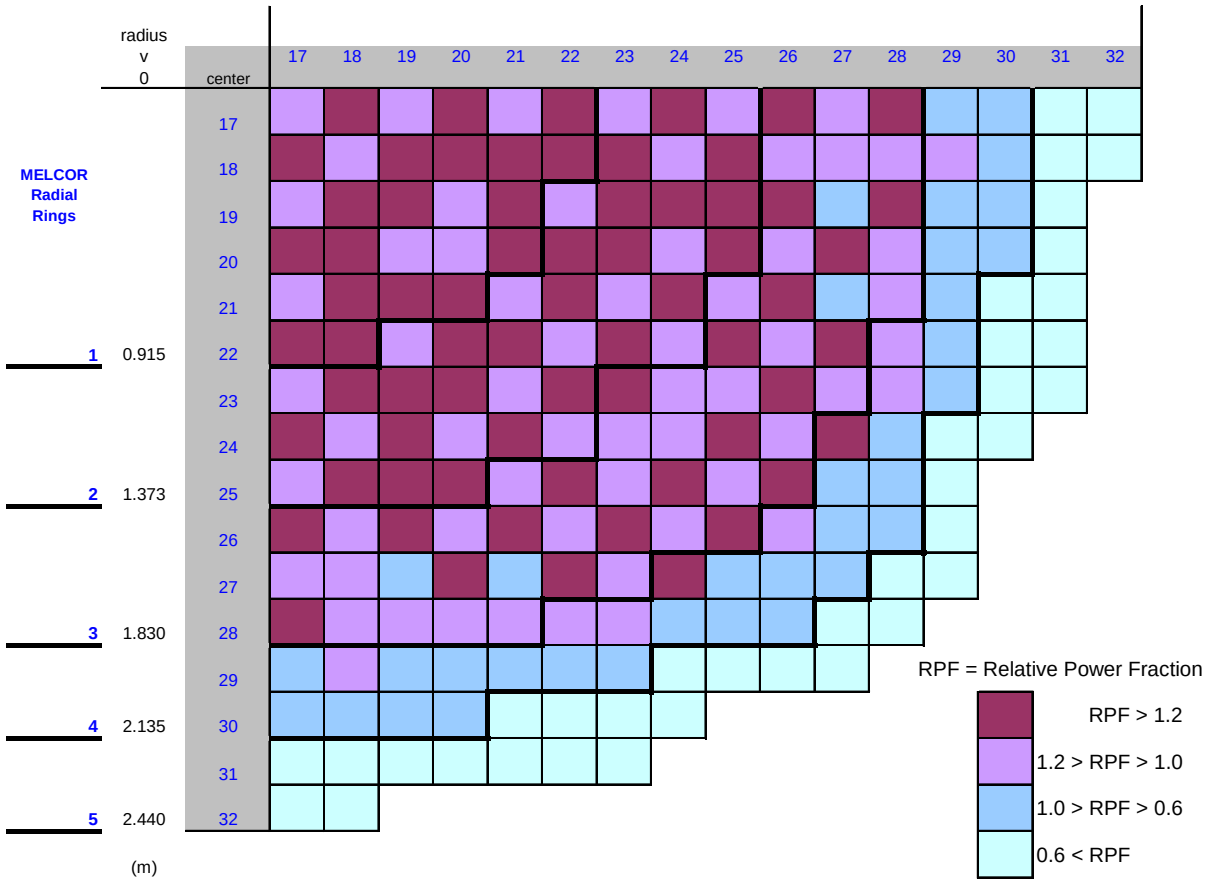

Figure 5 Radial Power Distribution and MELCOR Nodalization for Grand Gulf

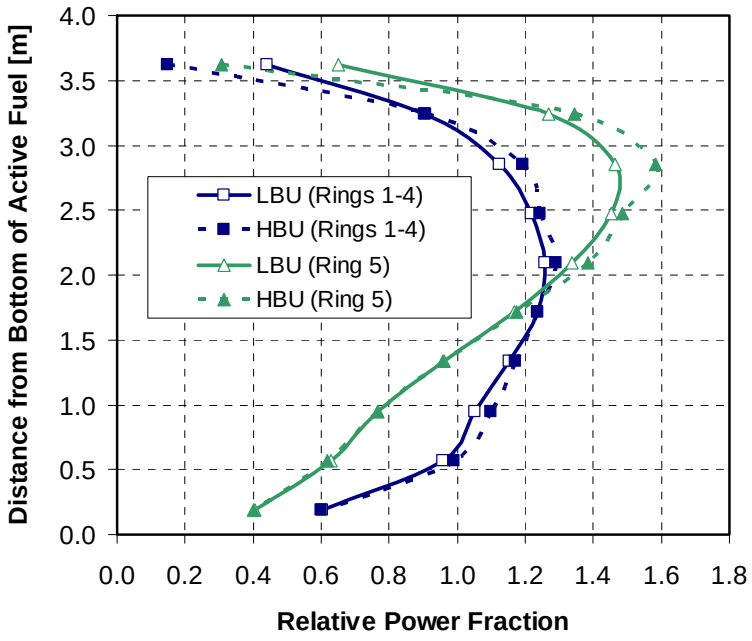

Figure 6 Axial Power Profiles from Early and Recent Fuel Cycles in Grand Gulf

The core power data described along with companion data regarding fuel enrichment, fresh feed loading and partially-burned assembly movements, core power histories were used to perform ORIGEN calculations of fuel depletion and fission product generation. These calculations were performed by Oak Ridge National Laboratory (ORNL) for the two core configurations described above [Ref. 20]. It was noted earlier that the input data used to perform the ORIGEN calculations for the HBU core were based on actual, current plant operating conditions, which represent assembly average discharge exposures less than the current regulatory limit of 59 MWd/kgU. However, the HBU ORIGEN calculations extrapolated the actual exposure history to reach the maximum permissible discharge burnup.

The ORIGEN calculations were performed on an assembly-specific basis for different axial locations along each assembly (based on the axial power distribution). This provided spatially-dependent fission product inventories and decay heat histories that were based on plant-specific operational history. This is a significant advancement over prior severe accident modeling practice of using results of whole core calculations of fission product inventories and decay heat[4]. The resulting core fission product inventories used in the MELCOR model are given in Table 5 and Table 6 for Peach Bottom and Grand Gulf, respectively. Only whole core values are shown in these tables, although the spatial distributions from the ORNL analysis were used in the MELCOR input. Whole-core decay heat histories for Peach Bottom and Grand Gulf are listed in Table 7 and Table 8, respectively.

Comparison of the fission product inventory results shows a significantly larger increase in the HBU inventory over the LBU inventory for Peach Bottom than for Grand Gulf. While both HBU models were based on data for very recent cycles and were extrapolated (slightly) to the same level of discharge burnup (59 MWd/kgU), there are several factors that contributed to the larger difference between LBU and HBU inventories in the Peach Bottom calculations versus the Grand Gulf calculations. First, the core average discharge burnup for the LBU core models was approximately 20% lower in Peach Bottom than in Grand Gulf (21 vs 26 MWd/kgU). Therefore, the difference between LBU and HBU discharge burnup was greater in the Peach Bottom calculations. Secondly, the average enrichment represented in the LBU versus HBU fuel in the Peach Bottom model (2.61 to 4.07 %wt ^{235}U) spanned a wider range than the Grand Gulf model (2.66 to 3.74 %wt ^{235}U). Some of this difference in attributable to differences in the types of assemblies used in current (HBU) core designs in Peach Bottom (GE14) versus Grand Gulf (Atrium-10). These factors also lead to a slightly larger increase in decay heat levels in the HBU versus LBU results for Peach Bottom then Grand Gulf, as shown in Table 7 and Table 8, respectively.

[4] The ORIGEN calculations were performed in a spatially-dependent manner to examine the extent to which the spatial distribution of decay heat and/or fission product inventories differed between the LBU and HBU configurations. Parallel calculations were also performed on a whole core (core average) basis to validate the traditional practice of distributing decay heat and radionuclide inventories within the core based on fission power distribution. The spatial distribution of decay heat and radionuclide inventories was confirmed to be consistent with the equilibrium fission power distribution. As noted earlier the LBU and HBU fission power data obtained from the BWR licensees indicate negligible differences.

Table 5 Initial Core Inventories Calculated for Peach Bottom

Radionuclide Group	Elements in Group	LBU Core (kg)	HBU Core (kg)	% Increase
Noble Gases	Xe, Kr	361.8	876.5	142%
Halogens	I, Br	14.0	34.0	143%
Alkali Metals	Cs, Rb	207.8	506.0	144%
Tellurium	Te, Se	33.2	81.5	145%
Alkaline Earths	Ba, Sr	154.1	372.0	141%
Platinoids *	Ru, Pd, Rh	234.3	633.4	170%
Early Transition *	Mo, Tc, Nb	263.7	640.9	143%
Lanthanides	La, Nd, Pr, Sm, Y, Pm, Eu, Am, Gd	485.7	1240.5	155%
Cerium Group	Ce, Pu, Zr, Np	1213.1	2280.6	88%

* The grouping scheme used in NUREG-1465 combines these two groups.

Table 6 Initial Core Inventories Calculated for Grand Gulf

Radionuclide Group	Elements in Group	LBU Core (kg)	HBU Core (kg)	% Increase
Noble Gases	Xe, Kr	472.3	868.4	84%
Halogens	I, Br	17.9	33.3	86%
Alkali Metals	Cs, Rb	273.5	497.1	82%
Tellurium	Te, Se	43.2	80.2	86%
Alkaline Earths	Ba, Sr	202.3	367.3	82%
Platinoids *	Ru, Pd, Rh	302.0	612.7	103%
Early Transition *	Mo, Tc, Nb	346.9	633.2	83%
Lanthanides	La, Nd, Pr, Sm, Y, Pm, Eu, Am, Gd	642.4	1218.9	90%
Cerium Group	Ce, Pu, Zr, Np	1462.5	2253.5	54%

* The grouping scheme used in NUREG-1465 combines these two groups.

Table 7 Initial Whole-Core Decay Heat Calculated for Peach Bottom

Time After Shutdown	LBU Core (MW)	HBU Core (MW)	% Increase
0.0 s	204.0	208.3	2%
1.0 s	188.3	192.5	2%
3.0 s	171.7	175.7	2%
7.0 s	154.7	158.5	2%
13.0 s	141.5	145.1	3%
27.0 s	126.2	129.6	3%
54.0 s	112.2	115.3	3%
1.8 min	98.4	101.2	3%
3.7 min	86.2	88.8	3%
7.4 min	75.4	77.9	3%
14.8 min	64.7	67.0	4%
29.8 min	53.5	55.6	4%
1.0 hr	42.9	44.8	4%
2.0 hr	34.5	36.3	5%
12.0 hr	21.1	23.0	9%
24.0 hr	17.3	19.0	10%
48.0 hr	13.9	15.4	11%
7.0 d	8.3	9.6	16%

Table 8 Initial Whole-Core Decay Heat Calculated for Grand Gulf

Time After Shutdown	LBU Core (MW)	HBU Core (MW)	% Increase
0.0 s	211.9	217.7	3%
1.0 s	195.6	201.2	3%
3.0 s	178.4	183.6	3%
7.0 s	160.8	165.7	3%
13.0 s	147.1	151.7	3%
27.0 s	131.3	135.5	3%
54.0 s	116.7	120.6	3%
1.8 min	102.4	105.9	3%
3.7 min	89.7	92.9	4%
7.4 min	78.5	81.5	4%
14.8 min	67.3	70.0	4%
29.8 min	55.7	58.1	4%
1.0 hr	44.7	46.8	5%
2.0 hr	36.0	37.9	5%
12.0 hr	22.2	23.9	8%
24.0 hr	18.2	19.8	9%
48.0 hr	14.7	16.1	10%
7.0 d	9.0	10.0	11%

2.4 Fission Product Release Kinetics

There are few data on the behavior of HBU fuel under severe reactor accident conditions. Results from a single VERCORS test have recently become available, and insights from these results were used to develop a Cs release rate model that accounts for observed trends from the RT-6 test. The VERCORS RT-6 test [Refs. 21, 22, 23] was performed using fuel from the Fessenheim nuclear power plant in France having a burnup of approximately ~55 MWd/tonne. The experiment measured the release of Cs (among other fission products) from small re-irradiated fuel samples as the temperature of the sample was gradually increased. The RT-6 test was run under oxidizing conditions. Figure 7 shows the measured Cs release from the RT-6 experiment. This data is used to develop release rate parameters for a diffusional release model as described in the following section.

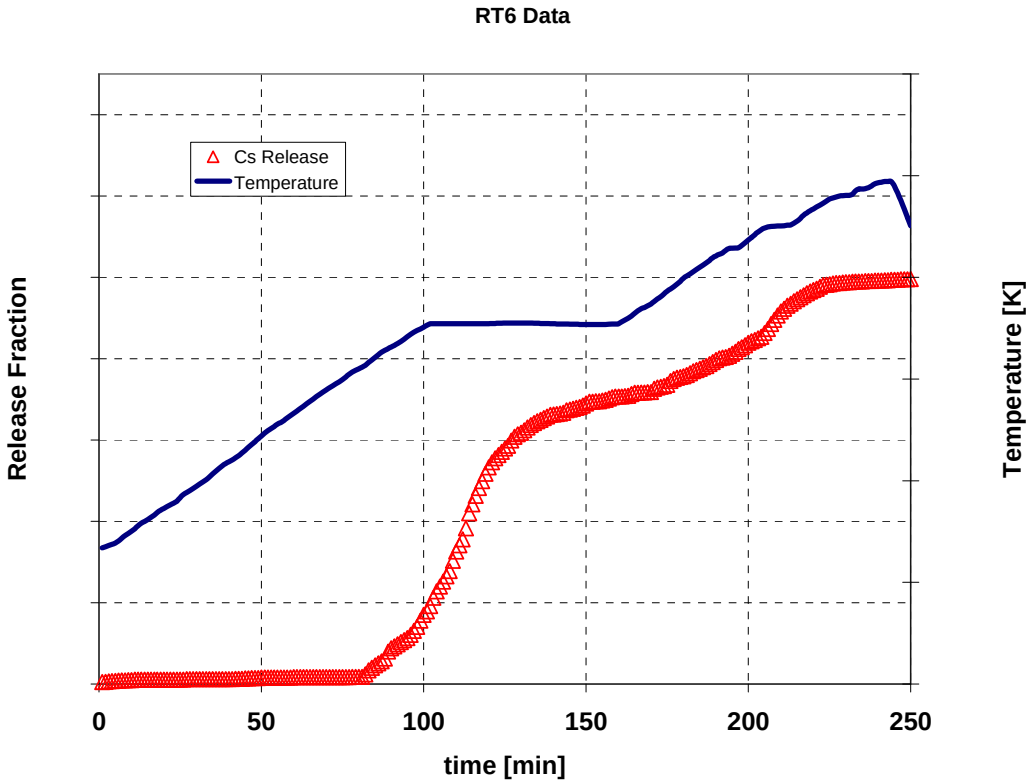

Figure 7 RT 6 Release of Cs as a Function of Test Sample Temperature
(units are suppressed to respect proprietary nature of the test results)

Modeling of Fission Product Releases from LBU and HBU Fuel

The Booth diffusion model is one model available in MELCOR for calculating the release of fission products from overheating fuel; the Booth model is selected for this study because of its more mechanistic nature in comparison to the CORSOR fractional release rate models. In this treatment, the release of Cs is modeled to match the kinetics of the measured release for Cs, and other fission product releases are simply scaled to the Cs release to match those observed experimentally. The Booth release model is described briefly below.

In the Booth model, as implemented in MELCOR, the release of Cs from the fuel is treated as a diffusion process where Cs migrates through the fuel matrix to the surface of a fuel grain. From there, a mass transport limitation based on specie vapor pressure is considered prior to release to the local atmosphere. The effective diffusion coefficient for Cs in the fuel grain is given by

$$D = D_0 \exp\left(-Q/RT\right) \qquad \text{Eq. 1}$$

where R is the universal gas constant, T is the temperature, Q is an activation energy, and the pre-exponential factor D_o is a function of the fuel burn-up. The Cs release fraction at time *t* is

calculated from an approximate solution of the diffusion equation for fuel grains of spherical geometry [Ref. 24],

$$f = 6\sqrt{\frac{D't}{\pi}} - 3D't \quad for\ D't < 1/\pi^2 \qquad \text{Eq. 2}$$

$$f = 1 - \frac{6}{\pi^2}\exp\left(-\pi^2 D't\right) \quad for\ D't > 1/\pi^2 \qquad \text{Eq. 3}$$

where

$D't$	=	Dt/a^2 (dimensionless), and
a	=	equivalent sphere radius for the fuel grain.

The parameters of the diffusion coefficient, D_o and Q, may be determined from experimental data by a fitting process described by Lorenz and Osborne [Ref. 25]. In this process, Eq. 4 and 5 are inverted to solve for the product Dt/a^2, as indicated below.

$$\frac{Dt}{a^2} = \frac{2}{\pi} - \frac{f}{3} - 2\sqrt{\frac{1}{\pi^2} - \frac{f}{3\pi}} \quad for\ f < 0.85 \qquad \text{Eq. 4}$$

$$\frac{Dt}{a^2} = \frac{-1}{\pi^2}\ln\left[\frac{\pi^2(1-f)}{6}\right] \quad for\ f > 0.85 \qquad \text{Eq. 5}$$

where f is the release fraction.

Using the method described above (Eq. 4 and 5), the experimental data may be cast in terms of the apparent instantaneous diffusion coefficient as shown in Figure 8. Also shown in this figure are expressions for the Booth temperature dependent diffusion coefficients for LBU fuel and the HBU fuel. Notice that the HBU data fit follows the trends of the RT-6 data. The parameters of the diffusion coefficient (D_o and Q) were adjusted subjectively to gain both a reasonable fit to the data shown in Figure 8 and the release rate versus temperature measurements shown in Figure 7. The release prediction using the Booth formula for HBU fuel is shown compared to the measured RT-6 data in Figure 9 along with the Booth prediction for LBU fuel under the same temperature history.

As can be seen, the HBU parameters match the initial release trends very well, underestimate the release rate at intermediate temperatures, and again match the release rate at high temperature very well. Note also that the LBU release model significantly underestimates the RT-6 observations. A more complex model for fission product release would be required to improve the intermediate temperature release rate; however, the HBU Booth model captures the important observed trends, namely the onset of the release at lower temperatures than typical for LBU fuels and the completeness of release at high temperature. The parameters used to represent Cs diffusion release from HBU and LEU fuel are summarized in Table 9.

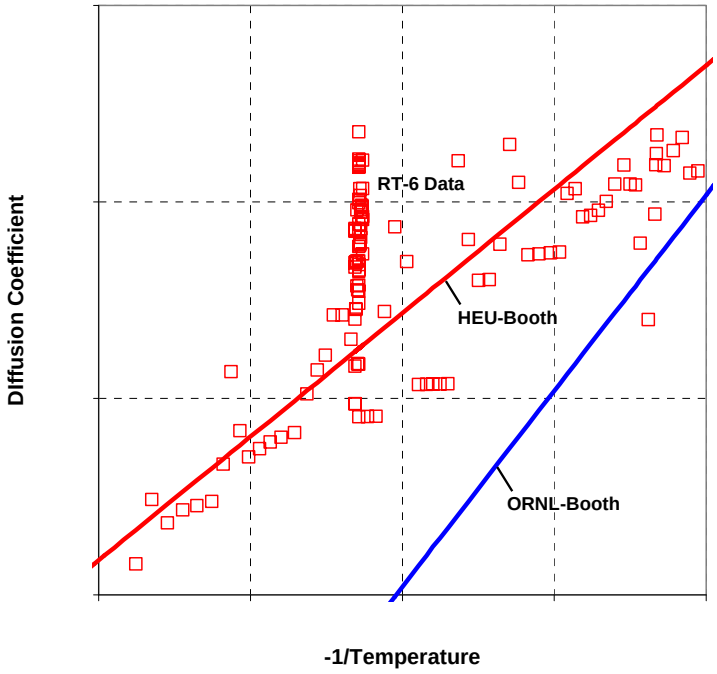

Figure 8 Instantaneous RT-6 diffusion coefficient and Booth model fits
(units are suppressed to respect proprietary nature of the test results)

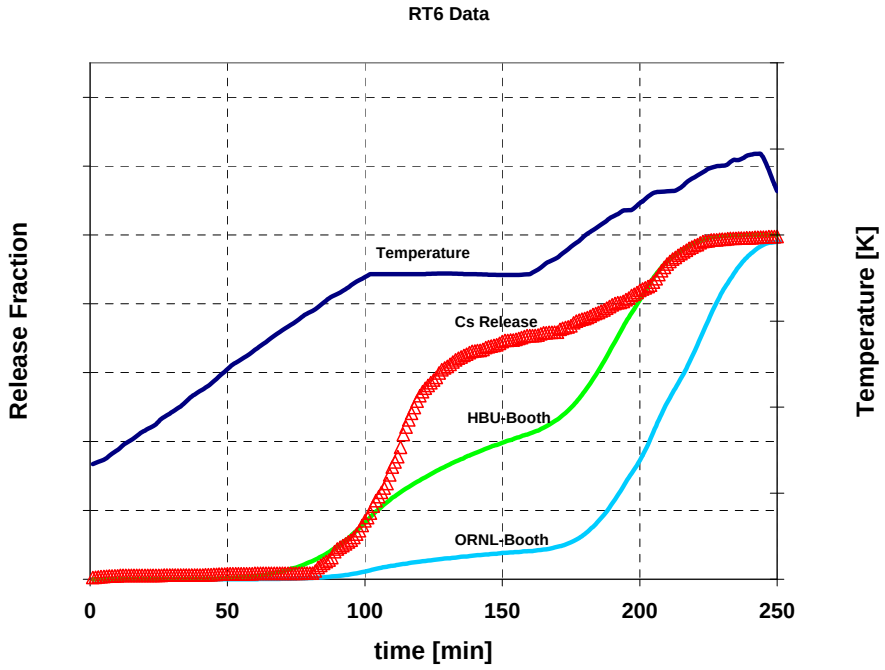

Figure 9 RT-6 release measurements compared to Booth model predictions for HBU and LBU fuels
(units are suppressed to respect proprietary nature of the test results)

Table 9 Parameters for Diffusion Coefficient for HBU and LEU Fuel.

	D_o [m^2/s]	Q [J/kg-mole]
LEU Fuel (ORNL-Booth)	$1x10^{-6}$	$3.814x10^{5}$
HBU Fuel (HBU-Booth)	$2.3x10^{-9}$	$2.411x10^{5}$
Grain radius	6μm	6μm

VERCORS tests involving HBU fuel will be reviewed for insights on melting point and other relevant issues [Refs. 21, 22, 23]. Insights gained may be folded into this HBU source term development in future revisions to this report.

3.0 Assumptions and Limitations of the MELCOR Analyses

3.1 Scope Limitations

As mentioned several times herein, the deterministic source term analyses performed using MELCOR do not explicitly include each risk-significant accident for all currently operating BWRs. Rather, the calculations performed are intended to be representative of typical sequences observed in BWR risk assessments. In addition, certain accident sequences identified in several BWR risk assessments were not considered because they did not offer particularly useful information regarding fission product release to the containment. For example, fission product transport from the reactor vessel in the so-called "V-sequence", which involves a LOCA outside containment, would bypass the containment, and thus not offer useful information to this study. This limitation, and how it is accounted for in the development of a HBU supplement to the NUREG-1465 Source Term, is further discussed in Sections 4 and 5.

3.2 Basis for Selection of Risk-Significant Accidents

It has been assumed here that reactor operating practices that extend maximum fuel rod burnup to the regulatory limit of 62 GWd/t does not change the core damage probability distribution for a plant to any significant extent. Consequently, accidents that contribute most to the frequency of core damage with a LBU core have been selected here for source term analysis. The details regarding the accident sequences selected for analysis are discussed in Section 4.

3.3 Onset of Release

A delay between the time coolant reaches the top of the active fuel and the onset of clad ballooning and rupture is included in the proposed HBU supplement to the NUREG-1465 Source Term. This delay was derived as the 25^{th} percentile of the distribution of delays inferred from the MELCOR calculations. This derivation is a departure from the practice established in NUREG-1465, where the minimum calculated delay between coolant falling to the top of active fuel and the onset of clad ballooning and rupture was adopted. This minimum delay arises, of course, in the LLOCA sequences which are viewed widely as having very low probability. If the NUREG-1465 practice were adopted, the delay found here and the delay recommended in NUREG-1465 would be the same.

3.4 Magnitude of Coolant Activity Release

MELCOR does not model activity of the coolant. Therefore, no mechanistic code information for the magnitude of the coolant release is directly available for use in the revised supplement. Therefore, the elemental composition of the release during the coolant release phase is not addressed, just as it is not addressed in the NUREG-1465 Source Term.

3.5 Fuel Damage Behavior

It is assumed that differences in the mechanical design features of fuel assemblies used in HBU and LBU cores do not significantly affect fuel degradation processes (such as metal-water reactions, eutectic formation, mechanical failure(s) of assembly components, and material relocation) or rates of fission product release from fuel. As a consequence of this necessary assumption, along with the similar decay heat profiles, the characteristics of core damage progression for LBU and HBU cores are similar in most cases. Observed differences can be traced primarily to relatively small differences in the decay heating rates. As noted earlier, long-term decay heat levels (>24 hours) tend to be 10-15% higher in the HBU cores than an LBU core. While these differences are usually small (< 1.5 MW), the fractional difference in total decay heat is large enough to produce slightly faster accident chronology in the HBU cases.

Should research in the future show there are significant differences in the core degradation behavior of HBU fuel in comparison to LBU fuel, these analyses of core degradation behavior will need to be revised. Expert opinions on HBU source terms [Ref.11] suggested that there might be differences between LBU and HBU fuels in the fuel melting point and fuel liquefaction process. Issues raised by the expert panel included potential differences in the interaction of melting cladding with the fuel due to the development of a restructured 'rim' region and by the formation of a significant oxide layer on the inner surface of the cladding. Also, it was postulated that degradation of high burnup fuel would involve 'fuel foaming' rather than fuel candling as observed with fuel at lower burnup levels.

An issue not addressed in Ref. 11, nor in recent fuel damage experiments, is the effect of significant perturbation in the rectangular array of rods in a fuel assembly, such as the large central water channel in the Atrium-10, on fuel damage morphology. If future fuel damage experiments using such assemblies identify new, or changes in, fuel damage progression, these calculations will need to be revised.

3.6 Modeling of Nonradioactive Aerosols

The effect of generation of nonradioactive aerosols during the core degradation process was not included in the development of the NUREG-1465 Source Term. As stated in Section 1, nonradioactive aerosols can be important to the total mass distribution between the RCS and containment, because these aerosols provide additional opportunity for radioactive fission product aerosols to agglomerate and settle onto RCS structures. The generation of nonradioactive aerosols generated during core degradation, in particular the release of tin alloy in Zircaloy cladding, was represented in these calculations using a simple, non-mechanistic method available in MELCOR.

3.7 Modeling of Fission Product Inventories and Associated Decay Heat

MELCOR requires spatial distribution of fission product mass and decay heat to be the same. Insights from the first-of-a-kind ORIGEN analysis described in Section 2 indicate that the spatial

distributions are different[5]. Changes to MELCOR itself would be required to permit the spatial distribution of fission product and decay heat to be specified independently (i.e., the specific decay power [w/kg] for any given fission product element / group can be specified on a cell-by-cell basis in the DCH package). Sensitivity calculations performed as part of this study suggest changes to the code to accommodate this modeling detail would not have a major impact on calculated results. In the calculations presented here, the MELCOR distribution of fission product mass and decay heat was made according to the fission product mass distribution calculated by ORIGEN.

3.8 Accident Progression Uncertainties

The MELCOR calculations performed for this study represent the current best-practices in MELCOR severe accident modeling techniques, as well as best-estimate treatment of accident progression. Quantitative uncertainty analyses were not performed to determine the full range of source terms that would be generated if credible alternative assumptions were made regarding uncertain severe accident processes or model input variables. However, several sensitivity calculations were performed to examine the individual effects of known uncertainties that could impact accident chronology and, therefore, the timing and/or duration of release periods. These are described in greater detail in Section 4.

[5] The ORIGEN calculations performed for this study suggest the local fractional quantity of fission product mass differs from the local fraction of core decay heat by as much as 50%. The spatial distribution of fission product mass was found to be in better agreement with the fission power distribution (local differences generally less than 16%). Agreement tended to better near the center of the core and much worse toward the peripheral ring and at the top and bottom of the active fuel length. Sensitivity calculations were performed using the Grand Gulf MELCOR model to measure the effects of distributing fission product inventory and decay heat according to one (but not both) of the distributions obtained from the ORIGEN results. The calculated chronology of key events was essentially unaffected by this input parameter. Insignificant differences were observed in the fractional release of radionuclides from fuel; and small differences were observed in the calculated quantity of individual species at a particular location in the reactor vessel, containment or transported to the environment (i.e., differences of less than a factor of two.)

4.0 Selection of Representative Accident Sequences

The effort to develop an alternative source term that is applicable for accidents involving high burnup cores is an extension of previous NRC work, namely the NUREG-1465 Source Term and the expert panel recommendations for extension of the revised source term to accidents involving high burnup fuels [Ref.11]. For this previous NRC work, the revised source term was based on a broad range of postulated accident sequences that were thought to be significant risk contributors at the time the work was performed. Note that large-break LOCAs are not represented on this list, as their contribution to the estimated core damage frequency of BWRs is typically found to not be significant. The accident sequences on which these source terms were based are listed in Table 10. Keys for BWR accident sequences symbols are provided in Table 11.

Table 10 BWR Accident Sequence Basis for Revised Source Term [Ref.6]

Plant	Sequence	Description
Peach Bottom (BWR/4, Mark I)	TC1	Anticipated Transient Without SCRAM (ATWS), reactor depressurized
	TC2	ATWS, reactor pressurized
	TC3	Same as TC2 except wetwell venting
	TB1	SBO, battery depletion
	TB2	Same as TB1 except CF at VF
	S2E1	2”-equivalent diameter LOCA, no ECCS, no Automatic Depressurization System (ADS)
	S2E2	Same as S2E1 except Peach Bottom concrete replaced by basaltic concrete
	V	Residual Heat Removal (RHR) pipe failure outside containment
	TBUX	SBO, loss of all DC power
LaSalle (BWR/5, Mark II)	TB	SBO, late CF
Grand Gulf (BWR/6, Mark III)	TC	ATWS, early CF fails ECCS
	TB1	SBO, battery depletion
	TB2	Same as TB1 except H2 burn-induced CF
	TBS	SBO, no ECCS, reactor depressurized
	TBR	Same as TBS except AC recovery after VF

Subsequent to the definition of the revised source term, the NRC completed a comprehensive review of licensee submittals for the Individual Plant Examination (IPE) Program. A complete summary of the risk-significance of all severe accident sequences postulated by the licensees, based on the IPE analyses, is provided in Volume 2 of NUREG-1560 [Ref. 26]. Significant improvement in the understanding of some severe accidents has been achieved based on NRC severe accident research conducted since the definition of the NUREG-1465 Source Term (1995) and the NRC review of the IPE Program (1997). Table 12 indicates risk significance, as discussed in NUREG-1560, for each of the accident sequences that provided the basis for the NUREG-1465 BWR Source Term. Note that NUREG-1560 listed various transient-initiated events as risk-significant accidents for many plants. However, failures of engineered safety features necessary for a sequence to result in core damage produce an event progression (and thus the estimated source term) that are very similar to the SBO accidents listed. Therefore, transient initiated sequences with independent failures of safety systems are not included in the tables.

Table 11 Key to BWR Accident Sequence Symbols [Ref. 27]

Symbol	Description
A	Rupture of reactor coolant boundary with equivalent diameter of greater than six inches
B	Failure of electric power to engineered safety features
C	Failure of the reactor protection system
D	Failure of vapor suppression
E	Failure of emergency core cooling injection
F	Failure of emergency core cooling functionability
G	Failure of containment isolation to limit leakage to less than 100 volume % per day
H	Failure of core spray recirculation system
I	Failure of low pressure recirculation system
J	Failure of high pressure service water system
M	Failure of safety/relief valves to open
P	Failure of safety/relief valves to recluse after opening
Q	Failure of normal feedwater system to provide core makeup water
S1	Small pipe break with an equivalent diameter of about 2 to 6 inches
S2	Small pipe break with an equivalent diameter of about 1/2 to 2 inches
T	Transient event
U	Failure of high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) to provide core makeup water
V	Failure of low pressure ECCS to provide core makeup water
W	Failure to remove residual core heat
α	Containment failure due to steam explosion in vessel
β	Containment failure due to steam explosion in containment
γ	Containment failure due to overpressure
δ	Containment isolation failure in drywell
ε	Containment isolation failure in wetwell
ζ	Containment leakage great than 2400 volume % per day
η	Reactor building isolation failure

Many of the accident sequences listed in Table 12 were identified as important contributors to risk in the NUREG-1560 assessment. Table 13 defines a MELCOR calculation matrix that would completely cover the breadth of accident sequences identified. Ideally, MELCOR calculations would be performed for each of these sequences, and the following process would be followed:

1. Develop current low burnup fuel state-of-the-art MELCOR models for each plant by updating physical core configuration and ensuring best-practice MELCOR modeling options are employed.

2. Perform MELCOR calculations for entire suite of accident sequences listed in Table 13 for low burnup fuel. This would provide a new baseline for source terms for low burnup fuel, and would provide a point of comparison for the NUREG-1465 revised source term.

3. Revise the MELCOR models with end-of-cycle fission product inventories, fission product decay heat levels, and core power shape information for high burnup fuels (see Section 2.3).

4. Repeat MELCOR calculations for entire suite of accident sequences listed in Table 13 for high burnup fuel. The results, when compared to the MELCOR calculations performed in Step 2, would provide the ability to assess the impact (on postulated severe accident source terms) of operating with fuels with end-of-cycle burnup in the range of 59 GWd/t (average) for the peak assembly. This corresponds to a burnup of approximately 65 GWd/t for the peak fuel rod.

Table 12 Risk-Significance of NUREG-1465 BWR Accident Sequences

Plant	NUREG-1465 Sequence	Sequence Description	Risk Significance (NUREG-1560)	Other Comments
Peach Bottom	TC1	ATWS, reactor depressurized	Low	
	TC2	ATWS, reactor pressurized	Low	
	TC3	TC2, wetwell venting	Low	
	TB1	SBO, battery depletion	High	
	TB2	Same as TB1 except CF at VF	High	Pressure @ VF or shell melt through
	S2E1	2”-equivalent diameter LOCA, no ECCS, no ADS	Low	
	S2E2	Same as S2E1except Peach Bottom concrete replaced by basaltic concrete	Low	
	V	ISLOCA	Low	
	TBUX	SBO, loss of all DC power	High	
LaSalle	TB	SBO, late CF	High	Late overpressure
Grand Gulf	TC	ATWS, early CF fails ECCS	Low	
	TB1	SBO, battery depletion	High	
	TB2	Same as TB1 except H2 burn-induced CF	High	
	TBS	SBO, no ECCS, reactor depressurized	High	
	TBR	Same as TBS except AC recovery after VF	High	

Resource limitations did not allow development of high- and low-burnup core models for each of the MELCOR models identified in Table 12 and Table 13. However, it was judged that a reasonably representative set of accident analyses could be performed by covering the sequences listed for Peach Bottom and Grand Gulf. This is primarily due to similarities in accident sequences across plant types. Therefore, this study only examines the accidents listed in Table 13. The ISLOCA event listed in Table 12, by definition, results in containment bypass. Protection from containment bypass events is considered separately in NRC regulations; therefore, no MELCOR calculation was performed for this scenario.

For both plants (Peach Bottom and Grand Gulf), a low burnup and high burnup calculation was performed for each of the cases identified in Table 11. This allows comparison of results from calculations where the only difference is the burnup level and the corresponding fuel type.

Table 13 MELCOR Calculation Matrix for BWR High Burnup Accident Sequences

Case	MELCOR Model	Init.	DC Power	Coolant Injection	RPV Pressure at VB	Containment Failure	Other
1A	Peach Bottom	ST-SBO	No	None	Low (SOSRV)	Early: DW liner melt-through	
1B	Peach Bottom	ST-SBO	No	None	Low (SOSRV)	Early DW liner melt-through	MCCI with basaltic concrete
1C	Peach Bottom	ST-SBO	No	None	Low (SOSRV)	Late: DW head flange leakage	
1D	Peach Bottom	ST-SBO	No	None	High	Early: DW liner melt-through	
2A	Peach Bottom	LT-SBO	8 hrs	RCIC	Low (SOSRV)	Early: DW liner melt-through	
2B	Peach Bottom	LT-SBO	8 hrs	RCIC	Low (SOSRV)	Late: DW head flange leakage	
2C	Peach Bottom	LT-SBO	8 hrs	RCIC	Low (SOSRV)	Late: Over-pressure in torus	
3	Peach Bottom	SLOCA	Yes	None	Low (LOCA)	Early: DW head flange leakage	Steam line (SRV tee) LOCA
4	Peach Bottom	LLOCA	Yes	None	Low (LOCA)	Early: DW liner melt-through	Recirculation suction line LOCA
5A	Grand Gulf	ST-SBO	No	None	Low (SOSRV)	Early: H_2 burn at vessel breach	H_2 burn also causes failure of drywell wall
5B	Grand Gulf	ST-SBO	No	No	High	Early: H_2 burn at vessel breach	H_2 burn also causes failure of drywell wall
5C	Grand Gulf	ST-SBO	No	None	Low (SOSRV)	Late: Over-pressure	
6A	Grand Gulf	LT-SBO	8 hrs	RCIC*	Low (SOSRV)	Early: H_2 burn at vessel breach	H_2 burn also causes failure of drywell wall
6B	Grand Gulf	LT-SBO	8 hrs	RCIC*	Low (SOSRV)	Late: Over-pressure	
7	Grand Gulf	ATWS	Yes	Yes	Low (ADS)	Prior to Onset of Core Damage	MSIV closure
8	Grand Gulf	LLOCA	Yes	RCIC only	Low (LOCA)	Late: Over-pressure	Recirculation suction line LOCA

Notes: Blank cells indicates that no specification will be made and MELCOR will calculate results.
'ST-SBO' = short-term station blackout
'LT-SBO' = long-term station blackout
'DW' = drywell
'SOSRV' = stuck open safety/relief valve: cycling of the lowest-setpoint relief valve at high temperatures causes the valve to seize in the open position during late-phase in-vessel core damage
RCIC* = cases with "*" indicate isolation of RCIC due to high suppression pool temperatures before station batteries are exhausted

5.0 Accident Source Terms Calculated Using MELCOR

Based on the accident sequence selection process described in Section 4, MELCOR calculations were performed for several sequences in two BWR plant designs using LBU and separately HBU core design data. In addition to considering a spectrum of different sequences, the calculations also consider the effects of certain uncertainties in severe accident progression, such as behavior of safety/relief valves when subjected to high temperatures arising from clad oxidation and fuel melting and the mode/timing of containment failure. Sensitivity calculations examining these topics reflect a wider range of plausible severe accident conditions for use in the HBU supplement to NUREG-1465.

Calculated results for fission product release timing and magnitude during the various release phases are presented in Sections 5.1 and 5.2, respectively.

5.1 MELCOR Results for Durations of the Release Phases

A complete listing of the timing of key events for the MELCOR accident sequences modeled is included in Appendix C. Key plots showing accident signatures for select sequences are provided in Appendix D. The remainder of Section 5.1 contains a summary of the MELCOR results that are pertinent to development of the HBU supplemental source term.

Definition of the release phases described in Sections 5.1.1 through 5.1.5 are based on the NUREG-1465 Source Term definitions shown in Figure 1. The calculated MELCOR parameters selected to determine the timing of each release phase for the accidents simulated are shown in Figure 10.

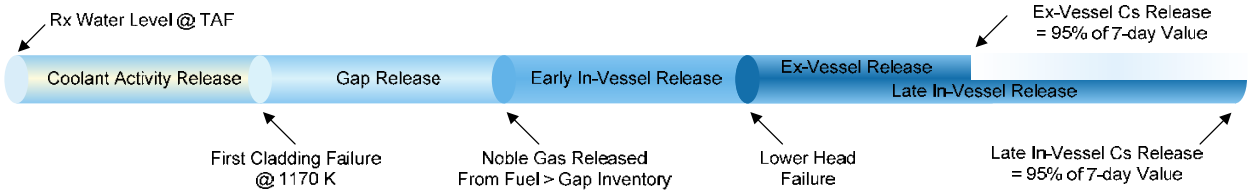

Figure 10 Release Phase Timing Definitions – Tie to Calculated MELCOR Results

5.1.1 Onset of Release

The onset of coolant activity release has been taken to be the time at which coolant reaches the top of the active fuel. Results are shown in Table 14.

Table 14 Time of Onset of Radionuclide Release for BWRs

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Peach Bottom	1A	ST-SBO, SOSRV, CF@VB	0.65	0.65
	1B	ST-SBO, VB@HP, CF Early	0.65	0.65
	1C	ST-SBO, SOSRV, CF@VB, Basaltic MCCI	0.65	0.65
	1D	ST-SBO, SOSRV, CF Late	0.65	0.65
	2A	LT-SBO, SOSRV, CF Early	11.50	10.69
	2B	LT-SBO, SOSRV, CF Early	11.50	10.69
	2C	LT-SBO, SOSRV, CF Late	11.50	10.69
	3	SLOCA, CF Early	0.66	0.65
	4	LLOCA, CF Early	0.00	0.00
Grand Gulf	5A	ST-SBO, SOSRV, CF Early	0.66	0.65
	5B	ST-SBO, VB@HP, CF Early	0.66	0.65
	5C	ST-SBO, SOSRV, CF Late	0.66	0.65
	6A	LT-SBO, SOSRV, CF Early	4.45	4.21
	6B	LT-SBO, SOSRV, CF Late	4.45	4.21
	7	ATWS, CF Early	0.03	0.04
	8	LLOCA, CF Early	0.00	0.00

Notes: SOSRV = stuck open S/RV (refer to notes in Table 13)
CF Early = containment failure before, or soon after, vessel breach (refer to Table 13 for specific failure mode)
CF Late = containment failure occurs more than one hour after vessel breach
VB@HP = vessel breach at high pressure

5.1.2 Duration of the Coolant Release Phase

The duration of the coolant release phase is taken to be the period between the onset of release and the time at which clad begins to balloon and rupture. These results are shown in Table 15.

Table 15 Duration of Coolant Release Phase for BWRs

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Peach Bottom	1A	ST-SBO, SOSRV, CF@VB	0.59	0.56
	1B	ST-SBO, VB@HP, CF Early	0.59	0.56
	1C	ST-SBO, SOSRV, CF@VB, Basaltic MCCI	0.59	0.56
	1D	ST-SBO, SOSRV, CF Late	0.59	0.56
	2A	LT-SBO, SOSRV, CF Early	1.33	1.16
	2B	LT-SBO, SOSRV, CF Early	1.33	1.16
	2C	LT-SBO, SOSRV, CF Late	1.33	1.16
	3	SLOCA, CF Early	0.59	0.56
	4	LLOCA, CF Early	0.15	0.13
Grand Gulf	5A	ST-SBO, SOSRV, CF Early	0.57	0.52
	5B	ST-SBO, VB@HP, CF Early	0.57	0.52
	5C	ST-SBO, SOSRV, CF Late	0.57	0.52
	6A	LT-SBO, SOSRV, CF Early	0.89	0.80
	6B	LT-SBO, SOSRV, CF Late	0.89	0.80
	7	ATWS, CF Early	9.11	7.66
	8	LLOCA, CF Early	0.15	0.13

Refer to notes in Table 14.

5.1.3 Duration of the Gap Release Phase

The duration of the gap release phase is taken to be the time between the start of the ballooning and rupture of the cladding and the onset of the in-vessel release. The distinction between gap release and the onset of in-vessel release has always been difficult to define. This distinction has become more obscure as modern, mechanistic codes simulate the spatially heterogeneous nature of core degradation. It is commonly predicted with these modern computer codes that fuel in central regions of the core can be damaged extensively, even to the point of liquefaction and relocation, before cladding in the peripheral regions of the core has ballooned and ruptured to allow release of the gap inventory. The idea of gap release has evolved, then, to simply be a measure of a release fraction of the more volatile radionuclides equal to the inventory of these radionuclides in the fuel-cladding gap throughout the core. So it is taken here. As shown in Figure 10, the onset of the in-vessel release phase has been taken as the time at which MELCOR calculates the released mass of noble gases to exceed the gap inventory.

The calculated durations of the gap releases are shown in Table 16. Comparisons are shown in these tables for results obtained with both LBU and HBU cores. The gap release durations are not significantly different between the LBU and HBU calculations.

Table 16 Duration of Gap Release Phase for BWRs

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Peach Bottom	1A	ST-SBO, SOSRV, CF@VB	0.17	0.15
	1B	ST-SBO, VB@HP, CF Early	0.17	0.15
	1C	ST-SBO, SOSRV, CF@VB, Basaltic MCCI	0.17	0.15
	1D	ST-SBO, SOSRV, CF Late	0.17	0.15
	2A	LT-SBO, SOSRV, CF Early	0.27	0.25
	2B	LT-SBO, SOSRV, CF Early	0.27	0.25
	2C	LT-SBO, SOSRV, CF Late	0.27	0.25
	3	SLOCA, CF Early	0.15	0.13
	4	LLOCA, CF Early	0.15	0.06
Grand Gulf	5A	ST-SBO, SOSRV, CF Early	0.19	0.16
	5B	ST-SBO, VB@HP, CF Early	0.19	0.16
	5C	ST-SBO, SOSRV, CF Late	0.19	0.17
	6A	LT-SBO, SOSRV, CF Early	0.26	0.18
	6B	LT-SBO, SOSRV, CF Late	0.26	0.18
	7	ATWS, CF Early	0.47	0.15
	8	LLOCA, CF Early	0.16	0.06

Refer to notes in Table 14.

5.1.4 Duration of the In-vessel Release Phase

The duration of the in-vessel release phase is taken to be the period from the end of the gap release phase to the time core debris penetrates the reaction vessel lower head. Not all of the core debris is expelled, of course, from the reactor vessel at the time of lower head failure. Release from any residual core materials that remain in the vessel is taken to be part of the late in-vessel release. Results for the durations of in-vessel release are shown in Table 17.

Table 17 Duration of In-Vessel Release Phase for BWRs

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Peach Bottom	1A	ST-SBO, SOSRV, CF@VB	7.81	6.34
	1B	ST-SBO, VB@HP, CF Early	6.80	5.90
	1C	ST-SBO, SOSRV, CF@VB, Basaltic MCCI	7.81	6.34
	1D	ST-SBO, SOSRV, CF Late	7.81	6.34
	2A	LT-SBO, SOSRV, CF Early	11.13	9.81
	2B	LT-SBO, SOSRV, CF Early	11.13	9.67
	2C	LT-SBO, SOSRV, CF Late	11.13	9.67
	3	SLOCA, CF Early	7.41	7.35
	4	LLOCA, CF Early	6.29	5.66
Grand Gulf	5A	ST-SBO, SOSRV, CF Early	9.05	9.15
	5B	ST-SBO, VB@HP, CF Early	7.28	6.97
	5C	ST-SBO, SOSRV, CF Late	9.65	8.58
	6A	LT-SBO, SOSRV, CF Early	12.11	11.02
	6B	LT-SBO, SOSRV, CF Late	12.11	11.02
	7	ATWS, CF Early	11.79	11.45
	8	LLOCA, CF Early	6.69	5.32

Refer to notes in Table 14.

5.1.5 Duration of the Ex-vessel Release Phase

Ex-vessel release is dominated by the release associated with the interactions of reactor core debris with the structural concrete in the reactor cavity. The attack persists for a very long time. During this attack radionuclides can be released in the form of aerosols along with very large amounts of non-radioactive aerosols. The release of radionuclides proceeds at a very slow rate once the residual metallic zirconium in the core debris has been oxidized and sensible heat of the core debris has been reduced by dilution with condensed products of concrete decomposition. Consequently, the duration of ex-vessel release is taken to be the period from vessel lower head failure and the time 95% of cesium in the core debris has been released to the containment atmosphere. Ex-vessel release durations are shown in Table 18.

Table 18 Duration of Ex-Vessel Release Phase for BWRs

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Peach Bottom	1A	ST-SBO, SOSRV, CF@VB	0.59	1.14
	1B	ST-SBO, VB@HP, CF Early	3.14	2.21
	1C	ST-SBO, SOSRV, CF@VB, Basaltic MCCI	0.59	1.14
	1D	ST-SBO, SOSRV, CF Late	1.26	2.67
	2A	LT-SBO, SOSRV, CF Early	2.61	2.02
	2B	LT-SBO, SOSRV, CF Early	4.34	2.36
	2C	LT-SBO, SOSRV, CF Late	6.61	3.32
	3	SLOCA, CF Early	1.31	1.02
	4	LLOCA, CF Early	2.11	2.80
Grand Gulf	5A	ST-SBO, SOSRV, CF Early	1.15	5.70
	5B	ST-SBO, VB@HP, CF Early	10.31	8.11
	5C	ST-SBO, SOSRV, CF Late	0.71	6.43
	6A	LT-SBO, SOSRV, CF Early	0.52	1.93
	6B	LT-SBO, SOSRV, CF Late	2.52	17.19
	7	ATWS, CF Early	1.50	3.32
	8	LLOCA, CF Early	1.04	9.00

Refer to notes in Table 14.

5.1.6 Duration of the Late In-vessel Release Phase

At the time that the NUREG-1465 Source Term was developed, the late in-vessel release was more hypothesized than explicitly calculated. For this supplement, explicit calculations of the late in-vessel release were performed. As illustrated in Figure 10, the 'late in-vessel release phase' is defined here to represent the release of fission products from the reactor vessel (to containment) during the time frame between vessel breach (lower head failure) the time at which 95% of the total (7-day) release of cesium occurs. These calculations showed that the late in-vessel release persists for a very long time at a very low rate. Results are shown in Table 19.

Table 19 Duration of Late In-Vessel Release Phase for BWRs

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Peach Bottom	1A	ST-SBO, SOSRV, CF@VB	19.61	10.41
	1B	ST-SBO, VB@HP, CF Early	13.06	11.15
	1C	ST-SBO, SOSRV, CF@VB, Basaltic MCCI	4.08	4.64
	1D	ST-SBO, SOSRV, CF Late	14.25	8.14
	2A	LT-SBO, SOSRV, CF Early	8.04	10.59
	2B	LT-SBO, SOSRV, CF Early	6.14	4.72
	2C	LT-SBO, SOSRV, CF Late	15.61	13.82
	3	SLOCA, CF Early	11.33	8.61
	4	LLOCA, CF Early	9.72	0.37
Grand Gulf	5A	ST-SBO, SOSRV, CF Early	10.05	14.13
	5B	ST-SBO, VB@HP, CF Early	1.01	14.25
	5C	ST-SBO, SOSRV, CF Late	140.85	149.91
	6A	LT-SBO, SOSRV, CF Early	17.05	14.69
	6B	LT-SBO, SOSRV, CF Late	144.19	147.45
	7	ATWS, CF Early	18.34	22.38
	8	LLOCA, CF Early	9.14	9.63

Refer to notes in Table 14.

5.2 MELCOR Results for Release Composition and Magnitude

Calculated elemental compositions of the releases during the various phases of an accident involving substantial meltdown of the reactor core are discussed here. The elemental composition of the release during the coolant release phase is not addressed, just as it is not addressed in the NUREG-1465 Source Term.

5.2.1 Gap Release

Gap release results are shown in Table 20 and Table 21 for accident sequences involving LBU and HBU cores, respectively. Results are only shown for the four most volatile radionuclide groups, following the example in NUREG-1465. It should be emphasized that these source term release fractions represent releases to the containment.

Table 20 Gap Release Fractions to Containment for BWRs with a LBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Tellurium
Peach Bottom	1A	6.1E-03	2.9E-03	2.7E-03	2.8E-03
	1B	6.1E-03	2.9E-03	2.7E-03	2.8E-03
	1C	6.1E-03	2.9E-03	2.7E-03	2.8E-03
	1D	6.1E-03	2.9E-03	2.7E-03	2.8E-03
	2A	2.2E-03	9.2E-04	6.4E-04	7.2E-04
	2B	2.2E-03	9.2E-04	6.4E-04	7.2E-04
	2C	2.2E-03	9.2E-04	6.4E-04	7.2E-04
	3	2.0E-02	1.3E-02	1.3E-02	1.3E-02
	4	2.7E-02	2.6E-02	2.6E-02	2.6E-02
Grand Gulf	5A	8.2E-03	2.9E-03	2.5E-03	2.6E-03
	5B	8.2E-03	2.9E-03	2.5E-03	2.6E-03
	5C	8.2E-03	2.9E-03	2.6E-03	2.7E-03
	6A	1.0E-02	3.9E-03	3.8E-03	3.8E-03
	6B	1.0E-02	3.9E-03	3.8E-03	3.8E-03
	7	4.1E-02	2.9E-02	2.9E-02	2.9E-02
	8	4.7E-02	4.4E-02	4.5E-02	4.4E-02

Table 21 Gap Release Fractions to Containment for BWRs with a HBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Tellurium
Peach Bottom	1A	5.6E-03	1.7E-03	1.3E-03	1.4E-03
	1B	5.6E-03	1.7E-03	1.3E-03	1.4E-03
	1C	5.6E-03	1.7E-03	1.3E-03	1.4E-03
	1D	5.6E-03	1.7E-03	1.3E-03	1.4E-03
	2A	3.8E-03	8.6E-04	5.3E-04	5.7E-04
	2B	3.8E-03	8.6E-04	5.3E-04	5.7E-04
	2C	3.8E-03	8.6E-04	5.3E-04	5.7E-04
	3	1.6E-02	5.4E-03	4.3E-03	4.5E-03
	4	1.4E-02	1.3E-02	1.2E-02	1.2E-02
Grand Gulf	5A	7.8E-03	2.0E-03	1.5E-03	1.6E-03
	5B	7.8E-03	2.0E-03	1.5E-03	1.6E-03
	5C	8.9E-03	2.2E-03	1.7E-03	1.8E-03
	6A	1.1E-02	2.7E-03	1.8E-03	1.9E-03
	6B	1.1E-02	2.7E-03	1.8E-03	1.9E-03
	7	3.9E-02	3.4E-02	3.4E-02	3.4E-02
	8	3.8E-02	3.7E-02	3.6E-02	3.6E-02

5.2.2 In-vessel Release

Calculated in-vessel release fractions for the various groups of radionuclides are shown in Table 22 and Table 23 for accident sequences involving LBU and HBU cores, respectively.

Table 22 In-vessel Release Fractions to Containment for BWRs with a LBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Te Group	Ba/Sr Group
Peach Bottom	1A	9.3E-01	3.0E-01	1.2E-01	1.7E-01	3.8E-03
	1B	9.2E-01	2.5E-01	1.1E-01	1.7E-01	3.8E-03
	1C	9.3E-01	3.0E-01	1.2E-01	1.7E-01	3.8E-03
	1D	9.3E-01	3.0E-01	1.2E-01	1.7E-01	3.8E-03
	2A	9.2E-01	3.5E-01	1.0E-01	2.1E-01	2.8E-03
	2B	9.2E-01	3.5E-01	1.0E-01	2.1E-01	2.8E-03
	2C	9.2E-01	3.5E-01	1.0E-01	2.1E-01	2.8E-03
	3	9.1E-01	6.6E-01	3.6E-01	4.9E-01	7.2E-03
	4	9.4E-01	8.5E-01	3.7E-01	5.3E-01	7.4E-03
Grand Gulf	5A	9.2E-01	7.7E-01	1.5E-01	7.7E-01	1.1E-02
	5B	9.0E-01	5.9E-01	1.3E-01	4.9E-01	4.1E-03
	5C	9.2E-01	7.5E-01	1.4E-01	7.3E-01	9.8E-03
	6A	9.1E-01	7.7E-01	2.0E-01	7.6E-01	1.3E-02
	6B	9.1E-01	7.7E-01	2.0E-01	7.6E-01	1.3E-02
	7	9.1E-01	5.5E-01	3.6E-01	3.7E-01	5.6E-03
	8	9.3E-01	6.7E-01	4.4E-01	5.1E-01	8.0E-03

continued

Plant	Case	Ru Group	Mo Group	Lanthan-ides	Ce Group
Peach Bottom	1A	2.6E-03	2.8E-02	2.1E-07	1.6E-07
	1B	2.2E-03	2.7E-02	2.1E-07	1.6E-07
	1C	2.6E-03	2.8E-02	2.1E-07	1.6E-07
	1D	2.6E-03	2.8E-02	2.1E-07	1.6E-07
	2A	1.4E-03	2.2E-02	9.1E-08	6.7E-08
	2B	1.4E-03	2.2E-02	9.1E-08	6.7E-08
	2C	1.4E-03	2.2E-02	9.1E-08	6.7E-08
	3	6.0E-03	8.5E-02	1.8E-07	1.7E-07
	4	5.9E-03	8.5E-02	4.5E-07	3.6E-07
Grand Gulf	5A	2.2E-03	2.7E-02	1.2E-07	9.0E-08
	5B	2.0E-03	2.4E-02	9.0E-08	8.4E-08
	5C	2.1E-03	2.6E-02	1.1E-07	9.2E-08
	6A	3.5E-03	4.1E-02	3.7E-07	2.3E-07
	6B	3.5E-03	4.1E-02	3.7E-07	2.3E-07
	7	1.2E-02	9.3E-02	4.4E-07	4.0E-07
	8	9.5E-03	1.1E-01	6.3E-07	4.6E-07

Table 23 In-vessel Release Fractions to Containment for BWRs with a HBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Te Group	Ba/Sr Group
Peach Bottom	1A	9.6E-01	2.4E-01	7.9E-02	2.4E-01	3.4E-03
	1B	9.5E-01	1.9E-01	7.1E-02	1.5E-01	2.6E-03
	1C	9.6E-01	2.4E-01	7.9E-02	2.4E-01	3.4E-03
	1D	9.6E-01	2.4E-01	7.9E-02	2.4E-01	3.4E-03
	2A	9.6E-01	1.6E-01	5.6E-02	1.1E-01	1.7E-03
	2B	9.6E-01	1.6E-01	5.6E-02	1.1E-01	1.7E-03
	2C	9.6E-01	1.6E-01	5.6E-02	1.1E-01	1.7E-03
	3	9.7E-01	3.2E-01	2.0E-01	2.4E-01	3.2E-03
	4	9.8E-01	8.6E-01	4.8E-01	7.6E-01	1.1E-02
Grand Gulf	5A	9.8E-01	8.1E-01	1.4E-01	7.6E-01	8.9E-03
	5B	9.8E-01	6.2E-01	1.3E-01	4.7E-01	3.9E-03
	5C	9.8E-01	8.6E-01	1.4E-01	8.3E-01	1.1E-02
	6A	9.6E-01	7.7E-01	1.9E-01	7.1E-01	1.0E-02
	6B	9.6E-01	7.7E-01	1.9E-01	7.1E-01	1.0E-02
	7	9.6E-01	6.4E-01	5.5E-01	5.6E-01	6.5E-03
	8	9.6E-01	7.5E-01	4.9E-01	5.6E-01	1.4E-02

continued

Plant	Case	Ru Group	Mo Group	Lanthan-ides	Ce Group
Peach Bottom	1A	2.4E-03	1.7E-02	1.4E-07	1.2E-07
	1B	2.3E-03	1.6E-02	1.3E-07	1.2E-07
	1C	2.4E-03	1.7E-02	1.4E-07	1.2E-07
	1D	2.4E-03	1.7E-02	1.4E-07	1.2E-07
	2A	1.4E-03	1.3E-02	6.8E-08	5.7E-08
	2B	1.4E-03	1.3E-02	6.8E-08	5.7E-08
	2C	1.4E-03	1.3E-02	6.8E-08	5.7E-08
	3	6.2E-03	5.0E-02	1.5E-07	1.4E-07
	4	8.6E-03	1.2E-01	1.5E-06	1.5E-06
Grand Gulf	5A	2.0E-03	2.2E-02	1.2E-07	9.6E-08
	5B	2.1E-03	2.3E-02	1.2E-07	9.7E-08
	5C	2.1E-03	2.1E-02	1.9E-07	1.7E-07
	6A	4.3E-03	3.8E-02	1.5E-07	1.4E-07
	6B	4.3E-03	3.8E-02	1.5E-07	1.4E-07
	7	1.4E-02	1.4E-01	4.1E-07	3.9E-07
	8	1.9E-02	1.2E-01	2.3E-06	2.3E-06

5.2.3 Ex-vessel Release

Ex-vessel release fractions are shown in Table 24 and Table 25 for accident sequences involving a LBU and HBU core, respectively.

Table 24 Ex-vessel Release Fractions to Containment for BWRs with a LBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Te Group	Ba/Sr Group
Peach Bottom	1A	5.4E-02	4.8E-02	5.6E-02	6.1E-03	5.3E-02
	1B	6.4E-02	6.4E-02	6.6E-02	1.2E-02	8.1E-02
	1C	5.4E-02	4.4E-02	5.5E-02	7.2E-03	6.8E-02
	1D	5.4E-02	3.1E-02	5.5E-02	3.3E-03	2.4E-02
	2A	7.2E-02	4.1E-02	7.5E-02	4.0E-03	8.0E-03
	2B	6.9E-02	5.6E-02	7.3E-02	8.5E-03	2.3E-02
	2C	7.2E-02	6.5E-02	7.6E-02	1.3E-02	4.4E-02
	3	6.4E-02	9.5E-02	6.7E-02	4.1E-02	2.4E-01
	4	2.1E-02	1.4E-02	2.5E-02	1.2E-03	1.4E-02
Grand Gulf	5A	6.8E-02	5.4E-02	6.7E-02	7.0E-03	3.6E-02
	5B	8.2E-02	7.6E-02	8.2E-02	8.7E-03	2.3E-03
	5C	5.6E-02	6.7E-02	5.8E-02	2.0E-02	7.5E-02
	6A	7.5E-02	4.2E-02	7.4E-02	3.8E-03	2.3E-02
	6B	7.4E-02	3.6E-02	7.3E-02	3.0E-03	9.8E-03
	7	4.5E-02	3.1E-02	5.4E-02	3.0E-03	1.9E-02
	8	1.2E-02	5.9E-03	1.5E-02	2.8E-04	5.2E-03

continued

Plant	Case	Ru Group	Mo Group	Lanthan-ides	Ce Group
Peach Bottom	1A	1.1E-10	1.5E-02	1.2E-04	4.0E-03
	1B	2.2E-10	1.8E-02	2.1E-04	6.5E-03
	1C	1.3E-10	1.5E-02	1.6E-04	5.6E-03
	1D	4.1E-11	1.5E-02	5.0E-05	1.6E-03
	2A	5.6E-12	2.1E-02	1.3E-05	2.1E-04
	2B	3.2E-11	2.0E-02	4.3E-05	9.4E-04
	2C	1.1E-10	2.1E-02	1.0E-04	2.5E-03
	3	5.8E-09	1.7E-02	1.6E-03	4.2E-02
	4	1.4E-11	6.9E-03	2.2E-05	6.5E-04
Grand Gulf	5A	5.4E-10	1.8E-02	1.5E-04	4.7E-03
	5B	9.8E-13	2.2E-02	1.8E-05	3.4E-05
	5C	5.8E-09	1.5E-02	8.3E-04	2.0E-02
	6A	2.8E-11	2.0E-02	3.8E-05	1.4E-03
	6B	7.4E-12	2.0E-02	1.5E-05	4.7E-04
	7	1.4E-11	1.5E-02	2.4E-05	8.8E-04
	8	2.1E-12	4.0E-03	6.9E-06	2.0E-04

Table 25 Ex-vessel Release Fractions to Containment for BWRs with a HBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Te Group	Ba/Sr Group
Peach Bottom	1A	8.6E-03	1.5E-02	9.7E-03	3.2E-03	1.1E-01
	1B	1.2E-02	2.0E-02	1.5E-02	6.6E-03	1.7E-02
	1C	8.4E-03	1.2E-02	9.3E-03	3.0E-03	4.2E-02
	1D	9.7E-03	1.6E-02	1.1E-02	5.5E-03	1.4E-01
	2A	1.3E-02	1.5E-02	1.6E-02	2.5E-03	2.6E-02
	2B	1.4E-02	1.5E-02	1.6E-02	2.7E-03	2.7E-02
	2C	1.4E-02	1.8E-02	1.7E-02	4.9E-03	5.9E-02
	3	3.9E-03	1.0E-02	4.7E-03	2.6E-03	1.5E-01
	4	1.3E-03	3.2E-03	1.6E-03	9.8E-04	1.9E-01
Grand Gulf	5A	3.6E-03	8.2E-03	4.4E-03	9.8E-04	1.6E-02
	5B	1.1E-02	1.5E-02	1.3E-02	1.5E-03	1.5E-03
	5C	9.2E-03	1.1E-02	1.0E-02	1.6E-03	2.0E-02
	6A	9.5E-03	1.4E-02	1.1E-02	1.6E-03	2.7E-02
	6B	7.8E-03	1.3E-02	9.1E-03	1.9E-03	1.5E-02
	7	1.5E-03	3.3E-03	2.2E-03	4.6E-04	2.2E-02
	8	1.0E-03	2.5E-03	1.2E-03	3.4E-04	8.0E-03

continued

Plant	Case	Ru Group	Mo Group	Lanthan-ides	Ce Group
Peach Bottom	1A	4.9E-10	2.5E-03	2.1E-04	1.3E-02
	1B	6.4E-10	3.9E-03	5.0E-05	2.0E-03
	1C	3.5E-10	2.4E-03	7.4E-05	3.6E-03
	1D	1.5E-09	2.8E-03	3.7E-04	1.8E-02
	2A	4.6E-11	4.2E-03	3.1E-05	1.3E-03
	2B	5.9E-11	4.4E-03	3.5E-05	1.5E-03
	2C	2.6E-10	4.4E-03	1.0E-04	4.8E-03
	3	1.3E-09	1.1E-03	4.0E-04	2.4E-02
	4	9.6E-10	3.9E-04	4.5E-04	2.7E-02
Grand Gulf	5A	1.6E-11	1.1E-03	2.3E-05	8.7E-04
	5B	6.3E-13	3.4E-03	1.7E-05	2.8E-05
	5C	8.6E-11	2.7E-03	4.0E-05	1.8E-03
	6A	4.3E-11	2.9E-03	3.6E-05	1.8E-03
	6B	2.4E-11	2.3E-03	2.9E-05	1.0E-03
	7	1.4E-11	5.5E-04	2.3E-05	9.6E-04
	8	4.3E-11	2.8E-04	2.7E-05	1.0E-03

5.2.4 Late In-vessel Release

Late in-vessel release fractions are shown in Table 26 and Table 27 for accident sequences involving a LBU and HBU core, respectively.

Table 26 Late In-vessel Release Fractions to Containment for BWRs with a LBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Te Group	Ba/Sr Group
Peach Bottom	1A	1.2E-02	5.9E-01	7.4E-02	6.7E-01	1.2E-02
	1B	3.5E-03	6.2E-01	7.1E-02	6.3E-01	1.3E-02
	1C	1.2E-02	5.4E-01	6.7E-02	4.1E-01	4.2E-03
	1D	1.2E-02	5.2E-01	5.3E-02	5.9E-01	8.6E-03
	2A	5.2E-03	4.5E-01	9.9E-02	5.2E-01	5.6E-03
	2B	4.4E-03	4.5E-01	6.5E-02	5.4E-01	6.9E-03
	2C	4.6E-03	5.4E-01	1.1E-01	5.9E-01	1.1E-02
	3	4.4E-03	1.9E-01	6.5E-02	3.1E-01	2.2E-03
	4	1.1E-02	8.2E-02	5.1E-02	3.9E-01	5.2E-03
Grand Gulf	5A	6.4E-03	2.4E-02	9.0E-03	2.0E-02	6.1E-04
	5B	2.2E-03	1.6E-01	1.6E-02	2.1E-01	9.3E-03
	5C	1.4E-02	1.5E-01	1.6E-02	1.6E-01	3.5E-03
	6A	3.0E-03	7.5E-02	1.3E-02	1.5E-02	7.8E-04
	6B	4.2E-03	1.1E-01	9.7E-03	9.7E-02	9.4E-03
	7	2.8E-03	2.0E-01	2.0E-02	3.1E-02	7.4E-05
	8	7.4E-03	1.9E-01	2.3E-02	9.5E-02	3.2E-04

continued

Plant	Case	Ru Group	Mo Group	Lanthan-ides	Ce Group
Peach Bottom	1A	3.8E-05	8.8E-03	3.9E-12	0.0E+00
	1B	3.2E-06	7.1E-03	7.4E-08	1.1E-09
	1C	3.9E-05	8.1E-03	3.9E-12	0.0E+00
	1D	7.4E-06	4.9E-03	3.9E-12	0.0E+00
	2A	5.2E-06	1.8E-02	2.1E-08	1.8E-12
	2B	1.6E-06	9.4E-03	9.3E-11	1.8E-12
	2C	9.8E-07	2.0E-02	9.3E-11	1.8E-12
	3	1.9E-05	1.4E-02	2.7E-08	3.8E-07
	4	2.5E-05	1.3E-02	1.1E-09	1.2E-09
Grand Gulf	5A	5.1E-05	1.7E-03	1.7E-07	2.1E-09
	5B	1.8E-05	1.4E-03	3.0E-09	5.8E-10
	5C	5.7E-06	2.5E-03	2.3E-12	2.7E-12
	6A	2.2E-05	2.2E-03	4.1E-07	7.6E-11
	6B	5.8E-06	2.3E-04	1.7E-08	6.8E-06
	7	1.3E-04	1.9E-03	6.3E-08	9.8E-10
	8	1.8E-04	2.8E-03	6.5E-08	6.3E-10

Table 27 Late In-vessel Release Fractions to Containment for BWRs with a HBU Core

Plant	Case	Noble Gases	Halogens	Alkali Metals	Te Group	Ba/Sr Group
Peach Bottom	1A	2.5E-02	6.4E-01	9.3E-02	5.6E-01	5.3E-03
	1B	2.6E-02	7.0E-01	9.4E-02	7.2E-01	1.4E-02
	1C	2.5E-02	5.0E-01	7.3E-02	2.2E-01	7.6E-04
	1D	2.5E-02	6.2E-01	6.4E-02	5.0E-01	3.0E-03
	2A	2.7E-02	6.9E-01	1.1E-01	7.3E-01	5.7E-03
	2B	2.5E-02	6.8E-01	8.9E-02	7.2E-01	5.1E-03
	2C	2.6E-02	7.5E-01	1.2E-01	7.6E-01	1.4E-02
	3	9.0E-03	6.5E-01	8.1E-02	7.3E-01	7.8E-03
	4	5.8E-03	9.4E-02	3.6E-02	2.5E-02	7.0E-04
Grand Gulf	5A	7.7E-03	4.6E-02	9.0E-03	5.6E-02	2.0E-03
	5B	2.9E-03	2.5E-01	2.0E-02	3.1E-01	5.5E-03
	5C	7.1E-03	1.1E-01	2.1E-02	1.0E-01	4.2E-03
	6A	1.5E-02	1.5E-01	2.2E-02	9.7E-02	1.1E-03
	6B	1.5E-02	1.7E-01	1.6E-02	2.4E-01	7.3E-03
	7	3.1E-03	1.7E-01	1.5E-02	1.7E-02	2.6E-05
	8	3.8E-03	1.7E-01	2.2E-02	1.1E-01	1.4E-02

continued

Plant	Case	Ru Group	Mo Group	Lanthan-ides	Ce Group
Peach Bottom	1A	3.5E-04	1.3E-02	3.9E-10	2.6E-10
	1B	7.6E-05	1.3E-02	2.3E-08	5.2E-09
	1C	3.1E-04	9.8E-03	3.3E-10	2.2E-10
	1D	1.7E-05	5.5E-03	3.9E-10	2.6E-10
	2A	4.7E-05	1.6E-02	2.0E-08	0.0E+00
	2B	1.8E-05	1.2E-02	0.0E+00	0.0E+00
	2C	3.8E-06	1.9E-02	0.0E+00	0.0E+00
	3	6.4E-04	1.0E-02	9.3E-09	1.9E-14
	4	6.7E-05	1.2E-03	4.7E-09	2.3E-09
Grand Gulf	5A	2.1E-04	1.4E-03	9.1E-08	5.1E-10
	5B	9.3E-05	6.8E-04	6.0E-09	1.4E-09
	5C	8.1E-05	2.3E-03	5.4E-09	3.0E-05
	6A	9.9E-05	3.0E-03	3.0E-07	1.4E-09
	6B	3.0E-05	2.2E-04	7.5E-09	3.8E-05
	7	1.7E-04	9.1E-04	2.2E-08	4.5E-10
	8	5.0E-04	2.5E-03	2.9E-08	2.2E-09

6.0 Discussion and Summary

6.1 Differences Between HBU and LBU Source Terms

The effects of discharge burnup on severe accident chronology and the magnitude of fission product releases to the containment can be discerned by examining differences in the HBU and LBU results tabulated in Chapter 5. For the purposes of this discussion, the differences are measured in terms of the amount (in percent) by which the duration or magnitude of a release changes from LBU to HBU conditions.

6.1.1 Release Duration

The average duration of each release period and percent difference between the HBU and LBU values are shown in Table 28. The durations of early release period (i.e., prior to vessel breach) for HBU fuel are shorter than those for LBU fuel, indicating an acceleration of event progression due to the slightly higher levels of decay heat. This general observation, based on average release period durations, is consistent with trends observed in the comparison of the timing of key events for individual sequences. That is, coolant evaporation, fuel heatup and the early in-vessel phase of core damage all occur slightly earlier (or faster) in cases with HBU fuel than with LBU fuel. This trend is illuminated in greater detail in Table 29, which compares the timing of key events in the chronology of the short-term station blackout (ST-SBO) sequence at Peach Bottom.

Table 28 Comparison of HBU and LBU Release Timing

Release Period	Average Duration (hr)		Difference
	LBU	HBU	(%)
Coolant Release Phase	1.24	1.08	-13%
Onset of Release from fuel	3.48	3.26	-6%
Gap Release Phase	13.19	9.68	-27%
Early In-vessel Release Phase	9.13	8.16	-11%
Ex-vessel Release Phase	2.52	4.40	75%
Late In-vessel Release Phase	27.65	27.80	1%

Several factors contribute to the observed differences in event timing. Decay heat levels during the first 24 hours after reactor shutdown are 2 to 10% higher in an HBU core than those in an LBU core (refer to Table 7 and Table 8 in Section 2.3). In addition, the physical characteristics of fuel used in an HBU versus LBU core also contribute to the slight acceleration in event timing. For example, the total surface area of Zircaloy cladding in a core of 10x10 fuel assemblies loaded in a typical HBU core is approximately 20% greater than the surface area of an LBU core loaded with 8x8 fuel assemblies. The increased surface area enhances heat transfer, but can also accelerate the rate of clad oxidation within the core. The combination of these and other factors cause the acceleration in event progression observed in the BWR calculations for all types of sequences.

Table 29 Comparison of Event Timing for HBU vs LBU Cases of ST-SBO at Peach Bottom

Event (Time in hours unless noted otherwise)	LBU			HBU		
	Case 1A	Case 1C	Case 1D	Case 1A	Case 1C	Case 1D
	STSBO 8x8 CF@VB	STSBO 8x8 CF Late	STSBO 8x8 High P. CF@VB	STSBO 10x10 CF@VB	STSBO 10x10 CF Late	STSBO 10x10 High P. CF@VB
Station blackout [battery failure]	0 [0]	0 [0]	0 [0]	0 [0]	0 [0]	0 [0]
Downcomer water level reaches TAF	0.65	0.65	0.65	0.65	0.65	0.65
First hydrogen production	1.19	1.19	1.19	1.17	1.17	1.17
First fuel-cladding gap release	1.23	1.23	1.23	1.20	1.20	1.20
First channel box failure	1.65	1.65	1.65	1.48	1.48	1.48
First core cell collapse due to time at temperature [cell]	1.86 [112]	1.86 [112]	1.86 [112]	1.70 [212]	1.70 [212]	1.70 [212]
Reactor vessel water level reaches bottom of lower core plate	2.55	2.55	2.55	2.00	2.00	2.00
First core support plate localized failure in supporting debris	3.05	3.05	3.05	2.54	2.54	2.54
SRV sticks open	3.17	3.17	-	2.66	2.66	-
RPV pressure first drops below LPI setpoint (450 psig)	3.53	3.53	8.25	3.04	3.04	7.28
Lower head dries out	4.14	4.14	3.95	3.66	3.66	3.58
Ring 1 CRGT Column Collapse	6.77	6.77	8.96	4.86	4.86	7.23
Ring 2 CRGT Column Collapse	6.67	6.67	8.91	5.23	5.23	7.42
Ring 3 CRGT Column Collapse	6.57	6.57	9.01	5.76	5.76	7.50
Ring 4 CRGT Column Collapse	6.50	6.50	8.45	5.72	5.72	7.48
Ring 5 CRGT Column Collapse	6.91	6.91	8.88	6.15	6.15	7.81
Lower head failure	9.22	9.22	8.21	7.69	7.69	7.25
Drywell liner melt-through	9.45	-	9.59	8.08	-	8.09
Drywell head flange leakage begins	13.13	10.47	14.19	11.33	9.63	11.20

The acceleration of event progression (or conversely, the shortening of initial phases of radionuclide release) does not extend to the ex-vessel release period. As shown in Table 28, the average duration of the ex-vessel release period was found to be considerably longer in the HBU cases than the LBU cases. However, the larger average duration of the HBU ex-vessel release period is strongly influenced by an unusually long period of corium-concrete interaction in a few of the calculations performed for one of the two BWRs (Grand Gulf), and may not be a general trend[6].

The duration of late in-vessel (revolatilization) releases from the reactor vessel is nearly identical in the HBU and LBU simulations. This might appear counter-intuitive, given the higher level of decay heat associated with the HBU fission product inventory. However, the increases in decay heat shown in Table 5 and Table 6 primarily result from increases in decay heat of non-volatile species; in contrast the decay heat of volatile species (such as iodine and cesium) increases by a

[6] Also, longer ex-vessel release periods were observed in the Grand Gulf (BWR/6 Mark III) calculations, but not in the Peach Bottom (BWR/4 Mark I) calculations. Reasons for this difference are not clear.

very small amount. For example, decay heat for fission products in the Trivalent (Lanthanide) group, which represents approximately 30% of total decay heat, is 13% higher in the HBU core than the LBU core. In contrast, decay heat associated with Cesium Iodine (MELCOR group 16), which represents approximately 15% of the total decay heat, increases by only 3% from LBU to HBU conditions. Therefore, differences in decay heat associated with volatile fission products deposited on in-vessel surfaces are not sufficiently large to cause a significant difference in long-term revolatilization (late in-vessel) releases.

6.1.2 Radionuclide Release Fractions for Each Phase

Differences between the HBU and LBU release fractions for each phase of release are shown in Table 30. The values shown represent the arithmetic average of the MELCOR results for the full spectrum of sequences in both types of BWRs. Differences between the HBU and LBU gap and early in-vessel release fractions for all volatile species differ by ±10% or less; and the HBU versus LBU release fractions for the ex-vessel and late in-vessel release periods are typically within a factor of 2 (i.e., -50% < ΔRF < 100%) of each other.

Table 30 Comparison of Average Release Fractions for LBU and HBU Variations of BWR Sequences

Release Period	Ave. Release Fraction LBU	Ave. Release Fraction HBU	Difference (%)
Noble Gases (Xe)			
Gap Release	1.3E-02	1.2E-02	-11%
Early In-vessel Release	9.2E-01	9.6E-01	5%
Ex-vessel Release	5.9E-02	8.1E-03	-86%
Late In-vessel Release	6.8E-03	1.6E-02	129%
Halogens (I)			
Gap Release	8.9E-03	6.9E-03	-23%
Early In-vessel Release	5.4E-01	4.9E-01	-9%
Ex-vessel Release	4.8E-02	1.2E-02	-75%
Late In-vessel Release	3.1E-01	4.0E-01	31%
Alkali Metals (Cs)			
Gap Release	8.8E-03	6.4E-03	-27%
Early In-vessel Release	1.9E-01	1.9E-01	-5%
Ex-vessel Release	6.1E-02	9.4E-03	-84%
Late In-vessel Release	4.8E-02	5.5E-02	16%
Te Group			
Gap Release	8.8E-03	6.4E-03	-27%
Early In-vessel Release	4.2E-01	4.3E-01	2%
Ex-vessel Release	9.0E-03	2.5E-03	-72%
Late In-vessel Release	3.3E-01	3.7E-01	12%

Release Period	Ave. Release Fraction LBU	Ave. Release Fraction HBU	Difference (%)
Ba/Sr Group			
Gap Release	1.7E-03	1.2E-03	-28%
Early In-vessel Release	6.5E-03	6.0E-03	-7%
Ex-vessel Release	4.5E-02	5.4E-02	19%
Late In-vessel Release	5.8E-03	5.7E-03	-1%
Ru Group			
Gap Release			
Early In-vessel Release	3.8E-03	4.8E-03	25%
Ex-vessel Release	8.0E-10	3.7E-10	-54%
Late In-vessel Release	3.4E-05	1.7E-04	392%
Mo Group			
Gap Release			
Early In-vessel Release	4.4E-02	4.3E-02	-4%
Ex-vessel Release	1.6E-02	2.5E-03	-85%
Late In-vessel Release	7.3E-03	6.8E-03	-6%
La Group			
Gap Release			
Early In-vessel Release	2.4E-07	3.6E-07	50%
Ex-vessel Release	2.1E-04	1.2E-04	-43%
Late In-vessel Release	5.3E-08	3.3E-08	-39%
Ce Group			
Gap Release			
Early In-vessel Release	2.4E-07	3.5E-07	43%
Ex-vessel Release	2.1E-04	6.4E-03	2932%
Late In-vessel Release	4.5E-07	4.3E-06	847%

6.2 Comparisons to NUREG-1465 Source Term Prescription

The quantity of empirical data and fidelity of analytical tools (computer codes) used to perform deterministic calculation of radiological sources have matured substantially since the publication of NUREG-1465. In particular, MELCOR (and the extensive experimental validation that lies behind it) reflects the current state-of-the-art in source term analysis. Therefore, it's useful to examine how the MELCOR results described in this report compare to the prescription outlined in the NRC's principal document for defining source terms used in regulatory applications.

The format in which this comparison is made is statistical. That is, each of the MELCOR results is treated as a single, representative sample of a distribution of possible, generic BWR results. If a particular result from a MELCOR calculation is treated as a random value from a distribution of possible values, and if the result from each MELCOR calculation is assumed to be equally probable, a distribution can be generated which depicts the range of credible values and identifies useful statistical quantities such as the median value. The distribution is, therefore, generated by organizing a particular set of MELCOR results in rank order (from lowest to highest) and assigning each value an equal probability of 1/(number of samples). The distribution is then plotted and compared to the point value specified in NUREG-1465. General observations are then made regarding the location of the NUREG-1465 value in comparison to the MELCOR distribution. A separate report will apply order statistics to explore more fully the trends implied from the observations made from this simple exercise.

6.2.1 Timing and Duration of NUREG-1465 Phases

Distributions for temporal metrics of radiological release in NUREG-1465 are shown in Figure 11 through 17. In particular, these figures show the calculated duration of the initial phase of reactor coolant release, the time at which fission product release from fuel begins, and the duration of the four sequential phases of fission product release to containment: gap release, in-vessel release, ex-vessel release and late in-vessel (re-volatilization) release. In each figure, the distribution is compared to the value specified in NUREG-1465.

In the vast majority of cases, the calculated duration of reactor coolant activity release to containment is less than one hour. Recall the duration of the coolant release phase is defined as the time interval between initial core uncovery (water level below TAF) and the onset of gap release (first occurrence of clad temperature greater than 1170 K.) As shown in Figure 11, the median duration of this phase is 0.6 hr (35 min), and is observed in most of the calculations. Coolant discharge to containment is faster (8 min) for large LOCA sequences, as one might expect. Release duration is much longer in only one sequence; i.e., an ATWS sequence, in which core cooling is maintained by intentionally reducing reactor water level below top of active fuel. This situation is atypical of most severe accident sequences. The duration of coolant activity release prescribed in NUREG-1465 is conservatively defined to be 30 seconds, based primarily on the prescription of a design basis LOCA.

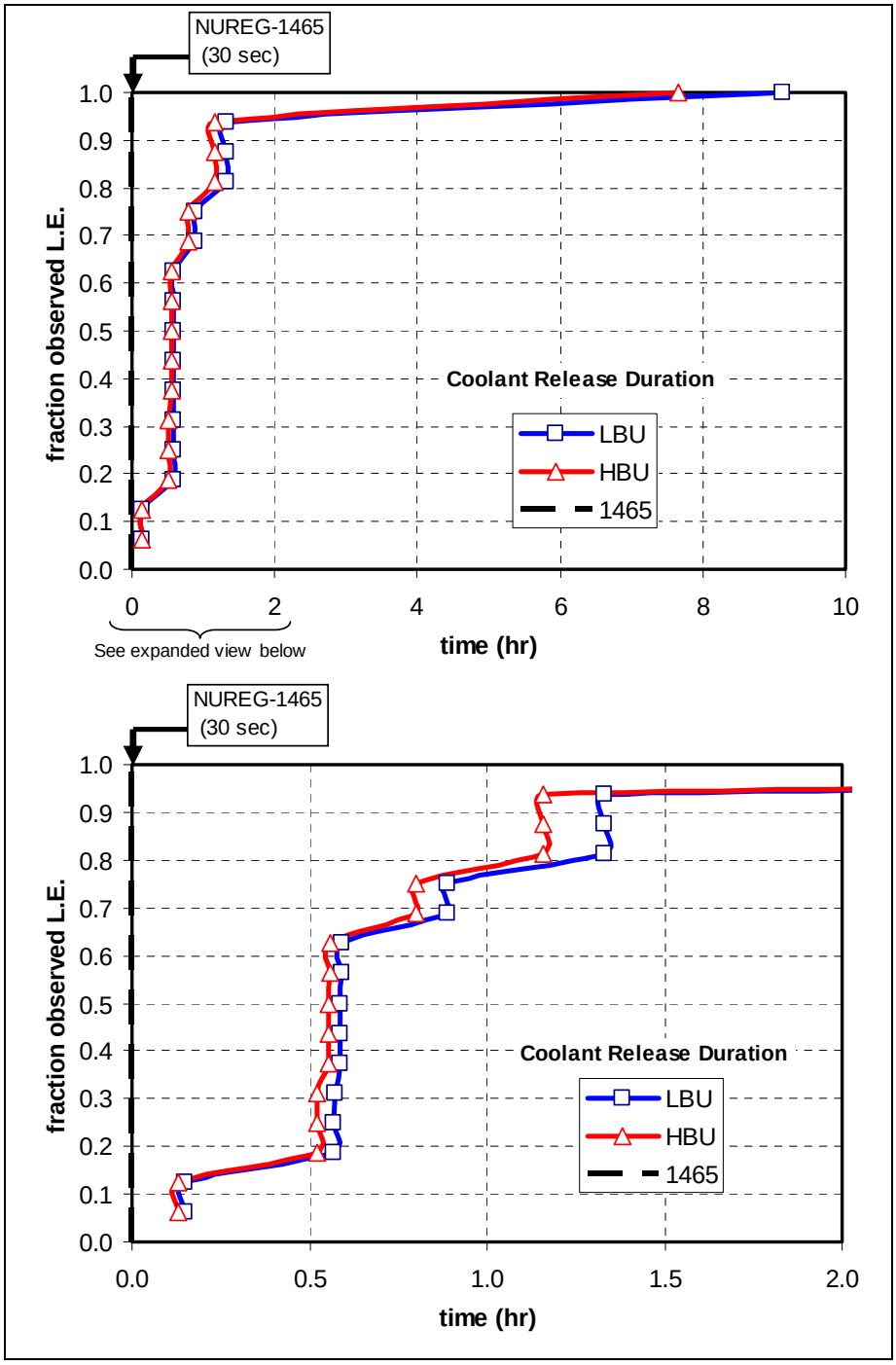

Figure 11 Duration of Coolant Release Phase (fraction of observations less than or equal to (LE) duration in hours)

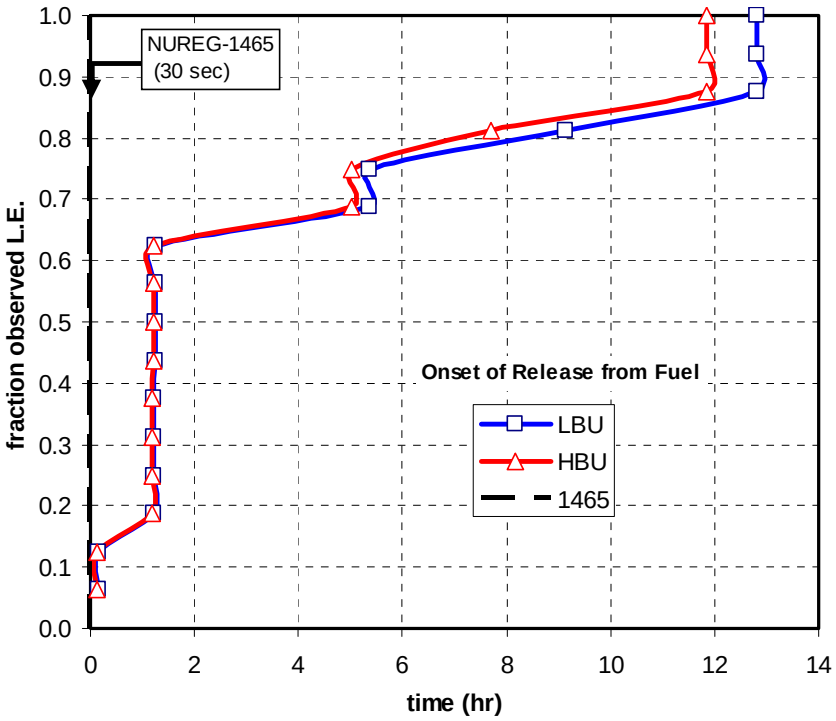

Figure 12 Time of the Onset of Fission Product Release from Fuel (clad failure)

The onset of fission product release from fuel is defined as the time at which the fuel cladding failure first occurs, releasing the gap inventory of fission products to reactor coolant. The time of this event is conservatively defined as 30 seconds in NUREG-1465, which corresponds to the end of the coolant activity release phase. In the current study, however, an alternative definition of the duration of the coolant activity phase is used (compare Figure 1 to Figure 10), which requires the time of clad failure and gap release to be calculated in an absolute context. That is, the onset of fission product release from fuel is calculated as time at which gap release occurs after the initiating event.

The calculated time at which gap release occurs is shown in Figure 12 to span a very wide range, from a lower limit of approximately 8 minutes to over 12 hours. This wide range is a direct reflection of the wide spectrum of scenarios considered. At one end of this range is the large-LOCA sequence in which all coolant injection systems fail to operate. This would clearly lead to the fastest time of release. At the other extreme is a long-term station blackout scenario in which a steam-driven coolant injection system operates for several hours until dc power provided by the station batteries is exhausted. The onset of fission product release is, therefore, strongly dependent on the accident scenario assumed.

The distinguishing characteristic of an accident sequence, in terms of estimating the time at which fission product release would be begin, is the time at which coolant injection terminates and the reactor water level decreases below the top of active fuel (TAF). The latter depends entirely on the system details of the accident sequence and is not influenced by severe accident phenomenology. However, once this time is known, the MELCOR calculations described here provide a deterministic basis for forecasting the time at which fission product (gap) release

occurs. For example, Figure 13 indicates a simple linear relationship between the time at which gap release occurs and the time at which level decreases below TAF [7]. The relationship shown in Figure 13 could be used to 'shift' the timeline for applications of the NUREG-1465 source terms to accident sequences with significant delays in the onset of core damage (e.g., transient events, rather than large-break LOCAs.)

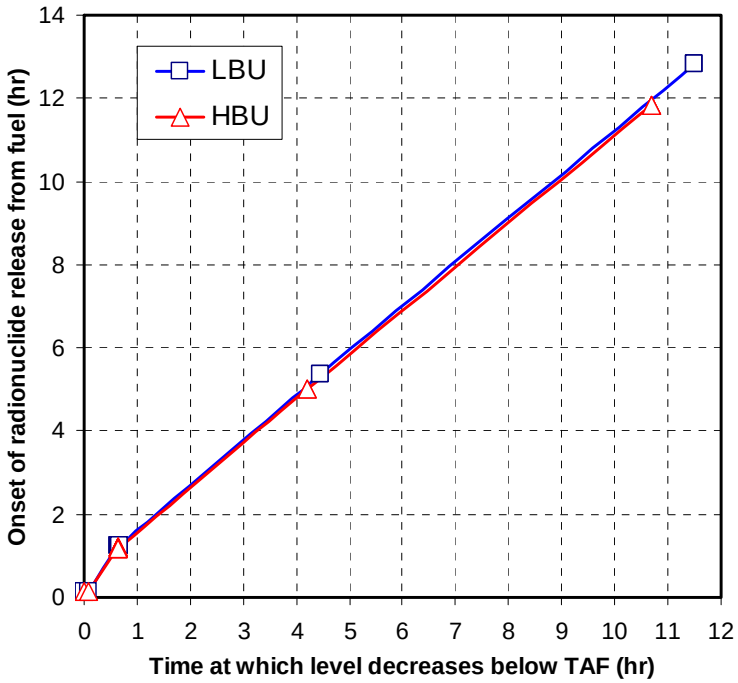

Figure 13 Correlation Between Timing of Core Uncovery and Gap Release

Transport and release of the gap inventory to containment is specified in NUREG-1465 occur over a 30 minute period. As indicated in Figure 14, however, typical MELCOR calculations indicate this release is accomplished in a shorter time frame (5 to 25 minutes, with a median value of 10 minutes.) This result must be tempered with the understanding the release of the gap inventory of fission products from fuel occurs instantaneously in MELCOR. That is, when fuel failure occurs, the entire inventory of fission products within the entire radial ring of the core[8] is released to reactor coolant without delay or holdup due to resistances within fuel rods. In addition, spatial (radial) resolution of the core used in the MELCOR model is rather coarse (five radial rings). A finer spatial resolution would conceivably stretch out the amount of time over which the clad failure (and therefore release of the gap inventory) would occur.

[7] Some minor corrections to the linear relationship are required if core uncovery occurs very quickly (i.e., within one hour of the initiating event.)

[8] As described in Section 2.3, the MELCOR models used in these calculations represent the core in five concentric rings. Therefore, release of the gap inventory occurs at five discrete points in time.

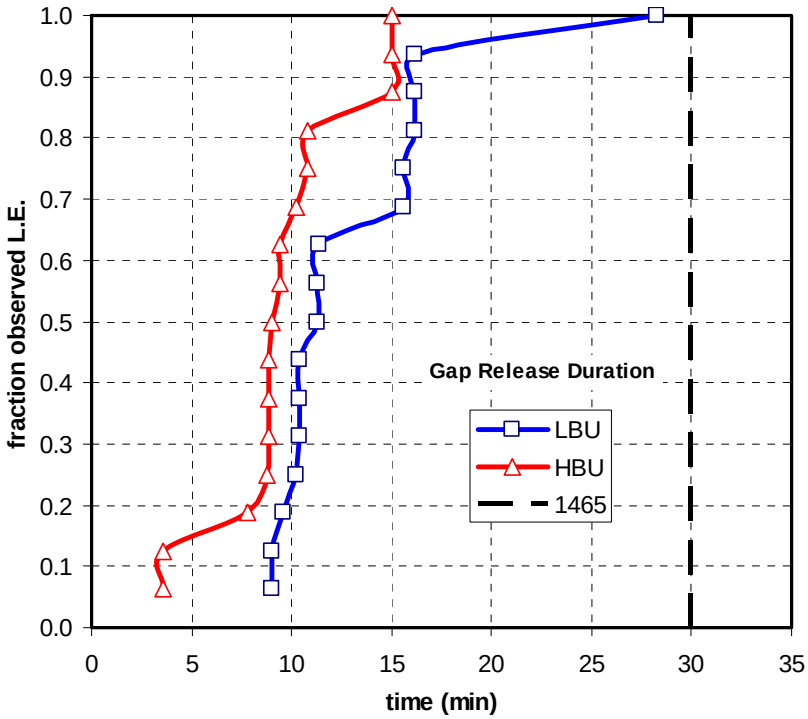

Figure 14 Duration of Gap Release Phase

The duration and magnitude of the in-vessel release phase are arguably the most important aspects of the NUREG-1465 prescription in terms of impact on typical regulatory applications. NUREG-1465 considers the duration of this release period to be 1.5 hours in length. As indicated in Figure 15, however, the MELCOR calculations suggest the actual duration is considerably longer. The minimum observed release duration is approximately six hours at results from various permutations of accident sequences involving a large (design basis) LOCA and coincident loss of ac power. The median release duration is approximately 7.5 hours and the maximum observed duration is 12 hours. This suggests the NUREG-1465 prescription for the in-vessel release could period could be extended by several hours, and it would still remain demonstrably conservative in terms of the earliest time at which a large fraction of the core inventory would be released to the containment.

One reason for the longer in-vessel release period is the protracted period of in-vessel accident progression in the current MELCOR calculations in comparison to similar results from the Source Term Code Package (STCP) calculations that contributed to the development of the NUREG-1465 prescription. For example, results of STCP calculations of short- and long-term station blackout sequences in Peach Bottom were reported in Ref. 28. In all three cases, the reported duration of the in-vessel release period is approximately 2 hours in length[9]. These results are consistent with the short duration specified in NUREG-1465, but reflect an outdated analysis of in-vessel core damage progression. New information from integral experimental programs, such as Phebus, and refined analytical models, such as those in MELCOR version

[9] This represents an average of the results for STCP calculations of the "TB1," "TB2" and "TBUX" accident sequences.

1.8.4 and beyond, clearly indicate the time window between onset of fuel damage and reactor vessel lower head failure is significantly longer than 2 hours. Fission product release to the containment occurs throughout this entire 'in-vessel' period and, as indicated in Figure 15, durations less than 5 hours are not predicted [10].

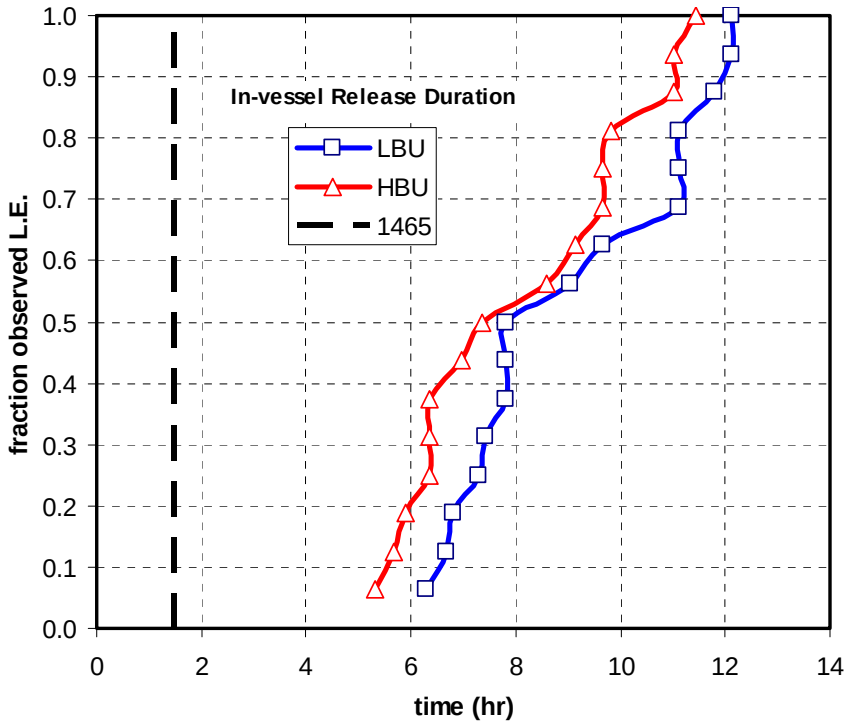

Figure 15 Duration of In-vessel Release Phase

In contrast to the observations of the duration of the in-vessel release phase, the calculated duration of the ex-vessel release phase appears fully consistent with the NUREG-1465 specification. Figure 16 suggests the 3-hour duration in NUREG-1465 slightly higher (longer) than the median value calculated by MELCOR. The MELCOR results span an effective range of one to ten hours.

The final phase of radionuclide release to the containment (late in-vessel) is dominated by evaporation and transport of volatile fission products from reactor coolant system surfaces to cooler locations in containment. NUREG-1465 defines this release to occur over a ten hour duration. This value is confirmed in the MELCOR calculations, which are shown in Figure 17 to have a median value of approximately 11 hours[11]. However, durations vary from 1 hour to 20

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[10] Results of MELCOR calculations for large LOCA sequences in both BWRs studied here (see Appendix C.9 and C.16) indicate this statement is holds for the design-basis LOCA event, coupled with a loss of coolant injection.

[11] This value would reduce to 10 hours if two aberrant cases are discarded or interpreted differently. The MELCOR calculations for two cases 5C and 6B generated extraordinarily long durations of the late in-vessel release phase. Both of these calculations involved a station blackout accident sequence at Grand

hours are observed, depending on details of the fission product deposition patterns prior to vessel breach and thermal conditions post vessel breach.

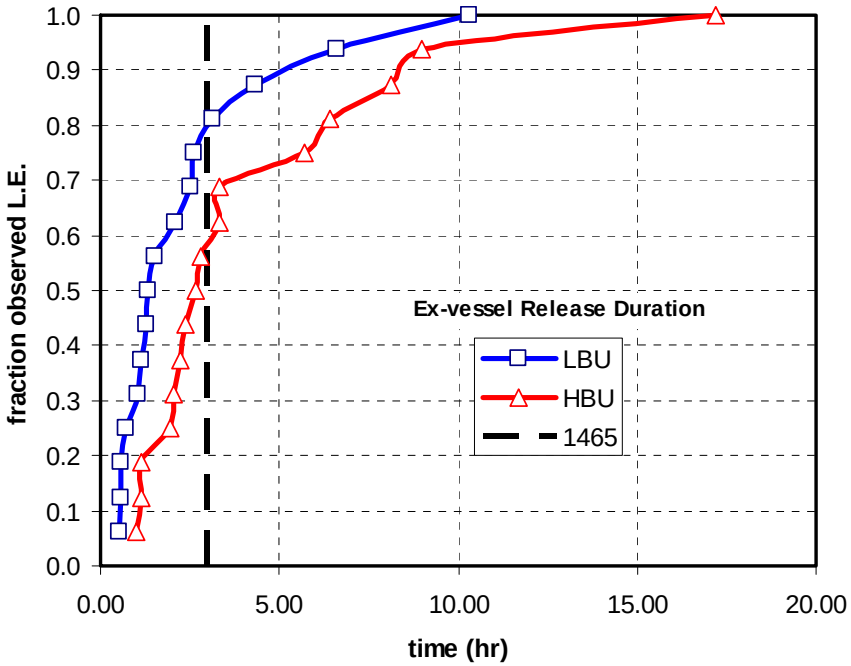

Figure 16 Duration of Ex-vessel Release Phase

Gulf with late containment failure by static over-pressure. The late release of iodine and tellurium from the reactor coolant system (RCS) to the containment had effectively stabilized within 10 hours of vessel breach in both calculations. However, a second period of revolatilization was observed after the containment and the RCS depressurized. The long duration of the 'late in-vessel release' period reported in Table 19 for these cases (140 to 145 hours) represents the combination of both release periods.

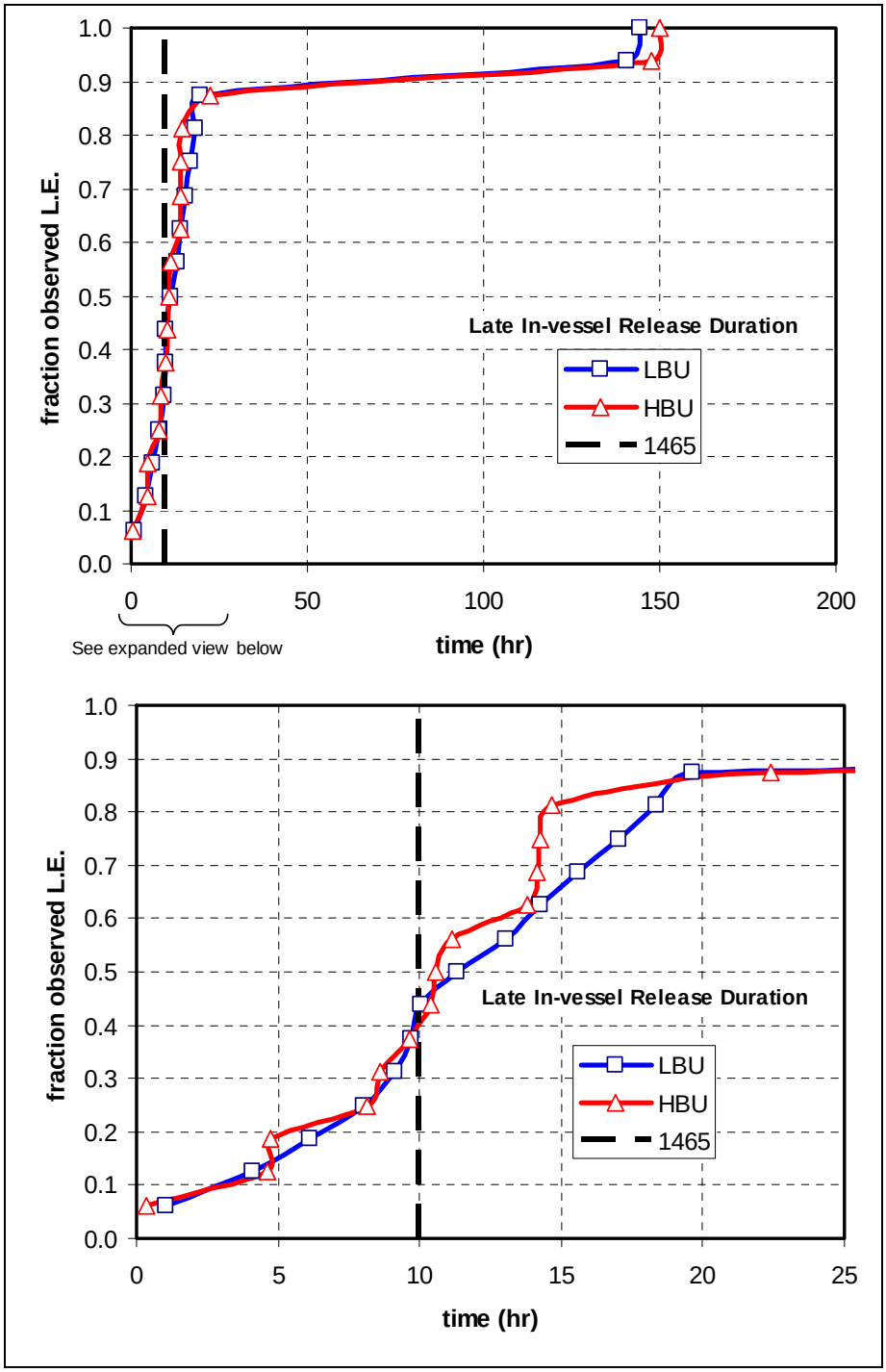

Figure 17 Duration of Late In-vessel Release Phase

6.2.2 Release Fractions During NUREG-1465 Accident Phases

The quantity of fission products released during each of the four sequential phases described in NUREG-1465 is calculated as a continuous, time-dependent result of the MELCOR simulations. Therefore, the time-integral quantity of radionuclides released during each of the phases is calculated from the MELCOR results, as tabulated in Section 5.2. These quantities are compared to the NUREG-1465 values in a series of figures for each release period:

- Gap Release: Figure 18 and Figure 19
- In-vessel Release: Figure 20 through Figure 27
- Ex-vessel Release: Figure 28 through Figure 31
- Late In-vessel Release: Figure 32 through Figure 37

Gap Release Fractions

The NUREG-1465 prescription provides for 5% of the noble gases, iodine and cesium inventory to be released to the containment during the gap release phase. This essentially represents the entire gap inventory of these species and neglects credit for deposition of this material in the reactor coolant system during the release period. Results of the MELCOR calculations clearly indicate a significant attenuation of iodine and cesium released from the fuel-cladding gap. As indicated in Figure 18 and Figure 19, a maximum of 4% release is predicted and the median value is less the one-half of one percent. The remainder of the gap inventory is retained on in-vessel structures in the MELCOR calculations.

In-vessel Release Fractions

The in-vessel release fractions are arguably the most important quantity specified in NUREG-1465. Analyses supporting most regulatory applications of the alternative source term, rely heavily on radiological releases that occur during this phase of a postulated event. Therefore, particular attention was placed on the MELCOR results for this phase of the calculations. In particular, results for a wider spectrum of radionuclide groups are examined here.

The in-vessel release fractions in NUREG-1465 for iodine and cesium are 0.25 and 0.20, respectively. As shown in Figure 20 and Figure 21, these values lie near the middle of the distribution of results obtained from MELCOR. The median MELCOR estimate for iodine (approx. 0.45) is roughly twice the NUREG-1465 value; the estimate for cesium (approx. 0.13) is slightly lower than the NUREG-1465 value.

It is also worth noting that the range of MELCOR release fractions for iodine span a significantly wider range than those for cesium, with the 90th percentile reaching a value of 0.8, whereas the 90th percentile for cesium is less than 0.5. This difference is due to changes in the chemical forms of cesium represented in the current MELCOR calculations from past analyses. That is, a portion of the core inventory of cesium is assumed to react with iodine to form CsI (consistent with all prior MELCOR analysis.) However, the remaining mass of cesium (nearly 90% of the total core inventory[12]) is assumed to react with molybdenum (Mo) to form cesium molybdate

[12] Only 10% of the total cesium inventory in the core is required to completely react the inventory of radioiodine and form CsI.

(Cs_2MoO_4). This change in speciation is based on relatively recent observations of differences the measured iodine and cesium distribution within the RCS in the Phebus test program. Cs_2MoO_4 is slightly less volatile than CsI, and therefore requires higher temperatures within the RCS to survive the long and tortuous transport pathway between the core and containment atmosphere.

Early differences in the distribution of CsI versus Cs_2MoO_4 are evident in Figure 38, which indicate the fraction[13] that is retained within the RCS and the fraction released to containment. This figure is developed from results of one example calculation (Case 1A for Peach Bottom), but the trends are representative of results of many of the other BWR calculations. Although both species are partially retained in the RCS during the in-vessel release period, CsI clearly exhibits a greater affinity for release from the RCS and subsequent transport to the containment. In this particular case, the retention of Cs_2MoO_4 on in-vessel structures is more than twice as effective as CsI. A detailed examination of deposition behavior of the two species reveals that the initial deposition rates of major structures, such as the steam separators and dryers above the core, are very similar. However, CsI re-evolves (evaporates) from these surfaces as their temperatures increase during the late phases of in-vessel fuel damage, whereas Cs_2MoO_4 evaporates at a much slower rate. This magnitude of this difference in re-vaporization behavior varies among different accident scenarios due to differences in the temperature history of major in-vessel structures.

Agreement between the calculated in-vessel release fractions for other species and their NUREG-1465 quantities vary considerably. The NUREG-1465 prescription significantly understates the calculated release fractions for tellurium and molybdenum, for example. In contrast, the calculated release fractions for the barium group (which includes strontium), lanthanum and cerium were all found to be significantly lower than the NUREG-1465 values. The calculated release fraction for ruthenium is the only group that demonstrated good agreement with the NUREG-1465 value.

Poor agreement for tellurium and molybdenum are likely due to changes in information regarding their chemical speciation. At the time the analyses were conducted to develop NUREG-1465, expert opinion tended to favor high levels of in-vessel retention for tellurium due to a (judged) reaction with tin alloying material in unoxidized Zircaloy cladding. Similarly, molybdenum was typically treated as a non-reactive metal with very low volatility. Neither of these views is supported as strongly as they were in the past, and data obtained from Phebus and other integral experimental programs is motivating changes toward the formation of more volatile chemical forms.

Reasons for the significant disagreement in the release fractions of non-volatile lanthanum and cerium are not clear. These species do not receive much attention in in-pile fission product release experiments, offering little data to support either the current MELCOR results or those of older calculations underpinning NUREG-1465. However, the observation that the median

[13] Unlike other figures in this report, which present radionuclide release information in terms of fraction of initial core inventory, this particular figure shows the fraction of the iodine/cesium mass released from fuel. Therefore, the sum of the two curves for CsI and for Cs_2MoO_4 sum to unity.

release fractions calculated by MELCOR for these species are three orders of magnitude lower than the NUREG-1465 prescription suggests some attention to these species may be warranted.

Ex-vessel and Late In-vessel Release Fractions

Comparisons of the calculated release fractions for late phases of a severe accident are tied to the observations noted above for the early in-vessel period. That is, the quantity of fission products available for release to containment after vessel breach depends strongly on the amount released prior to vessel breach. This interdependence is most clearly observed by examining the ex-vessel release fractions for iodine and cesium (Figure 28 and Figure 29, respectively.) Over 95% of the initial core inventory of both species is released from fuel during the in-vessel phase of a severe accident. As a result, very little is left for release from fuel (directly to containment) during the ex-vessel phase. This relationship is clearly reflected in the low calculated release fractions for both species. In contrast, the NUREG-1465 prescription reflects a much lower in-vessel release fraction (presumably reflecting less complete release from fuel), and thus a greater quantity available for release ex-vessel.

Ex-vessel releases of heavy metals, such as lanthanum (Figure 30) and cerium (Figure 31) show mixed results when compared to their NUREG-1465 release fractions. The NUREG-1465 release fraction for lanthanum is nearly a factor of 10 higher than the largest MELCOR-calculated release fraction, and a factor of 100 higher than the median MELCOR value. In contrast, the release fraction for cerium is within a factor of 2 of the MELCOR median.

A significant delayed release of some volatile species to the containment does occur in the MELCOR calculations, but not from ex-vessel processes [e.g., release from fuel due to molten corium-concrete interactions (MCCI).] Rather, the release occurs as a result of evaporation and transport from RCS surfaces after vessel breach. The magnitude of late in-vessel releases varies considerably among the "volatile" fission product species (refer to Figure 32 through Figure 35.) For example, iodine and tellurium are released in relatively large quantities (i.e., median release fractions greater than 0.2), whereas cesium and barium/strontium are not (median release fractions less than 0.05). [The difference between iodine and cesium behavior is again reflected in Figure 38; in contrast to the earlier reference to this figure[14], the focus is on changes in the release fraction to containment after vessel breach – i.e., the right-hand side of the figure.] Nevertheless, the calculated late in-vessel release fractions for all volatile species are significantly greater than the values recommended in NUREG-1465.

[14] Refer back to discussion of iodine/cesium in-vessel release fractions.

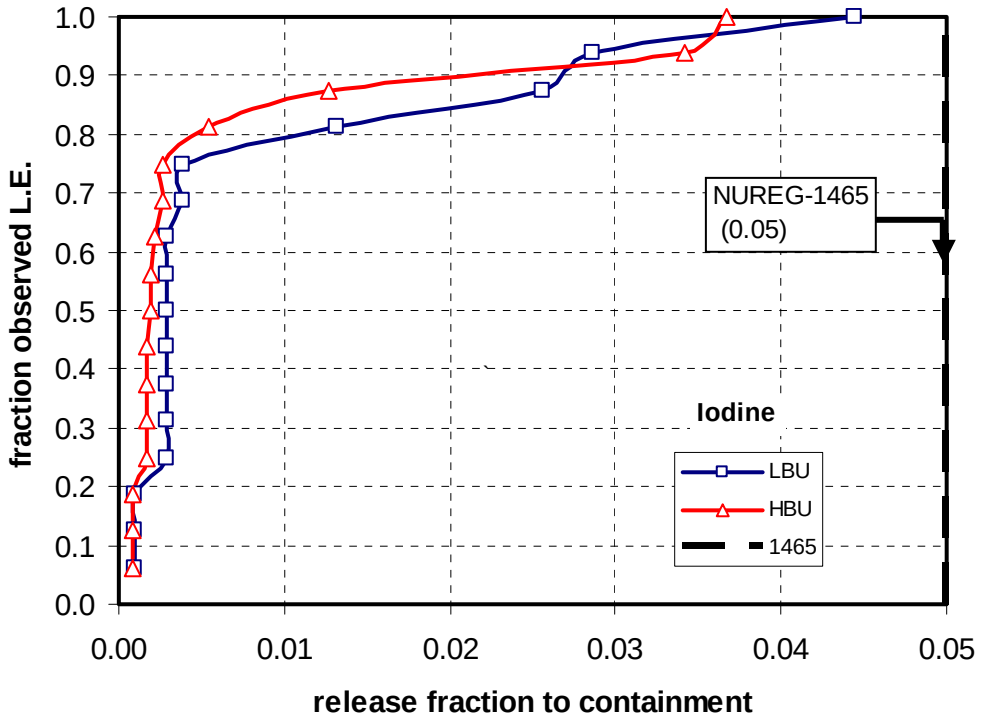

Figure 18 Gap Release Fraction for Iodine

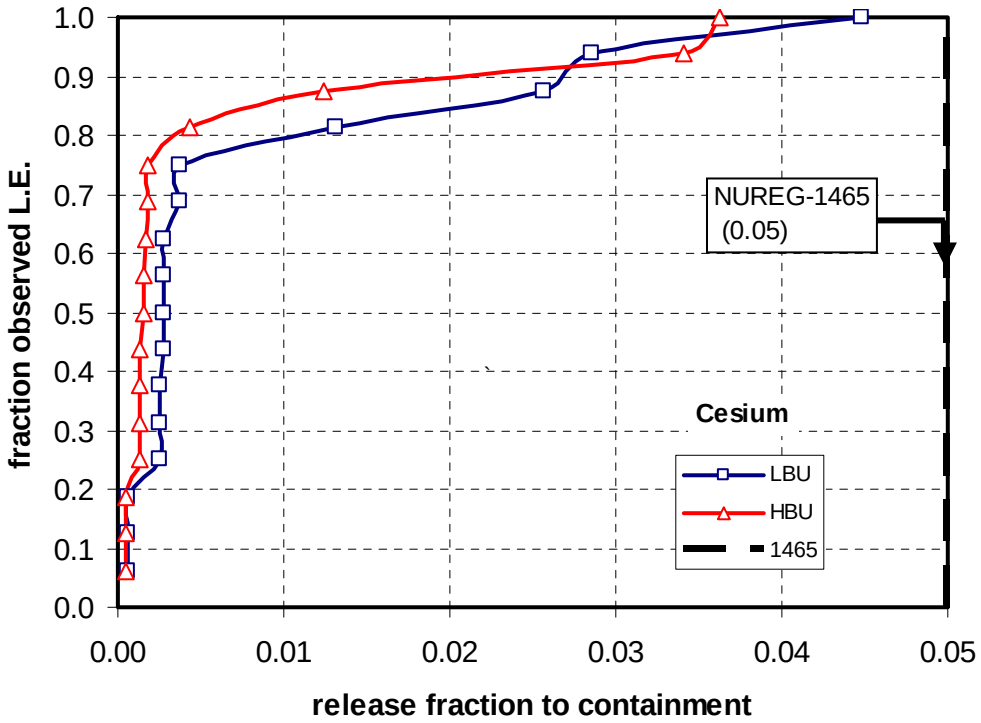

Figure 19 Gap Release Fraction for Cesium

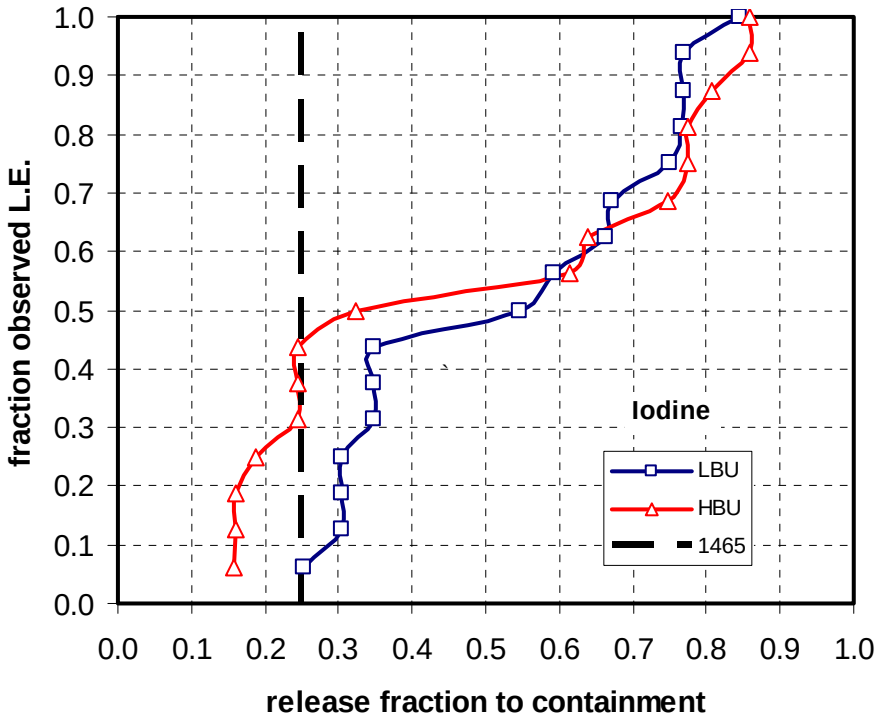

Figure 20 In-vessel Release Fraction for Iodine

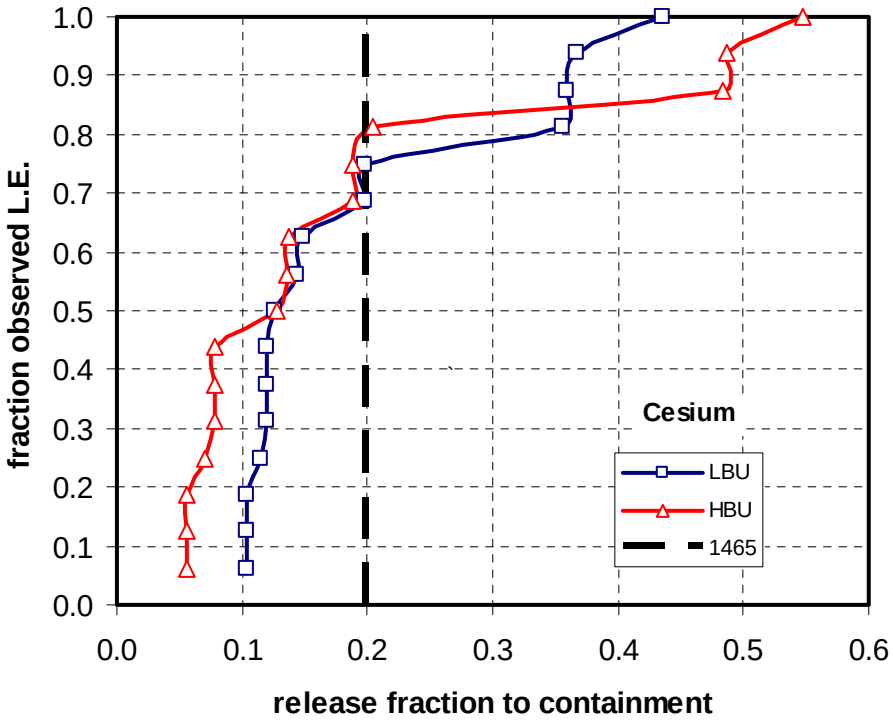

Figure 21 In-vessel Release Fraction for Cesium

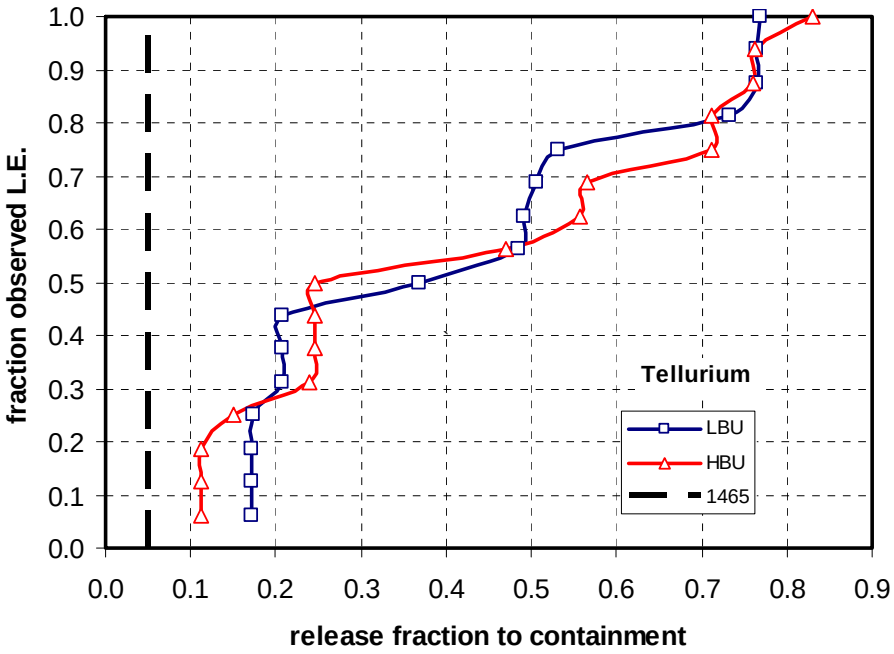

Figure 22 In-vessel Release Fraction for Tellurium

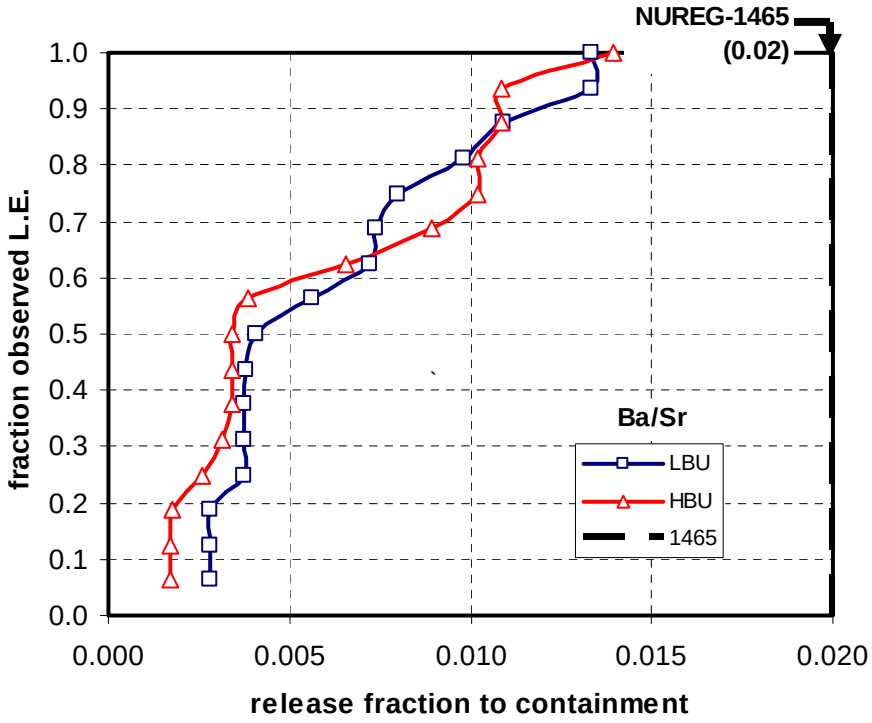

Figure 23 In-vessel Release Fraction for Barium/Strontium

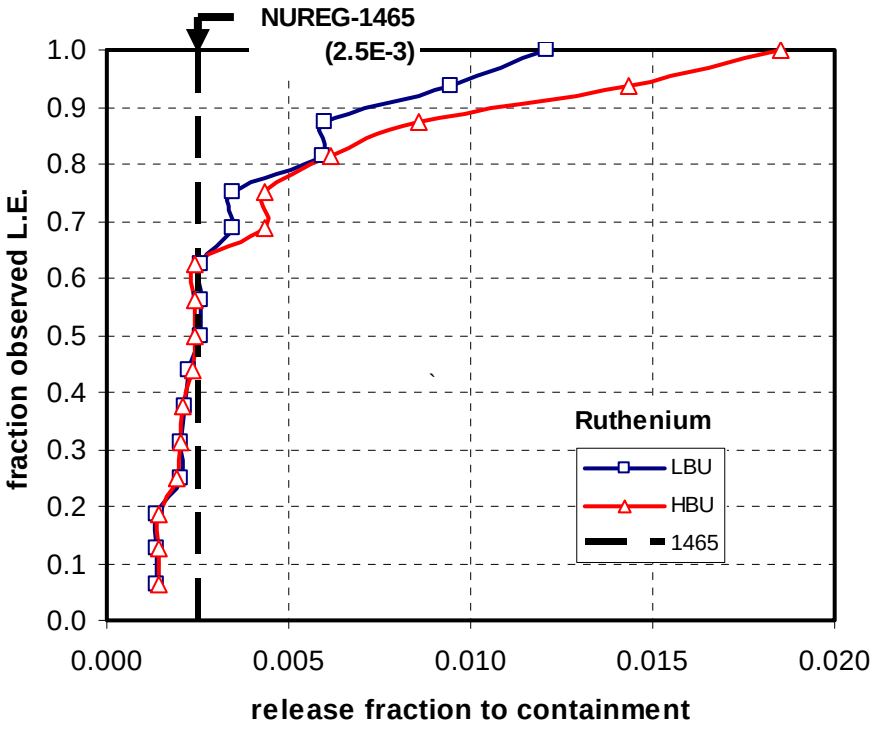

Figure 24 In-vessel Release Fraction for Ruthenium

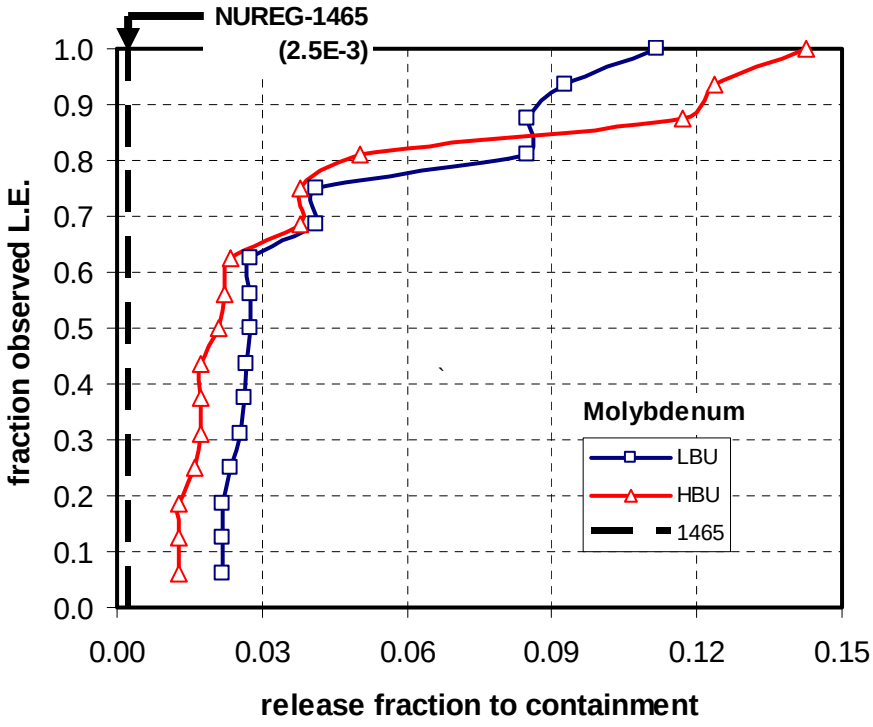

Figure 25 In-vessel Release Fraction for Molybdenum

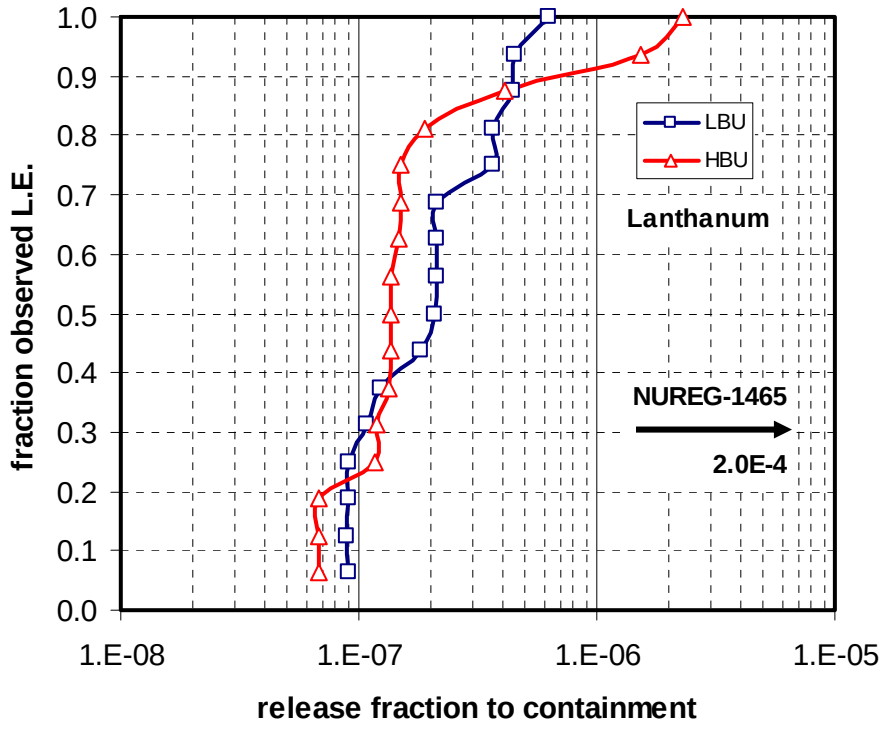

Figure 26 In-vessel Release Fraction for Lanthanum

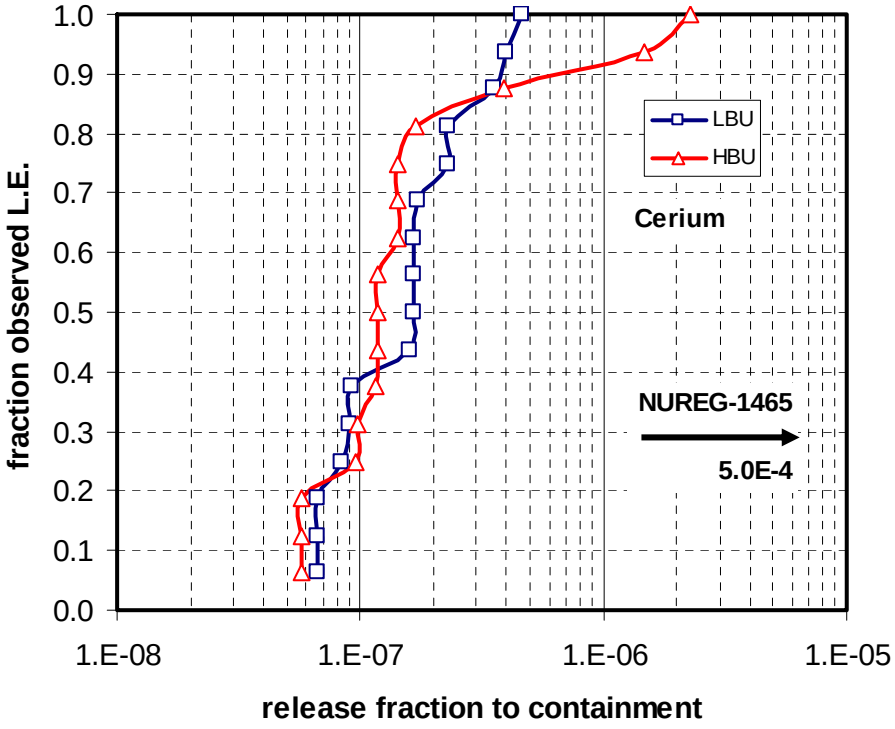

Figure 27 In-vessel Release Fraction for Cerium

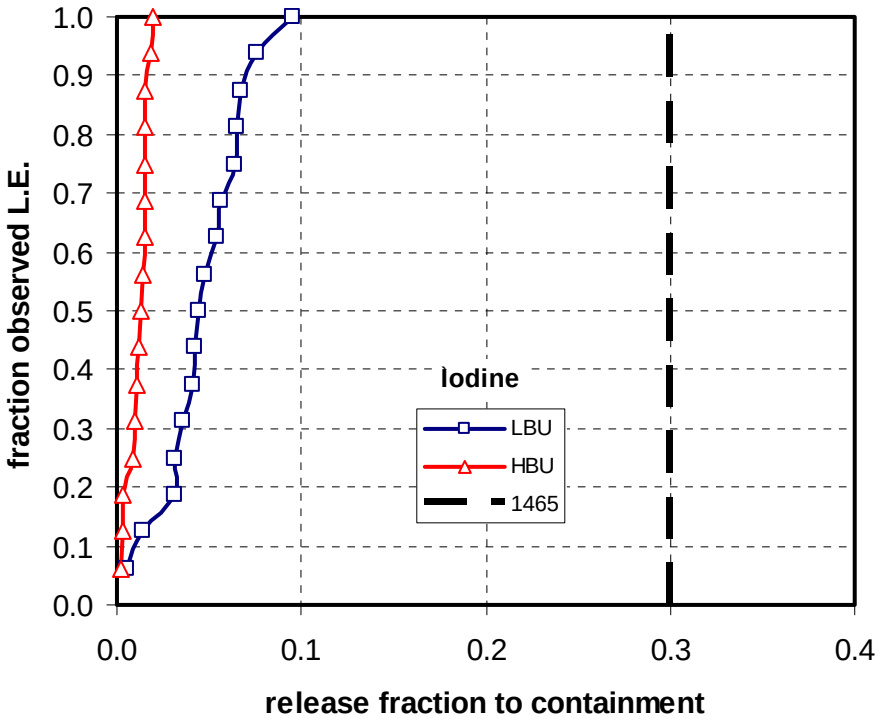

Figure 28 Ex-vessel Release Fraction for Iodine

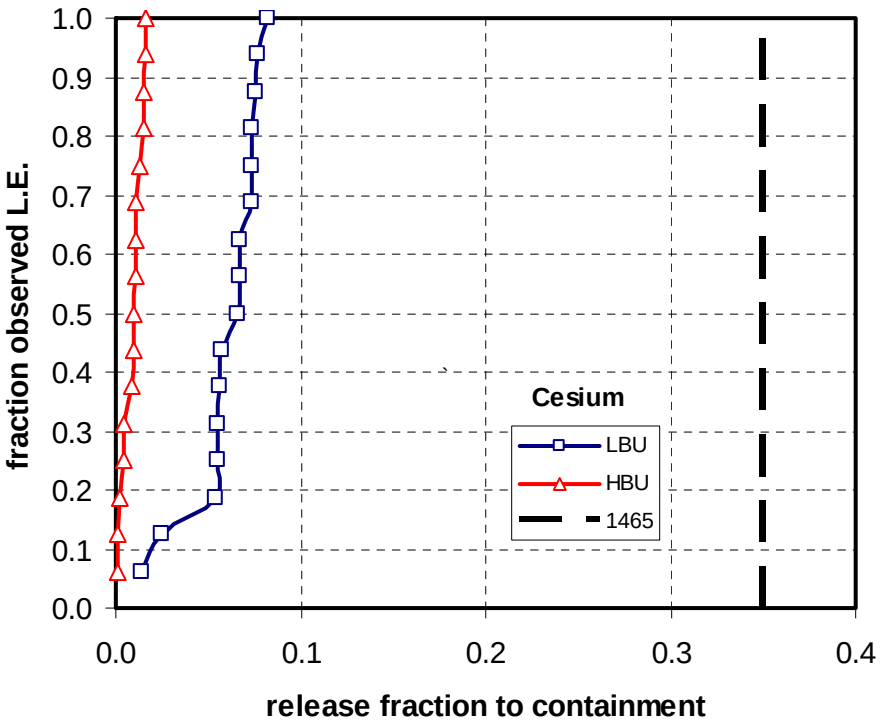

Figure 29 Ex-vessel Release Fraction for Cesium

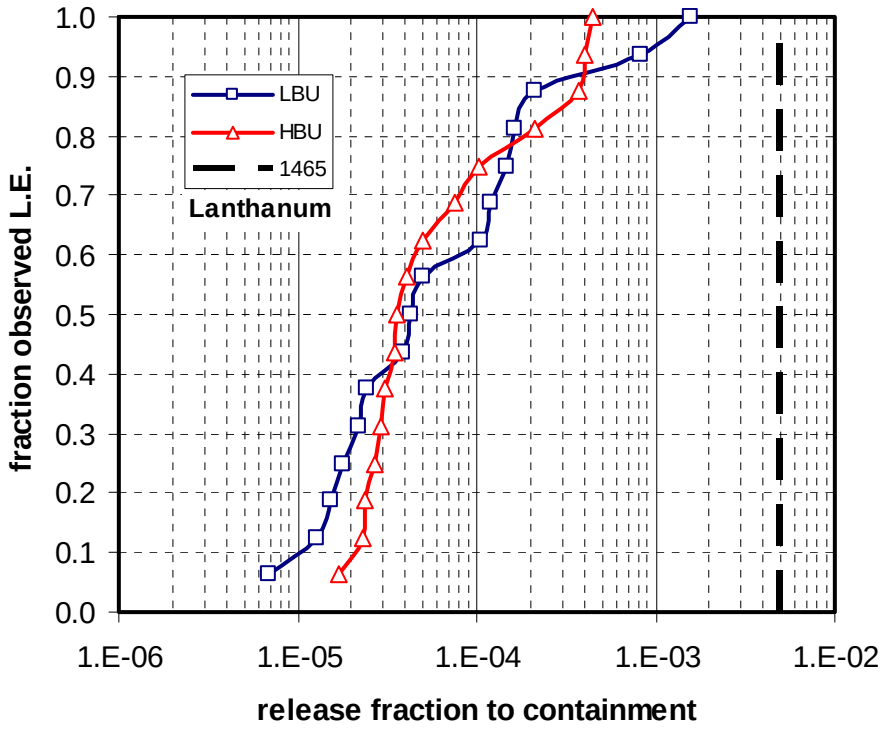

Figure 30 Ex-vessel Release Fraction for Lanthanum

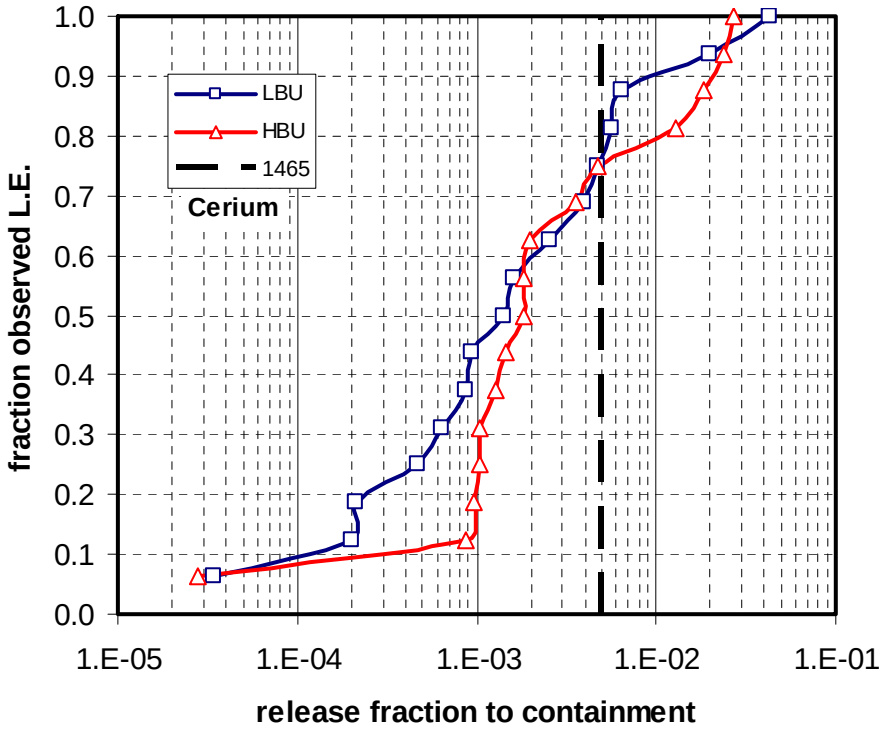

Figure 31 Ex-vessel Release Fraction for Cerium

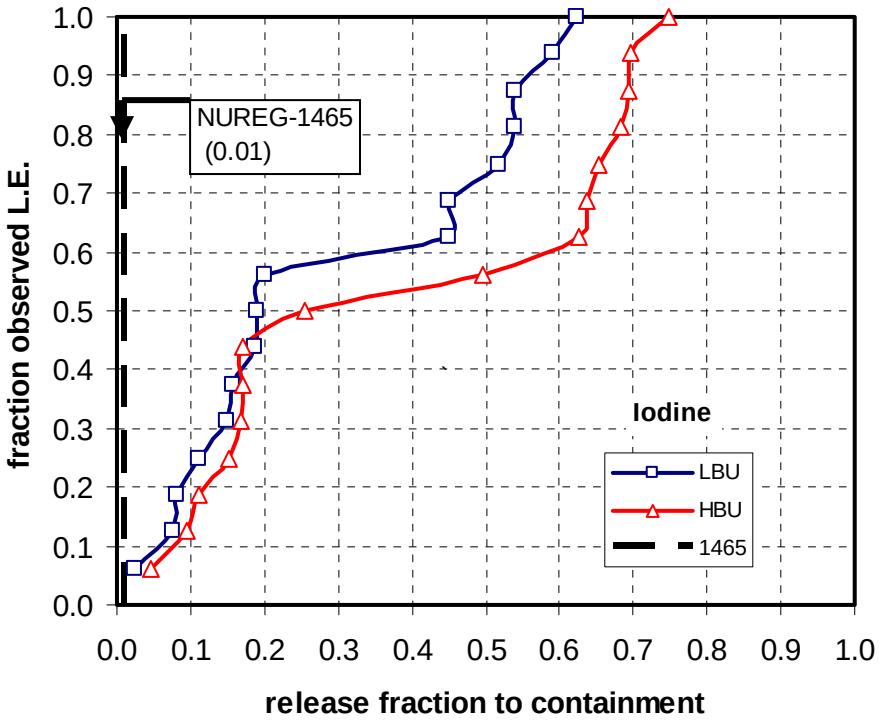

Figure 32 Late In-vessel Release Fraction for Iodine

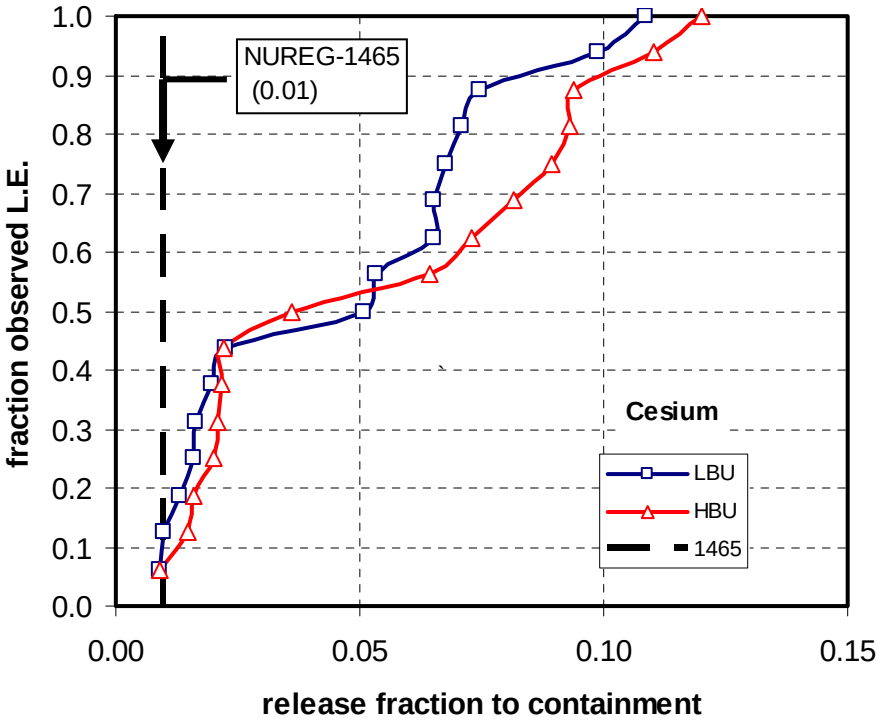

Figure 33 Late In-vessel Release Fraction for Cesium

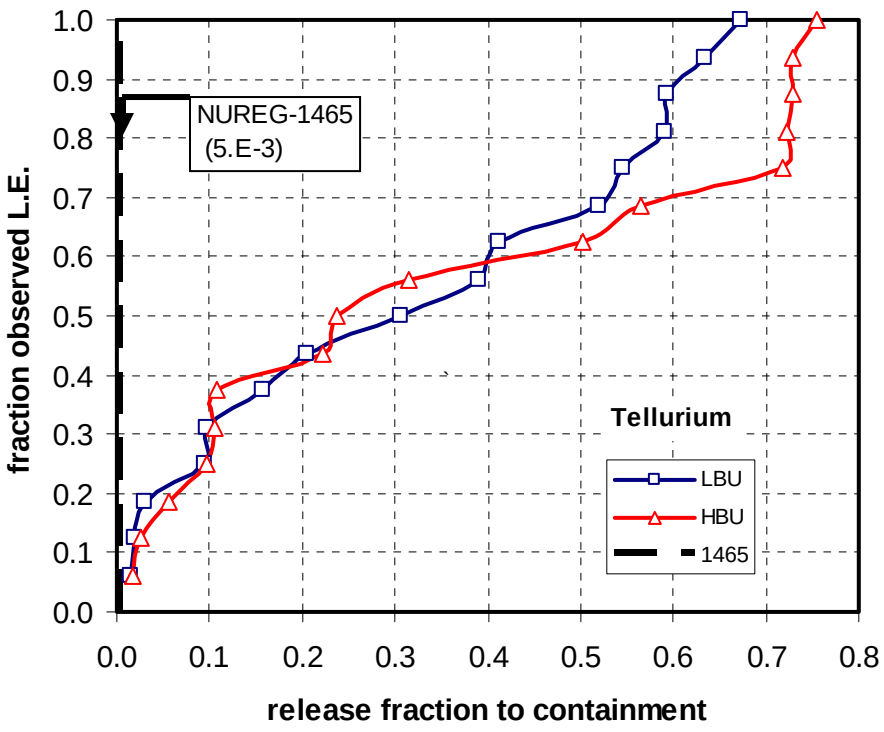

Figure 34 Late In-vessel Release Fraction for Tellurium

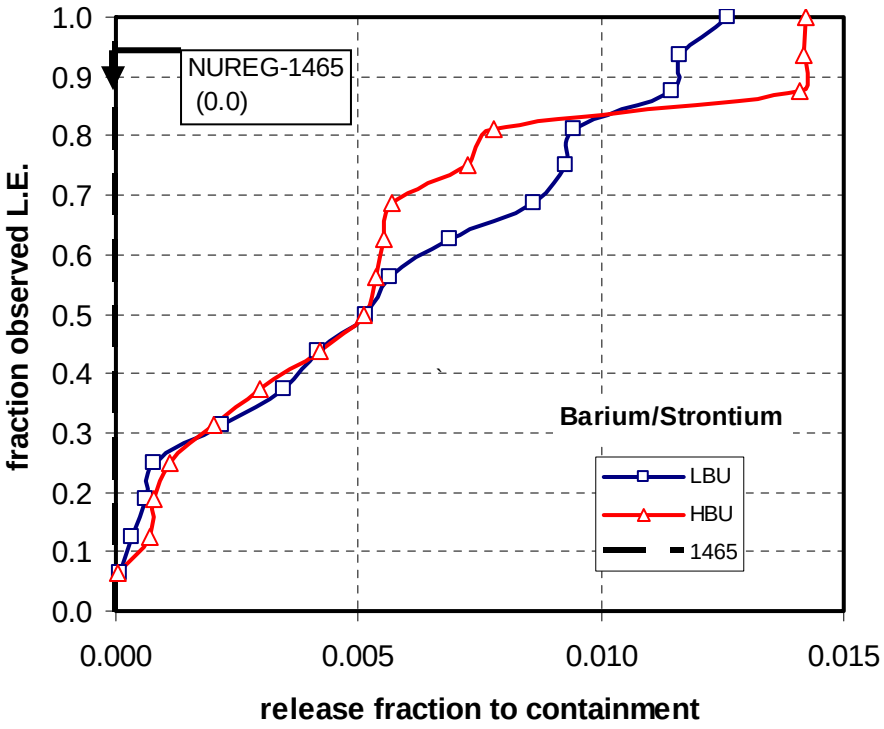

Figure 35 Late In-vessel Release Fraction for Barium/Strontium

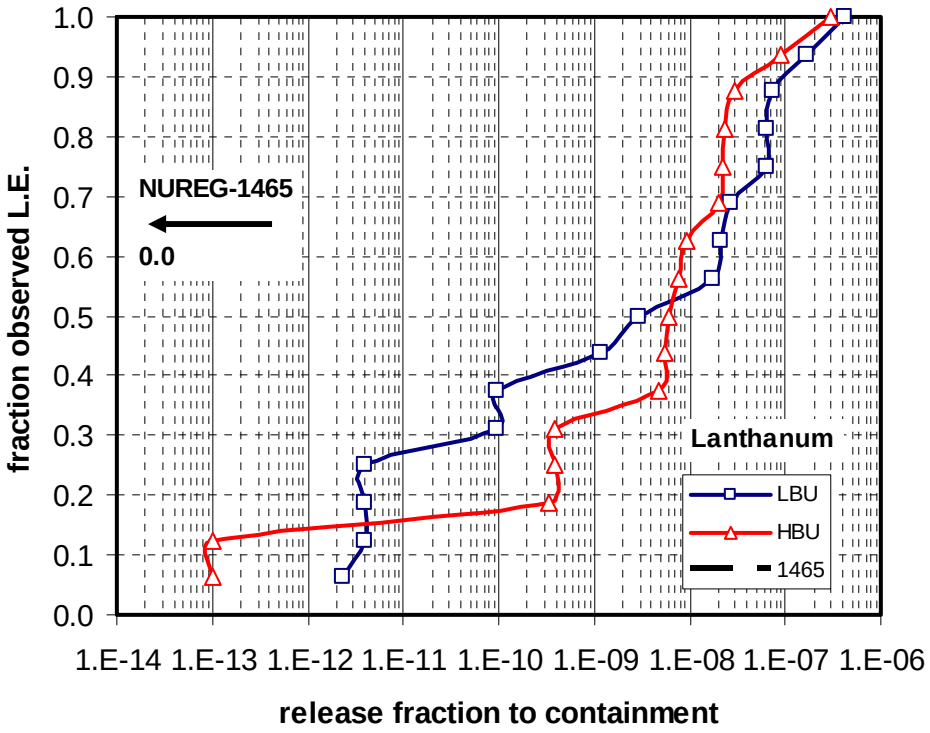

Figure 36 Late In-vessel Release Fraction for Lanthanum

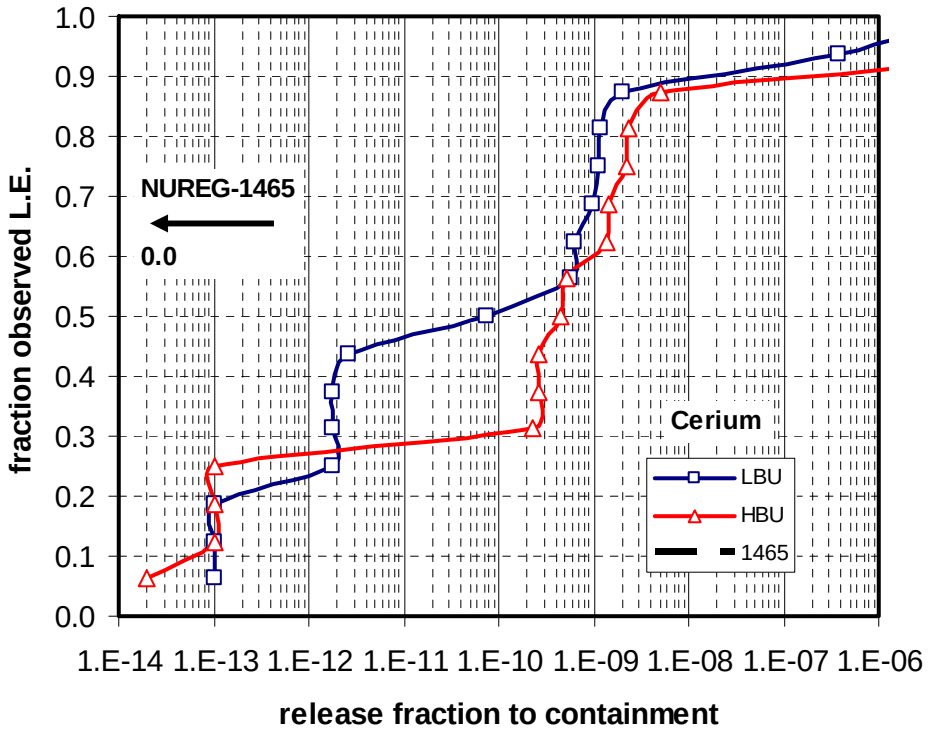

Figure 37 Late In-vessel Release Fraction for Cerium

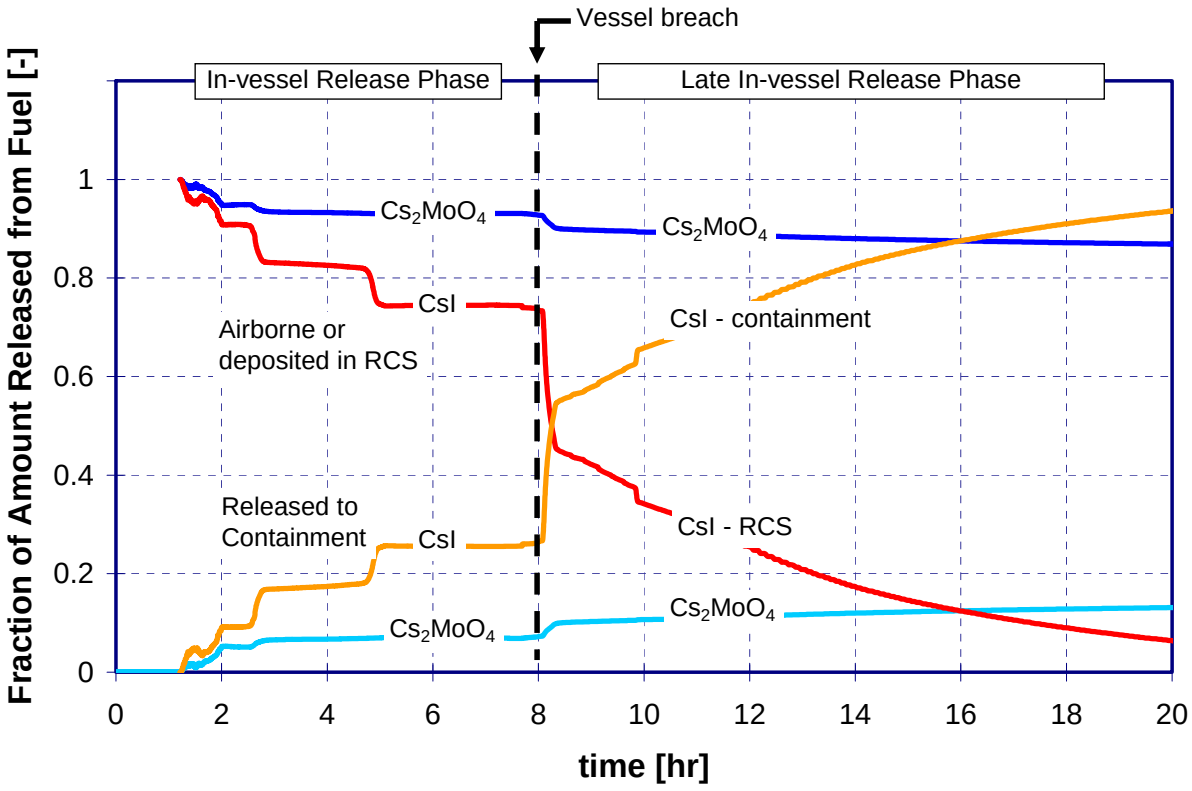

Figure 38 CsI versus Cs_2MoO_4 Release to Containment (from Case 1A)

7.0 Conclusions

This report describes the results of MELCOR calculations of fission product release to the containment for a wide spectrum of postulated core damage accident scenarios in a two types of BWRs. The calculations were performed to generate a technical basis for development of recommendations for updates to the NUREG-1465 Source Term for BWRs that will extend its applicability to accidents involving HBU cores (i.e., cores composed of fuel with an assembly average discharge burnup approaching the current regulatory limit of 59 MWd/kgU.) Although the primary objective of these calculations is to assess differences between radiological source terms from damaged BWR cores with HBU versus LBU characteristics, general insights regarding the calculated duration and magnitude of fission product release for a wide spectrum of postulate scenarios were also obtained. The latter aspect of this work offers an opportunity to examine the extent to contemporary MELCOR calculations of fission product release confirm or conflict with the NUREG-1465 prescription which is based on data and analyses that are outdated.

Results of the MELCOR calculations clearly indicate the effects of fuel discharge burnup on fission product source terms are well within the uncertainties inherent in typical source term calculations. A few examples of specific values for key metrics of the NUREG-1465 definition of source term are shown in Table 31. In each case, the average MELCOR-calculated values for the HBU analyses were within 10% of the parallel LBU calculations.

Table 31 Comparison of Average MELCOR Results for Key Metrics of the NUREG-1465 Source Term

Metric	LBU	HBU	Difference
Onset of Release from fuel (hr)	4.3	3.9	-8%
Duration of In-vessel Release Phase (hr)	9.1	8.1	-10%
Magnitude of In-vessel Release (fraction)			
Iodine	0.49	0.54	-9%
Cesium	0.20	0.19	-5%
Tellurium	0.43	0.42	+2%

Higher values of decay heat accompanying the high level of discharge burnup in the HBU analyses did result in a slight acceleration in the chronology of severe accident progression. This is evident in Table 31 by the decrease in the time at which fuel damage begins and the duration of the in-vessel damage period. However, these differences were generally found to be small in comparison to the range of values obtained over the spectrum of accident sequences studied. For example, the duration of the in-vessel release phase was found to span a range of 5.3 to 12.1 hours, depending on the scenario. Therefore, a one hour difference between the average HBU and LBU duration can be considered negligible. Similarly, the magnitude of the in-vessel release of iodine spanned a range from 16% to 86% of the initial core inventory. Therefore, the a difference of 5% between the average HBU and LBU release fraction can also be considered negligible.

In contrast to these findings, many aspects of the collective set of HBU and LBU results were found differ substantially from the NUREG-1465 values of duration and magnitude for several release periods. Examples include:

- The duration of the in-vessel release phase recommended in NUREG-1465 is 1.5 hours, whereas the median value from the MELCOR calculations (HBU and LBU) is closer to 7.5 hours. The minimum value obtained from the MELCOR calculations of the most rapid accident scenario (large LOCA with coincident loss of ac power) is 5.3 hours.
- The in-vessel release fractions for iodine, cesium and tellurium in NUREG-1465 are 0.25, 0.20 and 0.05, respectively. In contrast, the median MELCOR results were 0.52, 0.13 and 0.39. Again, these are median values; significantly larger releases were observed in half the calculations.
- Ex-vessel release (MCCI release from fuel) for all volatile species was found to greatly exaggerated in NUREG-1465 and, conversely, late in-vessel release fractions (due to revaporization from RCS surfaces) were found to be understated. In some cases, particularly for iodine and tellurium, these differences were quite large (e.g., typically, a factor of 10).

These differences can often be explained in terms of changes in the best-estimate characterization of in-vessel core damage progression or in the dominant chemical forms these radionuclides acquire when released from fuel. Specific examples are offered in the main body of this report.

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Appendix A. Standard MELCOR Modeling Practices, Modeling Parameters, and Sensitivity Coefficients for Analysis of Severe Accidents

A.1 BUR Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
BUR000	IACTV	1 (Not Active)	0 (Active)	Burn package activation
BUR1xx (xx = CV)	IGNTR	0.10	86 for CVs where ignition is to be prohibited.	Apply to RCS control volumes to preclude combustion.
	TFRAC	0.0	1.0	Time fraction of burn before propagation to neighboring CV is allowed. Value of 1.0 means a flame must travel the radius of the control volume before propagating to its neighbor.
Other Modeling Notes				
To insure that MELCOR properly estimates vertical burn propagation in containment, drywell, reactor building, and auxiliary building, it is necessary to define "vertical" flow path "from" and "to" elevations with a small dZ. If the "from" and "to" elevations are set equal (which has been historical practice to ensure complete vertical pool drainage), the MELCOR burn package uses criteria for horizontal burn propagation.				

A.2 CAV Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
CAVnnUB	BOILING	1	10	Multiplier for surface boiling coefficient
CAVnnUO	COND.OX	1	5	Multiplier for oxide conductivity
CAVnnUM	COND.MET	1	5	Multiplier for metal conductivity
CAVnnCa	TSOLCT TLIQCT TABLCT	BWR, PWR **, 1420K **, 1670K **, 1500K	Based on plant-specific concrete type.	
CAVnnRa	NOVC NCFRUP NCFREL	None	Overflow criteria defined via CFs	BWR, Mark I only: Overflow not allowed if T_{debris} is less than concrete solidus. Above $T_{solidus}$, overflow occurs when debris height exceeds temperature-dependent thresholds: Pedestal-to-90º-DW-sector: 6-in when $T_{debris} > T_{liquidus}$ 0.5-m, when $T_{debris} > T_{solidus}$ 90º-DW-sector to DW floor: 4-in when $T_{debris} > T_{liquidus}$ 0.5-m, when $T_{debris} > T_{solidus}$ Linearly interpolate at intermediate temperatures.
CAVnnSP	SOURCE	None	Spreading rate defined via CFs	BWR, Mark I only: "rate" defined in terms of transit time for debris to spread across region: Pedestal - instantaneous coverage 90º DW sector - linearly interpolate between: transit time=10 min if $T_{debris}>T_{liquidus}$ transit time=∞ if $T_{debris}<T_{solidus}$ DW floor -- linearly interpolate between: transit time=30 min if $T_{debris}>T_{liquidus}$ transit time=∞ if $T_{debris}<T_{solidus}$
	HTSIDE	Activated	Default	
CAVnnak	EMISS.OX EMISS.MET EMISS.SUR	0.6 0.6 0.6	0.9 0.9 0.9 Defaults for others	Emissivity of the oxide phase Emissivity of the metallic phase Emissivity of the surroundings

A.3 COR Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
COR00000	NPNTOT	None	1 per COR ring (BWR) 0 (PWR)	Number of penetrations
	NLH	None	10	Number of lower head nodes
COR00001	DRGAP	0.00012 (BWR) 0.00011 (PWR)	0.0	Thickness of gas gap between fuel pellets and cladding (0.0 to account for swelling)
COR00003	FCNCL	0.25	Default	Canister wall to fuel cladding
	FSSCN	0.25	Default	Control blades to fuel rods and debris
	FCELR	0.25	0.1	Cell to cell radial
	FCELA	0.25	0.1	Cell to cell axial
	FLPUP	0.25	Default	Liquid pool to core
COR00004	ICFFIS	0	CF for Chexal Layman	BWR ATWS only -- all others (SCRAM leads to termination of fission power.)
COR00005	HFRZUO	1000.0	7500.0	Candling HTC UO2
	HFRZZR		7500.0	Candling HTC Zircaloy
	HFRZSS		2500.0	Candling HTC steel
	HFRZZX		7500.0	Candling HTC ZrO_2
	HFRZSX		2500.0	Candling HTC steel-oxide
	HFRZCP		2500.0	Candling HTC control poison
COR00006	Specified defaults		Not in deck (defaults)	Model switches
COR00007	Specified defaults		Not in deck (defaults)	Candling secondary material transport parameters
COR00008	Specified defaults		Not in deck (defaults)	Component critical minimum thicknesses
COR00009	HDPBN	1000.0	0.0	Penetration model inactive. No heat transfer.
	HDBLH	1000.0	CF-Number Temp / HTC 2650. / 100. 2800. / 500. 3000. / 2000.	Specify HTC via control function as a function of debris temperature. Active only if mass of water in lower plenum < 500 kg. Otherwise HTC=1.0.
	TPFAIL	1273.15	3000.	Penetration model inactive.
	CDISPN	1.0	Default	Discharge coefficient debris from penetration
CORZjj03	FZPOW	1.0	Based on cycle-specific, plant-specific data.	Relative power density in axial level jj. Developed based on plant-specific data. In the absence of plant-specific data for PWRs, use legacy MELCOR input values (confirmed similar to available generic data).
CORRii03	FRPOW	1.0	Based on cycle-specific, plant-specific data.	Relative power density in radial ring ii. Developed based on plant-specific data.
CORijj04	DHYPD	None	Core - 0.01 LP - 0.002	Particulate debris equivalent diameter (LP values for DHYPD, HDBH2O, VFALL tuned to get appropriate end-of-pour debris temperature. 2mm based on FAERO fragmented debris size).

COR Package Modeling Parameters (continued)

Record	Field	MELCOR Default	Calculation Values	Description
CORijjFCL	ICFAI	None	CFV	CFV is control function that specified failure criteria (for collapse of fuel rods) in cell ijj. Once unoxidized cladding thickness drops below 0.1mm, the following damage function is tracked: clad temp (K) — time to failure (min) ≤ 1000. — ∞ 1001. — 60. 2100. — 30. ≥ 2500. — 1.
CORZjj01	PORDP	None	0.4	Porosity of particulate debris
COR00012	HDBH2O	100.0	2000.	HTC in-vessel falling debris to pool (W/m2-K)
COR00012	VFALL	1.0	0.01	Velocity of falling debris (m/s)
CORZjjSS	ISSMOD	PLATEG	BWR PLATEB COLUMN PWR PLATEG COLUMN PLATEG COLUMN PLATE	Core axial level ( ) (6) Core support plates (1-5) Control rod guide tubes (7) Core support plate (6) Vertical structure below support plate (5) Diffuser plate (4) Vertical structure below diffuser plate (3) Lower support plate
		ISSFAI	BWR PLATEB COLUMN PWR PLATEG COLUMN PLATEG COLUMN PLATE	Core axial level ( ) (6) COR Package stress model (1-5) COR Package stress model (7) COR Package stress model (6) CF: Remaining life < 0.01 (5) COR Package stress model (4) CF: Remaining life < 0.01 (3) COR Package stress model
CORZjjNS	TNSMAX	0.0	1520. 1700.	BWR only: control blades failure temperature core top guide failure temperature

Other Modeling Notes

1. BWR nodalization: Lower tie plate and fuel support piece steel are blended with core support plate mass, and extend top of lower plenum COR axial level to bottom of active fuel.
2. BWR nodalization: Use single lower plenum core cell large enough to hold all molten core debris. This insures that core debris won't artificially be isolated from overlying water pool.
3. PWR nodalization: Assembly lower nozzle steel is blended with core support plate mass.

A.4 COR Package Sensitivity Coefficients

Record	Field	MELCOR Default	Recommended Value	Description
SC-1001 (1,1)		29.6	Default	Zircaloy steam oxidation rate parameters
SC-1001 (2,1)		16820.0		
SC-1001 (3,1)		87.9		
SC-1001 (4,1)		16610.0		
SC-1001 (5,1)		1853.0		
SC-1001 (6,1)		1873.0		
SC-1001 (1,2)		50.4	27.883	Zircaloy air oxidation rate. High temperature values based on recommendation of Dana Powers. See NUREG/CR-6218 and NUREG/CR-0649. Low temperature values from recent ANL tests.
SC-1001 (2,2)		14630.0	15630.	
SC-1001 (3,2)		0.0	50.4	
SC-1001 (4,2)		0.0	14630.	
SC-1001 (5,2)		10000.0	1333.	
SC-1001 (6,2)		10000.0	1550.	
SC-1131 (2)		2400.0	Default	ZrO_2/Zr melt release temperature
SC-1131 (6)		2100.0	Default	ZrO2/Zr melt release and collapse temperature for BWR canisters
SC-1132 (1)		2500.0	2800.	Fuel rod collapse temperature
SC-1030 (2)		0.1	Default	COR dT/dz flow time constant
SC-1250 (1)		3200.0	2800.	Temperature constant for component conduction enhancement at melting temp.
SC-1250 (2)		0.01	Default	Leading scalar for component conduction enhancement at melting temperature
SC-1505 (1)		0.001	0.05	Minimum porosity for flow resistance
SC-1505 (2)		0.001	0.05	Minimum porosity for calculating area for heat transfer to the fluid
SC-1600 (1)		0.0	1.0	Zero-dimensional (0.0) or one-dimensional (1.0) stress/strain distribution in lower head
SC-1603 (2)		1800.0	1700.0	Temperature at which lower head yield stress vanishes [to force failure of LH prior to melting when dP is at/near zero].

A.5 CVH/FL Package Sensitivity Coefficients

Record	Field	MELCOR Default	Calculation Values	Description
SC-4401(3)		0.0	15	Default = # of flow paths in problem (not recommended) Limit maximum number of iterations permitted before subcycle.
SC-4413 (5)		0.001	0.05	Minimum porosity in the Ergun correlation
SC-4414		0.0001	0.01	Minimum hydrodynamic volume fraction
SC-4415		0.0	1.0	Fast iterative flow solver

A.6 DCH Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
"Best-practice" for decay heat data is to use plant-specific data from ORIGEN calculations. Default input acceptable only when ORIGEN data not available.				
DCHOPERPOW	OPRPOW	None	Plant-specific full-power steady state thermal operating power	Reactor operating power before shutdown (required record for code versions after 1.8RL)
DCHSHUT	TNSHUT	0.0	BWR CF: Time at which fission power < 2% PWR Time is keyed off reactor trip control function (CF100).	Accommodates ATWS sequences.
DCHNEMnn00	ELMNAM ELMMAS	None	Based on ORIGEN results for core.	Elemental fission product mass at shutdown for calculation of decay heat.
DCHNEMnnmm	TIME DCHEAT	None	Based on ORIGEN results for modeled core.	Elemental fission product decay heat per unit mass (based on shutdown RN inventory.) Data pairs are Time, decay heat/kg (with t=0 being shutdown). • Define specific decay heat for CsI (Class 16) as 0.51155 of value for Class 2 (Cs) plus 0.48845 of value for Class 4 (I). • Define specific decay heat for Cs_2MoO_4 (Class 17) as 0.7348 of value for Class 2 (Cs) plus 0.2652 of value for Class 7 (Mo).
DCHCLSnnn0	RDCNAM	None	Based on ORIGEN results for modeled core.	Synthesize ORIGEN data to define a single representative element for each class with decay heat data that reflects decay heat for all elements within the class (DCHNEMxxxx input.) Redefine each class to include only the representative element.
DCHCLSnnnm	CLSELM	None	Based on ORIGEN results for modeled core.	
DCHDEFCLS0	DEFCLS	None	13, 14, 15	Specifies that MELCOR DCH default classes are to be used.
DCHCLNORM	CLSNRM	YES	NO	New ORIGEN input for elements/classes defines the total core decay heat.

A.7 FDI Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
----				Model active with transfer from COR to FDI to CAV. Use bottom of lower head as interaction elevation.

A.8 HS Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
HScccccc400 & HScccccc600	CPFPL CPFAL	None CPFPL	0.9 0.9	Minimum value of CVH pool fraction such that heat transfer is calculated to Pool/Amosphere for heat structures within the RPV. All other structures modeled with 0.5/0.5. This value is important for upper plenum HS's and structures identified as COR radial boundary HS's in the COR package input.
HScccccc401 HScccccc601	EMISWL RMODL PATHL	Radiation disabled as default	0.27 EQUIV-BAND 0.1	Mean emissivity of SS type 316 [Siegel& Howell] Equivalent band radiation model. Nominal optical distance in steam (m). These settings are applied ONLY to heat structures within the reactor vessel and to PWR RCS heat structures being monitored for creep-rupture failure (sg tubes, hot leg nozzle, surge line).
HSDGccccc0	ISRCHS ISDIST GASNAM	None None None	core shroud HS # 1. ss	Heat structure for application of degas model. Number of mesh intervals for application. Name of released gas.
HSDGccccc1	RHOSRC HTRSRC TEMPL TEMPU	None None None None	7930. 2168.E+05 1695. 1705.	Gas source density. Gas source heat of reaction. Lower temperature for degassing. Upper temperature for degassing.
Other Modeling Notes				
Make sure miscellaneous heat structures in containment, drywell, reactor building are modeled with appropriate horizontal area (for aerosol settling) and mass (for thermal sink).				

A.9 HS Package Sensitivity Coefficients

Record	Field	MELCOR Default	Calculation Values	Description
SC-4055(2)		5.e-4	0.5	This is the HS temperature convergence criterion. MELCOR periodically fails on HS temperature convergence in a single timestep. Calculations have been performed with this criterion set at default and at 0.5. No differences in calculated results have been noticed.

A.10 MP Package Data

Record	Field	MELCOR Default	Calculation Values	Description
MPMATxxxx	ENH TMP MLT	UO_2: Properties based on T_{melt}=3113K ZrO: Properties based on T_{melt}=2990K	UO_2: Properties based on T_{melt}=2800K ZrO: Properties based on T_{melt}=2800K	Adjustments in UO2 / ZrO enthalpy to represent the effects of eutectic interactions.

A.11 RN Package Modeling Parameters

Record	Field	MELCOR Default	Calculation Values	Description
RNFP000	ICRLSE	-2	-3	Use ORNL-Booth coefficients and other parameters developed from Phebus/VERCORS
RN1002	IHYGRO	0 (Not Active)	1 (Active)	Hygroscopic model activation
RNCA100	ICAON	1 (On)	Default	Chemisorption model activation
RNFPNijjXX	NINP RINP1 RINP2	None	RN Class mass in Cell ijj 1.0	Tells RN to use fission product masses defined in DCH input. Distributes mass based on distribution developed with ORIGEN.

Other Modeling Notes

Additional Guidelines for implementing the RN speciation recommended by Gauntt:

Initial mass distribution of Cs, I, and Mo:

- Place 5% of the noble gas inventory in the fuel gap.
- Stoiciometrically combine all I with Cs and place in Class 16 as CsI.
- Place 5% of CsI in the fuel gap. This represents 5% of the Iodine inventory, but a much smaller fraction of the Cs inventory.
- Determine the quantity of Cs required in addition to that represented by CsI in the gap (above) to reach a total or 5% of the core inventory. Place this additional mass in Class 2 and position the entire Class 2 inventory in the fuel gap.
- The quantity of remaining Cs (95% of the core inventory) should be of a sufficient quantity to completely react with a fraction of the core inventory of Mo to form Cs_2MoO_4. Place this mass and the stoichiometric fraction of Mo inventory in Class 17.
- Place all remaining (excess) Mo in Class 7.

Physical properties of RN Classes 2, 4, 7, 16 and 17

- Use ORNL-Booth coefficients, scaling factors and vapor pressures recommended by Gauntt, with the following clarifications, or exceptions:
  - Class 2 (CsOH): Apply vapor pressure data for Cs_2MoO_4. Release rates for all other classes are referenced to the release rate for Class 2. Scaling factors developed by Gauntt were based on Class 2 having release rate properties of Cs_2MoO_4. However, apply the default value of molecular weight, which applies to CsOH.
  - Class 7 (Mo metal): Use default values of all physical properties for this class (i.e., properties recommended by Gauntt for class 7 are not to be used.
  - Class 16 (CsI): Apply a non-default Cs release rate multiplier of 0.64 to. This anchors the release rate of CsI to the effective release rate of I (Class 4) in Gauntt's work.
  - Class 17 (Cs_2MoO_4): Use a Cs release rate multiplier of 1.0. Apply molecular weight, solubility, density and vapor pressure data for Cs_2MoO_4.
    - SC7120(1,17) = 361.75 MW [Cs_2 / Mo]
    - SC7120(2,17) = 425.75 MW [Cs_2MoO_4]
    - SC7170(9,17) = 4030 kg/m^3 rho [Cs_2MoO_4]
    - SC7170(3,17) = 0.67 solubility [Cs_2MoO_4]
    - SC7170(4,17) = 0.67

CORSOR-VANESA cross reference

- Class 2 (CsOH) and 17 (Cs_2MoO_4) mapped to VANESA as Cs. All Cs transferred out of VANESA should be mapped to RN Class 17 (Cs_2MoO_4.) All other class transfers can be treated with default scheme.
- Class 16 (CsI) mapped to VANESA CsI and return.

Guidelines for modeling release of non-radioactive, structural aerosol

- For PWRs, invoke the Ag-In-Cd release model in RN.
- For BWRs, apply the non-fuel release model (RNCRCLxx records). Assign aerosol generated from Zr and ZrO_2 to RN Class 12 (Sn). The mass will be added as a non-radioactive mass to this class. The fraction of material mass available for release as an aerosol from these materials is 0.0145 (Sn fraction in Zirc-2 and -4.) Apply the following release rate factors: Unoxided-Zr: 0.1, ZrO_2: 1.0. The multiplier for fuel should remain at the default value (1.0). Factors for all other materials should be set to 0.0.

Appendix B. Description of MELCOR Models Used

B.1 Peach Bottom MELCOR Model

The general hydrodynamic nodalization scheme used for the Peach Bottom MELCOR model is shown in Figures B-1 through B-4. Additional information on the model of the reactor core is given in Section 2.3. Modeling options selected for the MELCOR calculations represent current best-practice and are listed in Appendix A.

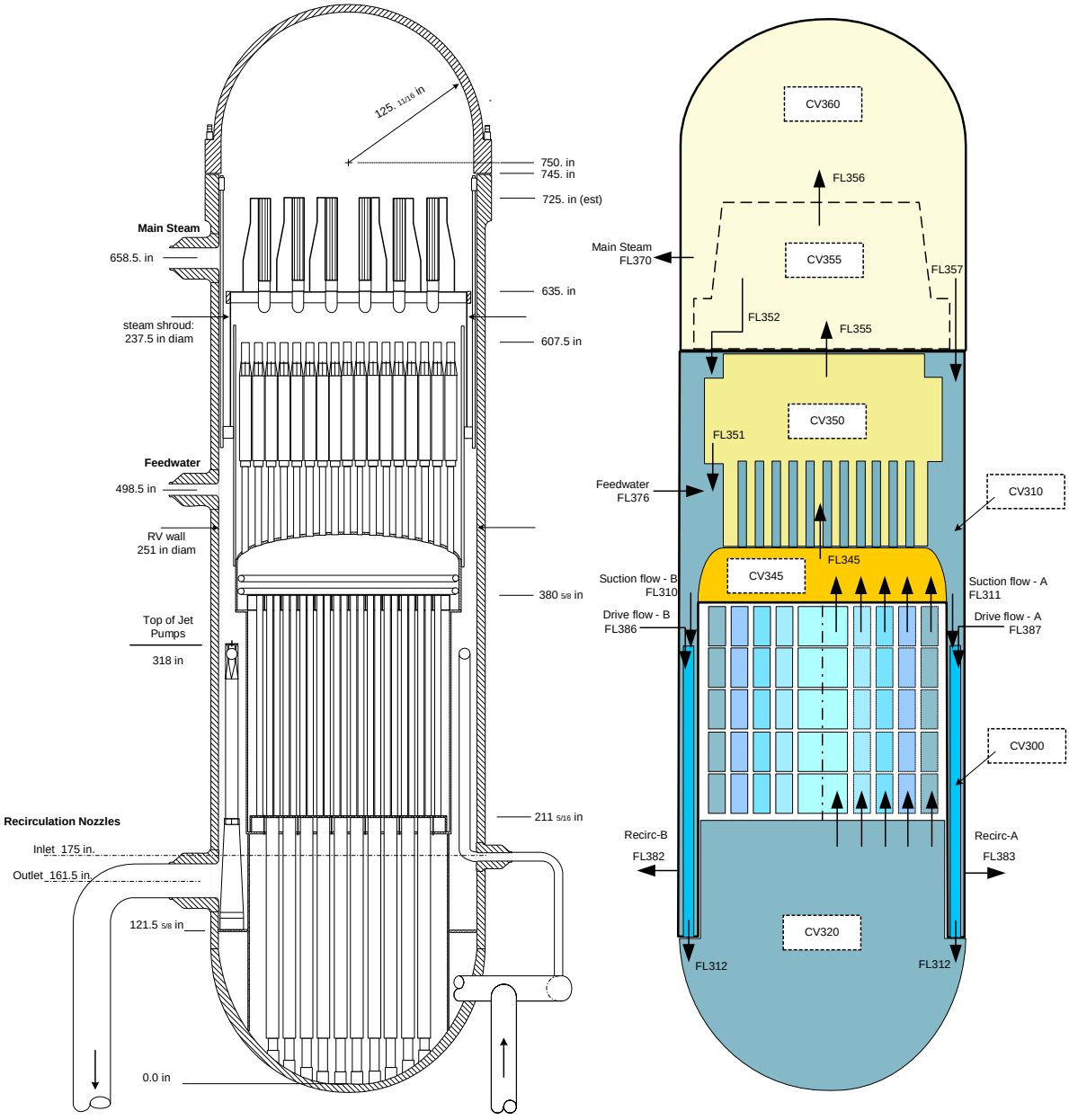


Figure B.1 Reactor Vessel MELCOR CVH Nodalization Used in the Peach Bottom Model

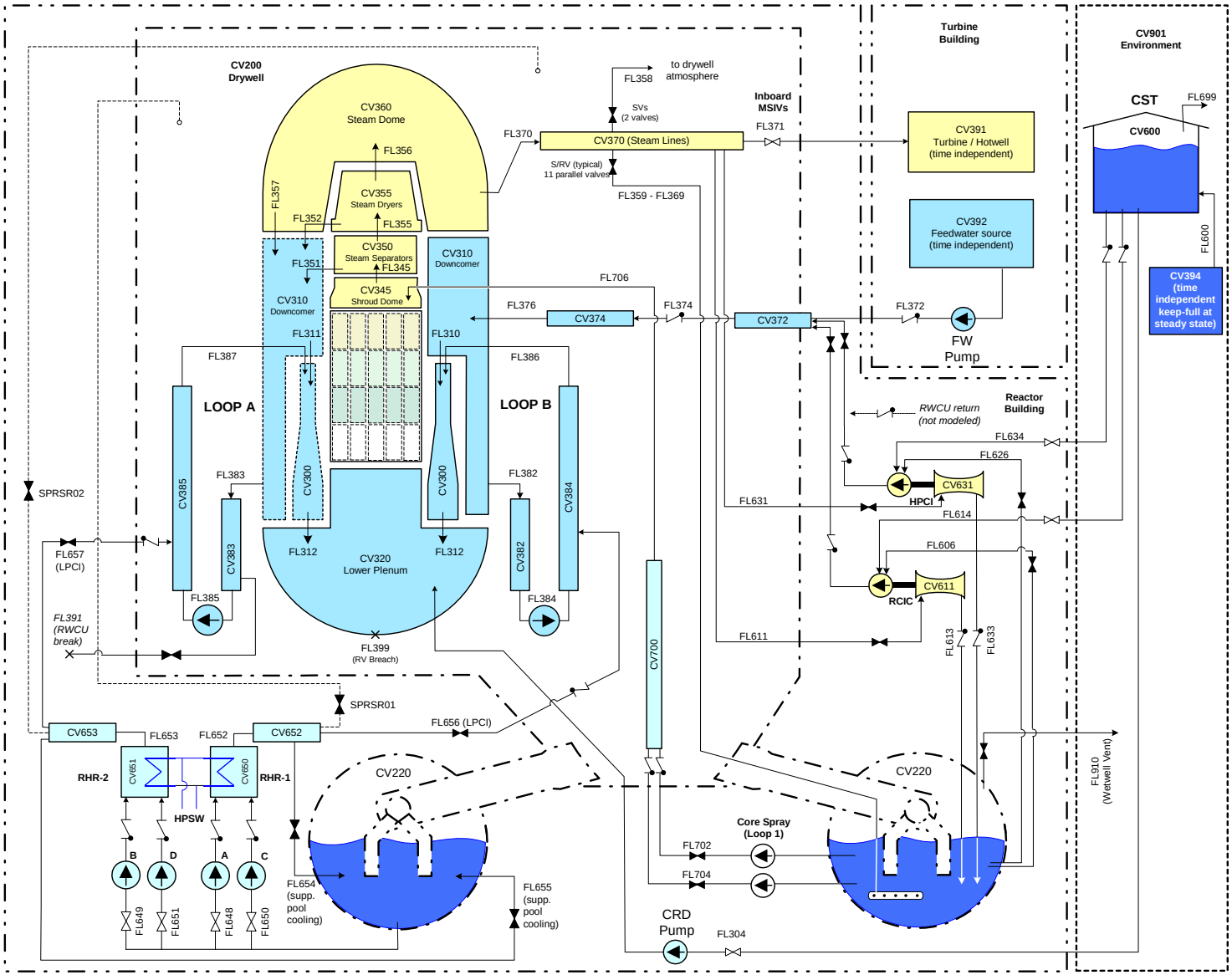


Figure B.2 Reactor Coolant Systems MELCOR CVH Nodalization Used in the Peach Bottom Model

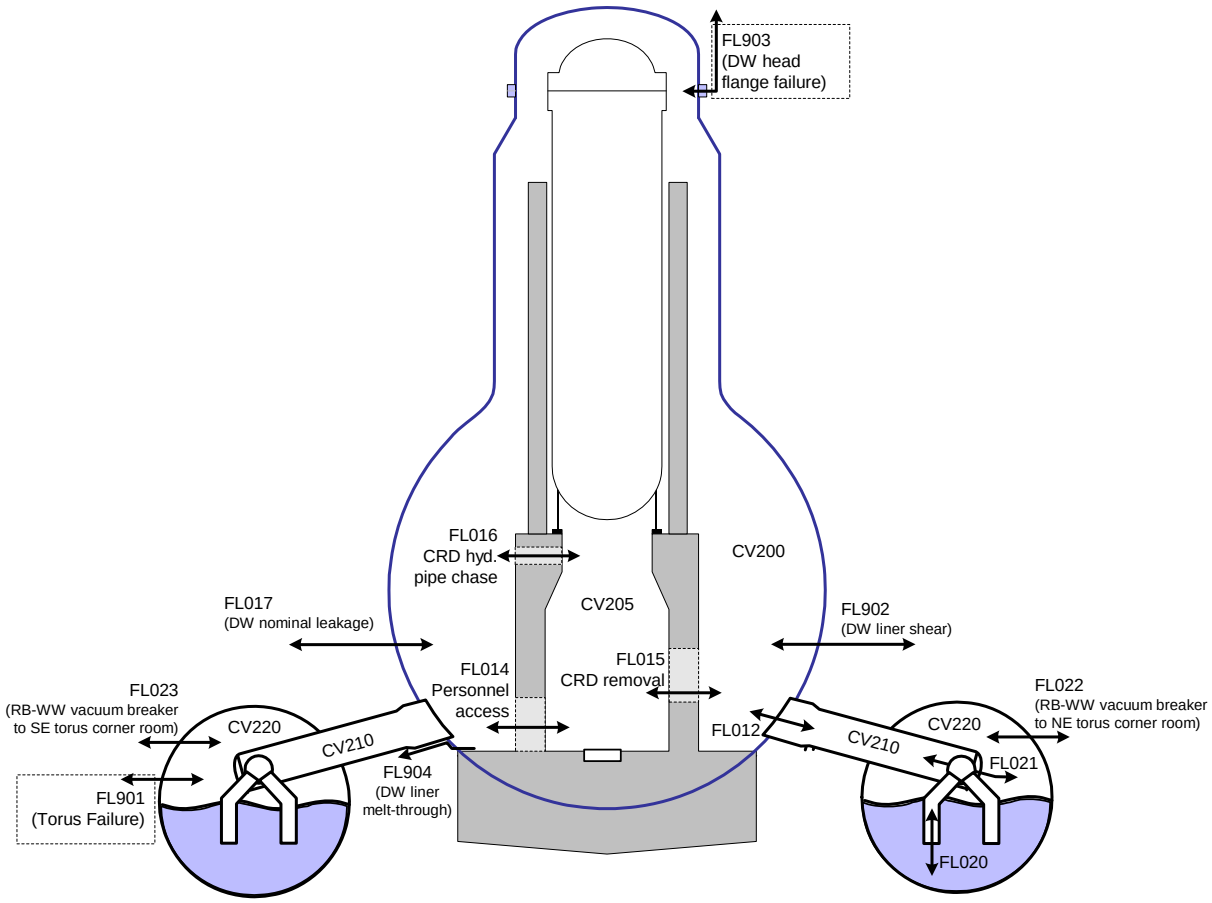


Figure B.3 Nodalization of BWR Mark I Containment Used in the Peach Bottom MELCOR Model

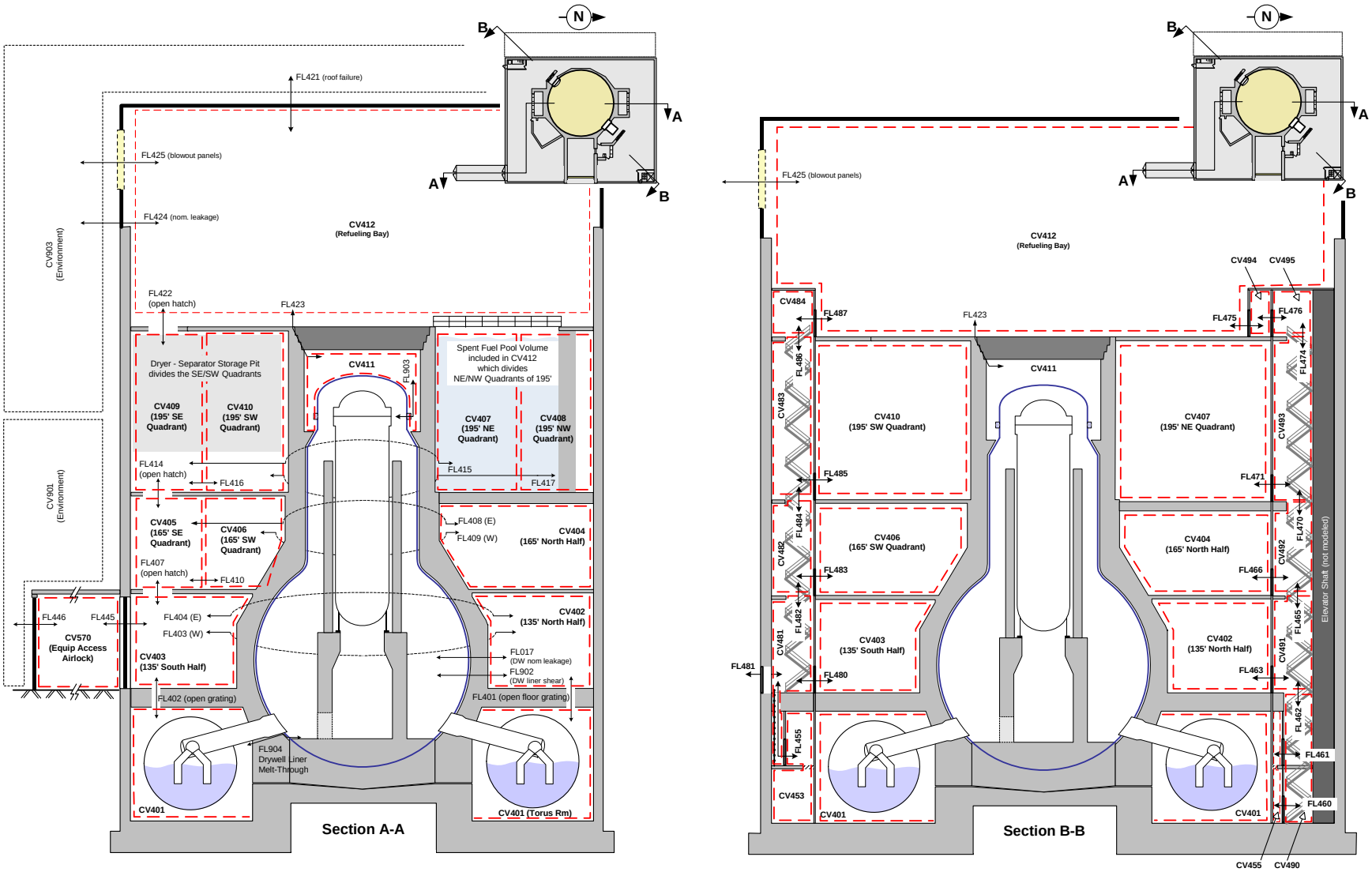


Figure B.4 Nodalization of BWR Mark I Reactor Building Used in the Peach Bottom MELCOR Model

B.2 Grand Gulf MELCOR Model

The hydrodynamic nodalization scheme of the Grand Gulf reactor vessel is conceptually the same as the one used for Peach Bottom; the models are quantitatively different, however, due to differences in reactor scale and core (fuel) design. The containment nodalization scheme for Grand Gulf differs substantially from Peach Bottom, as shown in Figures B-5 through B-8. Additional information on the model of the reactor core is given in Section 2.3. Modeling options selected for the MELCOR calculations represent current best-practice and are listed in Appendix A.

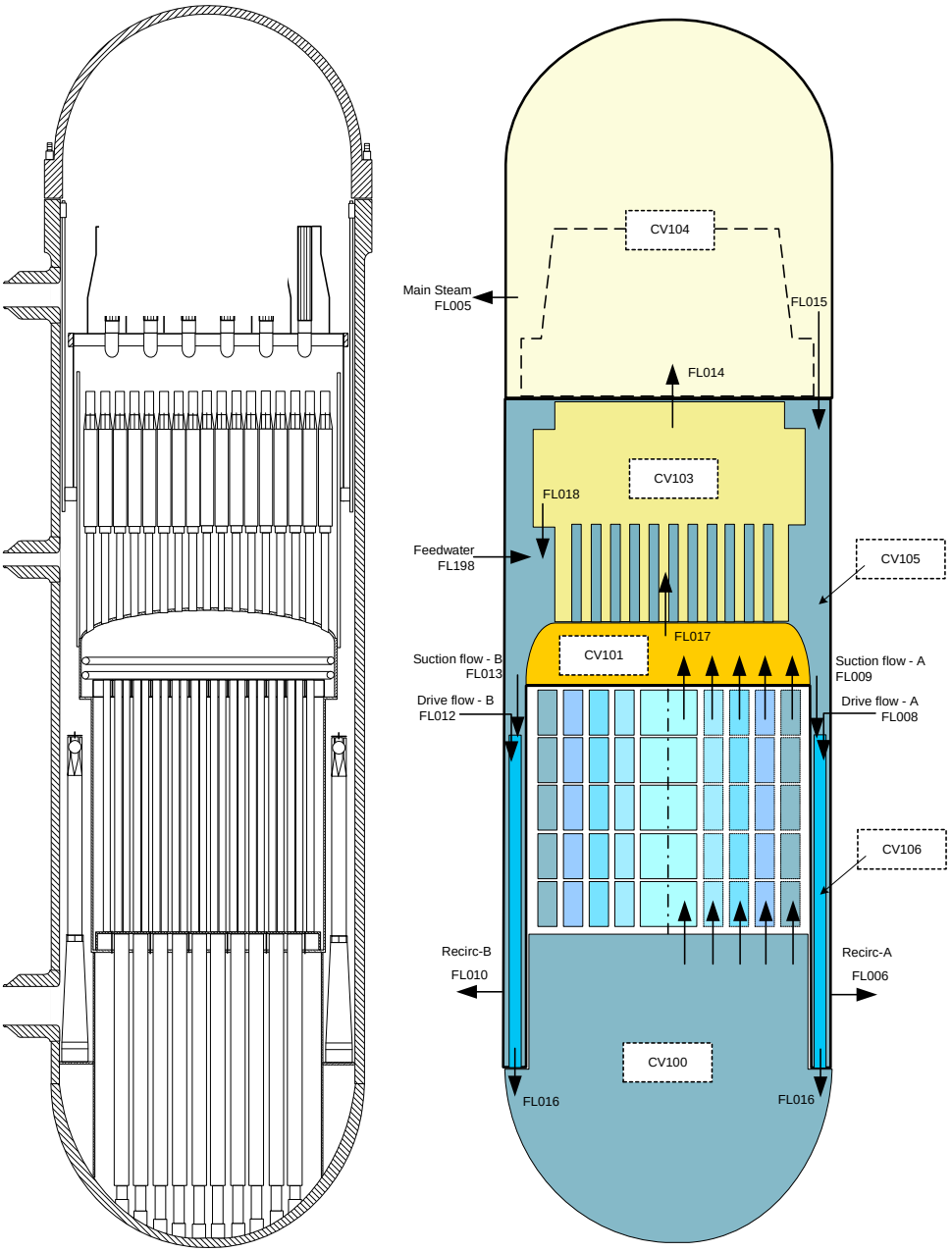


Figure B.5 Reactor Vessel MELCOR CVH Nodalization Used in the Grand Gulf Model

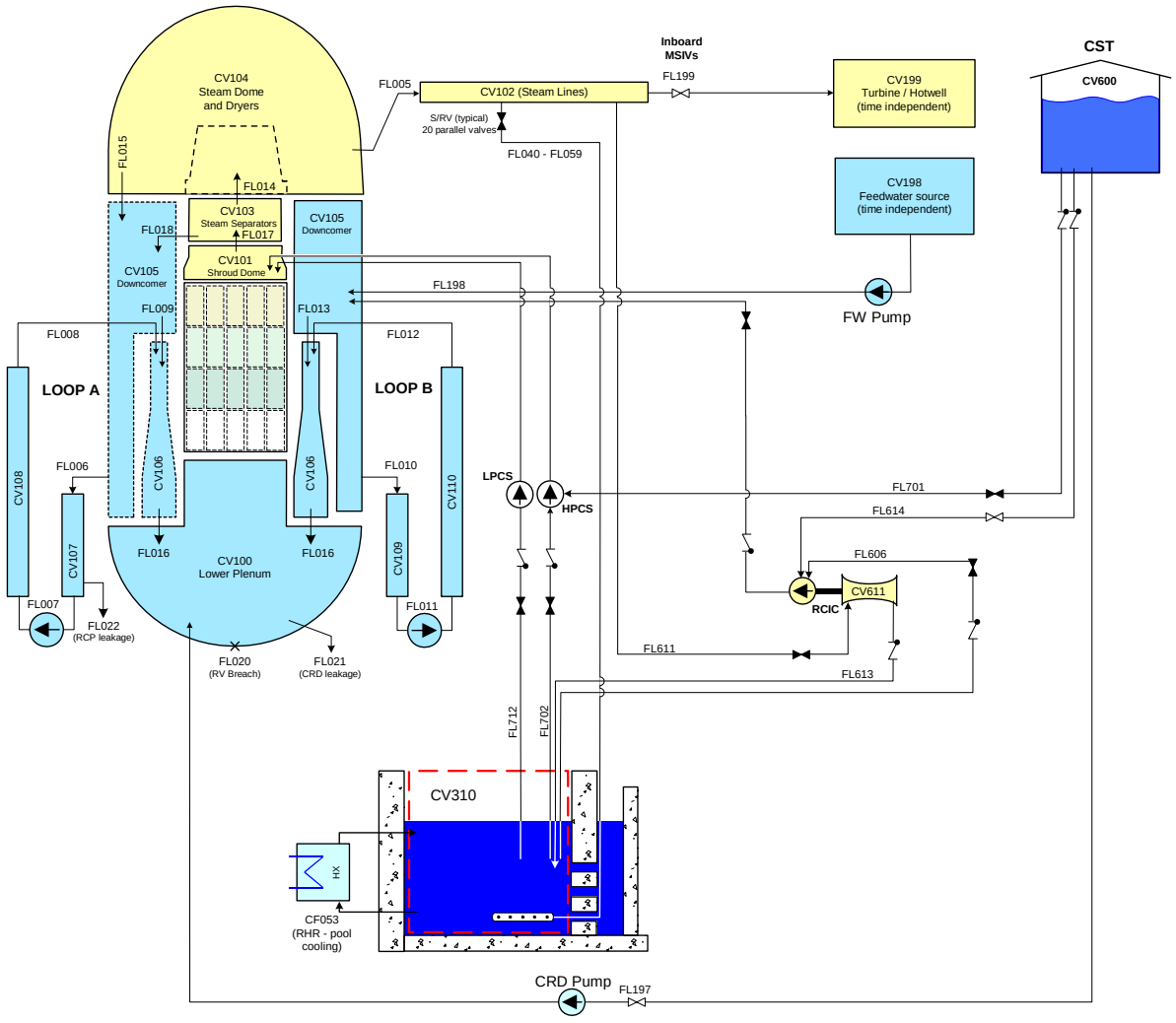


Figure B.6 Reactor Coolant System Nodalization Used in the Grand Gulf Model

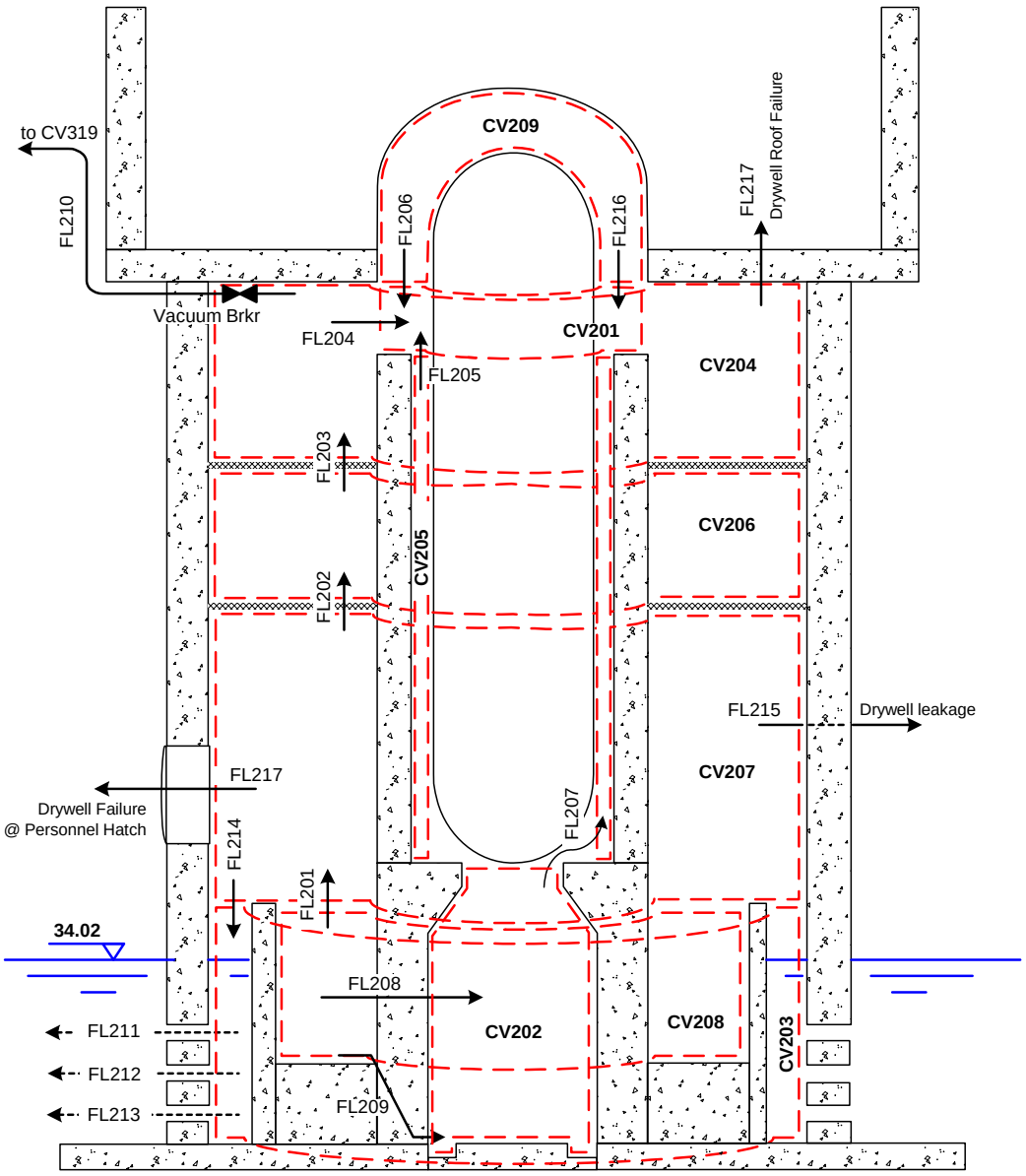


Figure B.6 Mark III Drywell Nodalization Used in the Grand Gulf Model

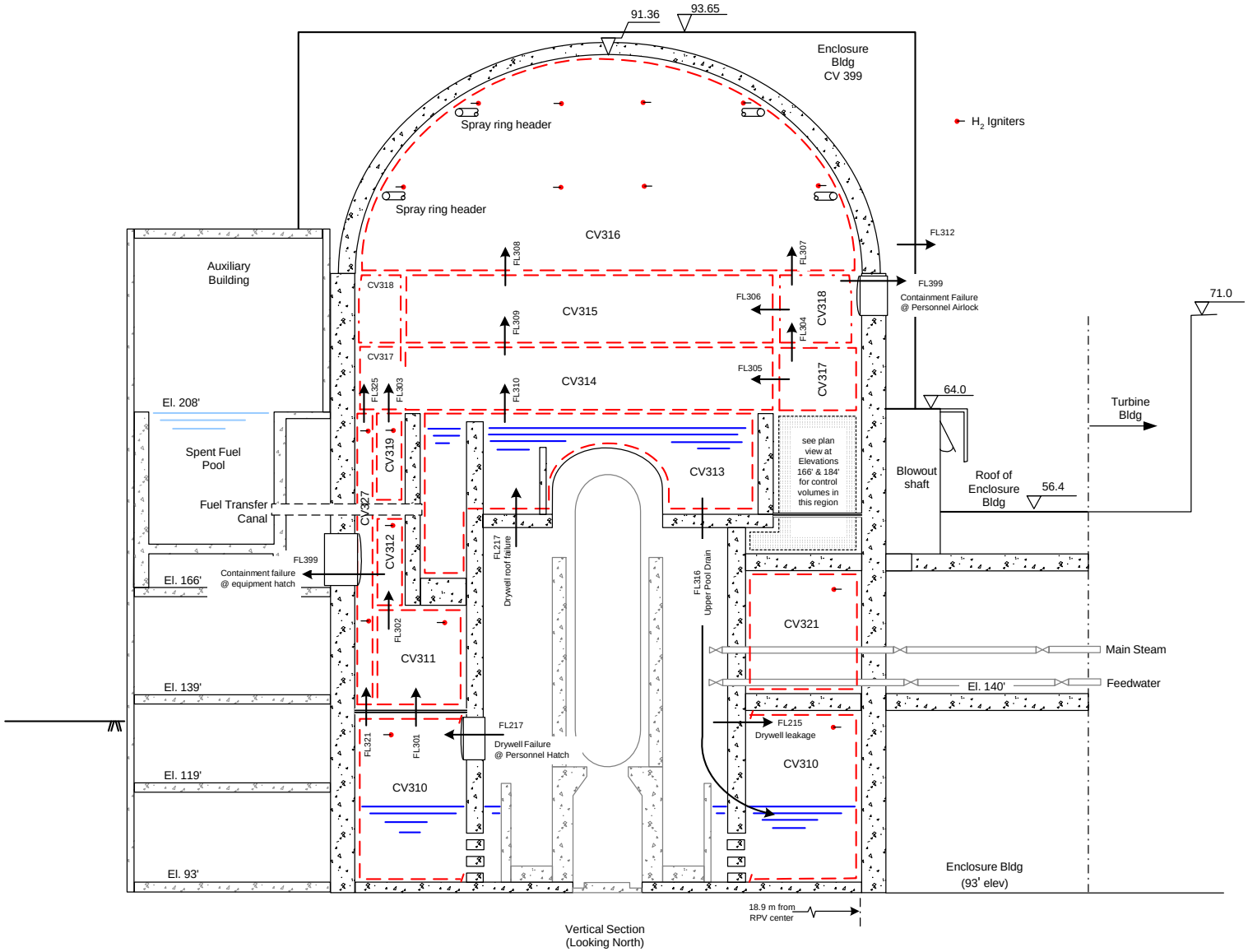


Figure B.8 Mark III Containment MELCOR CVH Nodalization Used in the Grand Gulf Model

Appendix C. Key Event Timing Tables for MELCOR Accident Sequences

C.1 Peach Bottom, Case 1A

Short-term Station Blackout. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure initially occurs as a result of drywell shell melt-through.

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU CF@VB**	**STSBO 10x10 HBU CF@VB**
Station blackout [battery failure]	0 [0]	0 [0]
Downcomer water level reaches TAF	0.65	0.65
First hydrogen production	1.19	1.17
First fuel-cladding gap release	1.23	1.20
First channel box failure	1.65	1.48
First core cell collapse due to time at temperature [cell]	1.86 [112]	1.70 [212]
Reactor vessel water level reaches bottom of lower core plate	2.55	2.00
First core support plate localized failure in supporting debris	3.05	2.54
SRV sticks open	3.17	2.66
RPV pressure first drops below LPI setpoint (450 psig)	3.53	3.04
Lower head dries out	4.14	3.66
Ring 1 CRGT Column Collapse	6.77	4.86
Ring 2 CRGT Column Collapse	6.67	5.23
Ring 3 CRGT Column Collapse	6.57	5.76
Ring 4 CRGT Column Collapse	6.50	5.72
Ring 5 CRGT Column Collapse	6.91	6.15
Lower head failure	9.22	7.69
Drywell liner melt-through	9.45	8.08
Refueling bay to environment blowout panels open	9.45	8.09
First hydrogen deflagration in reactor building	9.46	8.09
Equipment access door opens to environment due to overpressure	9.46	8.09
Reactor building to turbine building blowout panel opens	9.46	8.09
Time Iodine release to environment exceeds 1%	9.51	8.16
Door to environment in southwest stairwell opens due to overpressure	9.60	8.23
Refueling bay roof overpressure failure	9.60	8.23
Drywell liner penetration shear failure	10.98	9.85
Drywell head flange leaks	13.13	11.33
Calculation terminated (CAV rupture)	32.4	20.5

C.4 Peach Bottom, Case 1B

Short-term Station Blackout. All SRV operate as designed in safety relief mode, therefore, reactor vessel breach occurs at high pressure. Containment failure initially occurs as a result of drywell shell melt-through.

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU High P. CF@VB**	**STSBO 10x10 HBU High P. CF@VB**
Station blackout [battery failure]	0 [0]	0 [0]
Downcomer water level reaches TAF	0.65	0.65
First hydrogen production	1.19	1.17
First fuel-cladding gap release	1.23	1.20
First channel box failure	1.65	1.48
First core cell collapse due to time at temperature [cell]	1.86 [112]	1.70 [212]
Reactor vessel water level reaches bottom of lower core plate	2.55	2.00
First core support plate localized failure in supporting debris	3.05	2.54
SRV sticks open	N/A	N/A
RPV pressure first drops below LPI setpoint (450 psig)	8.25	7.28
Lower head dries out	3.95	3.58
Ring 1 CRGT Column Collapse	8.96	7.23
Ring 2 CRGT Column Collapse	8.91	7.42
Ring 3 CRGT Column Collapse	9.01	7.50
Ring 4 CRGT Column Collapse	8.45	7.48
Ring 5 CRGT Column Collapse	8.88	7.81
Lower head failure	8.21	7.25
Drywell liner melt-through	9.59	8.09
Refueling bay to environment blowout panels open	9.59	8.09
First hydrogen deflagration in reactor building	9.60	8.09
Equipment access door opens to environment due to overpressure	9.72	8.09
Reactor building to turbine building blowout panel opens	9.71	8.09
Door to environment in southwest stairwell opens due to overpressure	9.72	8.13
Time Iodine release to environment exceeds 1%	9.73	8.13
Drywell liner penetration shear failure	11.48	9.86
Drywell head flange leaks	14.19	11.20
Calculation terminated (CAV rupture)	23.5	20.2

C.2 Peach Bottom, Case 1C

Short-term Station Blackout. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure initially occurs as a result of drywell shell melt-through. This case differs from Case 1A only in the type of concrete represented on the drywell floor (Basaltic).

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU Basaltic CF@VB**	**STSBO 10x10 HBU Basaltic CF@VB**
Station blackout [battery failure]	0 [0]	0 [0]
Downcomer water level reaches TAF	0.65	0.65
First hydrogen production	1.19	1.17
First fuel-cladding gap release	1.23	1.20
First channel box failure	1.65	1.48
First core cell collapse due to time at temperature [cell]	1.86 [112]	1.70 [212]
Reactor vessel water level reaches bottom of lower core plate	2.55	2.00
First core support plate localized failure in supporting debris	3.05	2.54
SRV sticks open	3.17	2.66
RPV pressure first drops below LPI setpoint (450 psig)	3.53	3.04
Lower head dries out	4.14	3.66
Ring 1 CRGT Column Collapse	6.77	4.86
Ring 2 CRGT Column Collapse	6.67	5.23
Ring 3 CRGT Column Collapse	6.57	5.76
Ring 4 CRGT Column Collapse	6.50	5.72
Ring 5 CRGT Column Collapse	6.91	6.15
Lower head failure	9.22	7.69
Drywell liner melt-through	9.45	8.03
Refueling bay to environment blowout panels open	9.45	8.03
First hydrogen deflagration in reactor building	9.46	8.04
Equipment access door opens to environment due to overpressure	9.46	8.04
Reactor building to turbine building blowout panel opens	9.46	8.04
Time Iodine release to environment exceeds 1%	9.51	8.11
Door to environment in southwest stairwell opens due to overpressure	9.60	8.18
Drywell liner penetration shear failure	11.40	10.76
Drywell head flange leaks	13.11	12.42
Calculation terminated (CAV rupture)	14.7	12.9

C.3 Peach Bottom, Case 1D

Short-term Station Blackout. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure occurs as a result of drywell head flange leakage.

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU CF Late**	**STSBO 10x10 HBU CF Late**
Station blackout [battery failure]	0 [0]	0 [0]
Downcomer water level reaches TAF	0.65	0.65
First hydrogen production	1.19	1.17
First fuel-cladding gap release	1.23	1.20
First channel box failure	1.65	1.48
First core cell collapse due to time at temperature [cell]	1.86 [112]	1.70 [212]
Reactor vessel water level reaches bottom of lower core plate	2.55	2.00
First core support plate localized failure in supporting debris	3.05	2.54
SRV sticks open	3.17	2.66
RPV pressure first drops below LPI setpoint (450 psig)	3.53	3.04
Lower head dries out	4.14	3.66
Ring 1 CRGT Column Collapse	6.77	4.86
Ring 2 CRGT Column Collapse	6.67	5.23
Ring 3 CRGT Column Collapse	6.57	5.76
Ring 4 CRGT Column Collapse	6.50	5.72
Ring 5 CRGT Column Collapse	6.91	6.15
Lower head failure	9.22	7.69
Drywell head flange leaks	10.47	9.63
Refueling bay to environment blowout panels open	10.47	9.63
First hydrogen deflagration in reactor building	10.47	9.63
Drywell liner penetration shear failure	10.48	9.98
Equipment access door opens to environment due to overpressure	10.48	9.98
Reactor building to turbine building blowout panel opens	10.48	9.98
Time Iodine release to environment exceeds 1%	10.49	9.69
Calculation terminated (CAV rupture)	30.8	19.4

C.5 Peach Bottom, Case 2A

Long-term Station Blackout. RCIC operates for 8 hrs then terminates due to battery exhaustion. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure initially occurs as a result of drywell shell melt-through.

Event (Time in hours unless noted otherwise)	**LTSBO 8x8 LBU CF@VB**	**LTSBO 10x10 HBU CF@VB**
Station blackout [battery failure]	0 [8]	0 [8]
Downcomer water level reaches TAF	11.50	10.69
First hydrogen production	12.73	11.78
First fuel-cladding gap release	12.83	11.85
First channel box failure	13.58	12.35
First core cell collapse due to time at temperature [cell]	13.84 [111]	12.64 [211]
Reactor vessel water level reaches bottom of lower core plate	14.05	13.00
First core support plate localized failure in supporting debris	15.45	14.04
SRV sticks open	15.71	14.13
RPV pressure first drops below LPI setpoint (450 psig)	16.01	14.49
Lower head dries out	16.69	15.68
Ring 1 CRGT Column Collapse	22.68	19.26
Ring 2 CRGT Column Collapse	22.65	19.48
Ring 3 CRGT Column Collapse	22.07	19.35
Ring 4 CRGT Column Collapse	21.61	18.98
Ring 5 CRGT Column Collapse	21.88	19.31
Lower head failure	24.22	21.92
Drywell liner melt-through	24.45	22.14
Refueling bay to environment blowout panels open	24.45	22.14
First hydrogen deflagration in reactor building	24.46	22.15
Equipment access door opens to environment due to overpressure	24.46	22.15
Reactor building to turbine building blowout panel opens	24.50	22.21
Door to environment in southwest stairwell opens due to overpressure	-	22.21
Time Iodine release to environment exceeds 1%	24.53	22.20
Drywell liner penetration shear failure	32.03	29.53
Drywell head flange leaks	33.90	31.43
Calculation terminated (CAV rupture)	42.3	38.4

C.6 Peach Bottom, Case 2B

Long-term Station Blackout. RCIC operates for 8 hrs then terminates due to battery exhaustion. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure occurs as a result of drywell head flange leakage.

Event (Time in hours unless noted otherwise)	**LTSBO 8x8 LBU CF Late**	**LTSBO 10x10 HBU CF Late**
Station blackout [battery failure]	0 [8]	0 [8]
Downcomer water level reaches TAF	11.50	10.69
First hydrogen production	12.73	11.78
First fuel-cladding gap release	12.83	11.85
First channel box failure	13.58	12.36
First core cell collapse due to time at temperature [cell]	13.84 [111]	12.61 [111]
Reactor vessel water level reaches bottom of lower core plate	14.05	13.00
First core support plate localized failure in supporting debris	15.45	14.04
SRV sticks open	15.71	14.15
RPV pressure first drops below LPI setpoint (450 psig)	16.01	14.50
Lower head dries out	16.69	15.45
Ring 1 CRGT Column Collapse	22.68	19.07
Ring 2 CRGT Column Collapse	22.65	19.30
Ring 3 CRGT Column Collapse	22.07	19.22
Ring 4 CRGT Column Collapse	21.61	18.78
Ring 5 CRGT Column Collapse	21.88	19.19
Lower head failure	24.22	21.78
Drywell head flange leaks	25.25	22.30
Refueling bay to environment blowout panels open	25.25	22.31
First hydrogen deflagration in reactor building	25.25	22.31
Time Iodine release to environment exceeds 1%	25.30	22.37
Equipment access door opens to environment due to overpressure	-	-
Reactor building to turbine building blowout panel opens	-	-
Door to environment in southwest stairwell opens due to overpressure	-	-
Drywell liner penetration shear failure	29.63	26.01
Calculation terminated (CAV rupture)	53.9	45.1

C.7 Peach Bottom, Case 2C

Long-term Station Blackout. RCIC operates for 8 hrs then terminates due to battery exhaustion. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure occurs as a result of static over-pressure (high-temperature failure mechanisms are neglected).

Event (Time in hours unless noted otherwise)	**LTSBO 8x8 LBU CF -Pressure**	**LTSBO 10x10 HBU CF - Pressure**
Station blackout [battery failure]	0 [8]	0 [8]
Downcomer water level reaches TAF	11.50	10.69
First hydrogen production	12.73	11.78
First fuel-cladding gap release	12.83	11.85
First channel box failure	13.58	12.36
First core cell collapse due to time at temperature [cell]	13.84 [111]	12.64 [111]
Reactor vessel water level reaches bottom of lower core plate	14.05	13.00
First core support plate localized failure in supporting debris	15.45	14.04
SRV sticks open	15.71	14.15
RPV pressure first drops below LPI setpoint (450 psig)	16.01	14.50
Lower head dries out	16.69	15.45
Ring 1 CRGT Column Collapse	22.68	19.07
Ring 2 CRGT Column Collapse	22.65	19.30
Ring 3 CRGT Column Collapse	22.07	19.22
Ring 4 CRGT Column Collapse	21.61	18.78
Ring 5 CRGT Column Collapse	21.88	19.19
Lower head failure	24.22	21.78
Wetwell failure (above water level) due to overpressure	28.90	23.99
Refueling bay to environment blowout panels open	28.90	23.99
First hydrogen deflagration in reactor building	28.90	23.99
Equipment access door opens to environment due to overpressure	28.90	23.99
Reactor building to turbine building blowout panel opens	28.90	23.99
Door to environment in southwest stairwell opens due to overpressure	28.90	23.99
Refueling bay roof overpressure failure	28.90	23.99
Time Iodine release to environment exceeds 1%	28.97	24.03
Calculation terminated (CAV rupture)	42.6	37.2

C.8 Peach Bottom, Case 3

Small Break LOCA in a Main Steam Line. All forms of coolant injection fail or are unavailable for coolant makeup. Containment failure occurs as a result of drywell head flange leakage.

Event (Time in hours unless noted otherwise)	**Small LOCA 8x8 LBU**	**Small LOCA 10x10 HBU**
Station blackout and small LOCA in MSL SRV header	0	0
Reactor water level at TAF	0.66	0.65
First hydrogen production	1.20	1.18
First fuel-cladding gap release	1.24	1.21
Reactor vessel water level reaches bottom of lower core plate	1.44	1.44
First channel box failure	1.58	1.44
RPV pressure first drops below LPI setpoint (450 psig)	1.78	1.84
First core cell collapse due to time at temperature [cell]	1.78 [213]	1.61 [114]
First core support plate localized failure in supporting debris	2.49	2.16
SRV sticks open	2.62	2.36
Dryout of lower head	3.40	3.90
Ring 1 CRGT Column Collapse	6.57	5.30
Ring 2 CRGT Column Collapse	6.70	6.14
Ring 3 CRGT Column Collapse	6.50	6.40
Ring 4 CRGT Column Collapse	6.37	6.14
Ring 5 CRGT Column Collapse	7.06	6.48
Drywell head flange leaks	8.76	11.84
First hydrogen deflagration in reactor building	8.76	9.09
Refueling bay to environment blowout panels open	8.76	9.06
Lower head failure	8.80	8.69
Time Iodine release to environment exceeds 1%	8.81	9.14
Drywell liner melt-through	9.48	9.06
Equipment access door opens to environment due to overpressure	9.52	9.20
Reactor building to turbine building blowout panel opens	9.52	9.20
Door to environment in southwest stairwell opens due to overpressure	-	9.20
Drywell liner penetration shear failure	10.13	10.27
Calculation terminated (CAV rupture)	23.8	22.8

C.9 Peach Bottom, Case 4

Large Break LOCA in Recirculation Suction. All forms of coolant injection fail or are unavailable for coolant makeup. Containment failure occurs as a result of drywell head flange leakage.

Event (Time in hours unless noted otherwise)	**Large LOCA 8x8 LBU**	**Large LOCA 10x10 HBU**
Station blackout and large LOCA in recirculation loop A suction piping	0	0
RCIC and HPCI initiated	3.7 sec.	3.8 sec.
Reactor water level at TAF	6.7 sec.	6.9 sec.
Reactor vessel water level reaches bottom of lower core plate	18 sec.	19 sec.
RPV pressure first drops below LPI setpoint (450 psig)	32 sec.	32 sec.
RCIC and HPCI isolation due to low steam line pressure	1.3 min.	1.3 min.
First hydrogen production	7.9 min.	7.0 min.
First fuel-cladding gap release	8.8 min.	7.7 min.
First channel box failure	0.47	0.38
First core cell collapse due to time at temperature [cell]	0.63 [112]	0.52 [112]
First core support plate localized failure in supporting debris	1.49	1.20
Ring 1 CRGT Column Collapse	1.51	0.52
Ring 2 CRGT Column Collapse	1.51	1.24
Ring 3 CRGT Column Collapse	1.55	0.55
Dryout of lower head	2.12	1.91
Ring 4 CRGT Column Collapse	3.92	3.49
Ring 5 CRGT Column Collapse	4.51	3.90
Lower head failure	6.58	5.85
Drywell liner melt-through	6.96	6.42
Refueling bay to environment blowout panels open	6.96	6.42
First hydrogen deflagration in reactor building	7.04	6.50
Equipment access door opens to environment due to overpressure	7.77	6.71
Time Iodine release to environment exceeds 1%	7.79	6.83
Reactor building to turbine building blowout panel opens	9.23	8.42
Door to environment in southwest stairwell opens due to overpressure	-	-
Drywell liner penetration shear failure	10.63	9.57
Drywell head flange leakage	13.87	12.19
Calculation terminated (CAV rupture)	17.4	17.4

C.10 Grand Gulf, Case 5A

Short-term Station Blackout. One (1) SRV seizes in the open position before reactor vessel breach. Containment and drywell failure occur as a consequence of hydrogen combustion immediately following reactor vessel breach.

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU CF@VB**	**STSBO 10x10 HBU CF@VB**
Station blackout [battery failure]	0 [0]	0 [0]
Reactor water level at TAF	0.66	0.65
First hydrogen production	1.18	1.14
First fuel-cladding gap release	1.23	1.17
First channel box failure	1.57	1.48
First core cell collapse due to time at temperature [cell]	1.85 [113]	1.69 [112]
First core support plate localized failure in supporting debris	3.10	2.86
SRV sticks open [open fraction]	4.30 [0.041]	4.57 [0.041]
Dryout of lower head	3.84	3.64
Ring 1 CRGT Column Collapse	8.04	8.31
Ring 2 CRGT Column Collapse	7.94	8.12
Ring 3 CRGT Column Collapse	7.67	8.14
Ring 4 CRGT Column Collapse	7.09	7.80
Ring 5 CRGT Column Collapse	7.67	7.73
Lower head failure	10.46	10.48
First hydrogen burn initiated in reactor pedestal	10.46	10.48
First hydrogen burn initiated in containment	10.46	10.48
Containment quasi-static overpressure failure	10.46	10.48
Drywell external quasi-static overpressure failure	10.46	10.48
Suppression pool water forced over weir wall into reactor pedestal	10.46	10.48
Hydrogen burns cease in drywell	10.47	10.49
Hydrogen burns cease in containment	10.47	10.49
Time Iodine release to environment exceeds 1%	11.58	11.04
Calculation terminated	33.1	34.5

C.12 Grand Gulf, Case 5B

Short-term Station Blackout. All SRVs operate as designed in safety relief mode, therefore, reactor vessel breach occurs at high pressure. Containment and drywell failure occur as a consequence of hydrogen combustion immediately following reactor vessel breach.

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU High P. CF@VB**	**STSBO 10x10 HBU High P. CF@VB**
Station blackout [battery failure]	0 [0]	0 [0]
Reactor water level at TAF	0.66	0.65
First hydrogen production	1.18	1.14
First fuel-cladding gap release	1.23	1.17
First channel box failure	1.57	1.48
First core cell collapse due to time at temperature [cell]	1.85 [113]	1.69 [112]
First core support plate localized failure in supporting debris	3.10	2.86
Dryout of lower head	3.84	3.64
Ring 1 CRGT Column Collapse	9.21	8.77
Ring 2 CRGT Column Collapse	9.21	8.69
Ring 3 CRGT Column Collapse	9.16	8.78
Ring 4 CRGT Column Collapse	9.08	8.68
Ring 5 CRGT Column Collapse	9.31	8.71
Lower head failure	8.70	8.30
First hydrogen burn initiated in reactor pedestal	8.70	8.30
First hydrogen burn initiated in containment	8.70	8.30
Containment quasi-static overpressure failure	8.70	8.30
Drywell external quasi-static overpressure failure	8.70	8.30
Suppression pool water forced over weir wall into reactor pedestal	8.70	8.30
Hydrogen burns cease in drywell	8.71	8.30
Time Iodine release to environment exceeds 1%	8.78	8.77
Hydrogen burns cease in containment	9.37	8.79
Calculation terminated	37.4	36.0

C.11 Grand Gulf, Case 5C

Short-term Station Blackout. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure occurs as a consequence of static over-pressure late in time.

Event (Time in hours unless noted otherwise)	**STSBO 8x8 LBU CF Late**	**STSBO 10x10 HBU CF Late**
Station blackout [battery failure]	0 [0]	0 [0]
Reactor water level at TAF	0.66	0.65
First hydrogen production	1.18	1.14
First fuel-cladding gap release	1.23	1.17
First channel box failure	1.58	1.48
First core cell collapse due to time at temperature [cell]	1.81 [112]	1.70 [113]
First core support plate localized failure in supporting debris	3.19	2.78
SRV sticks open [open fraction]	4.35 [0.041]	4.05 [0.041]
Dryout of lower head	3.99	3.93
Ring 1 CRGT Column Collapse	8.75	8.18
Ring 2 CRGT Column Collapse	8.61	7.74
Ring 3 CRGT Column Collapse	8.33	7.43
Ring 4 CRGT Column Collapse	7.90	7.13
Ring 5 CRGT Column Collapse	8.35	7.29
Lower head failure	11.06	9.92
Containment quasi-static overpressure failure	64.05	62.15
Time Iodine release to environment exceeds 1%	64.06	62.25
Calculation terminated	168.	168.

C.13 Grand Gulf, Case 6A

Long-term Station Blackout. dc control power for RCIC is available for 8 hrs. However, automatic RCIC isolation occurs earlier as a result of high suppression pool temperature. One (1) SRV seizes in the open position before reactor vessel breach. Containment and drywell failure occur as a consequence of hydrogen combustion immediately following reactor vessel breach.

Event (Time in hours unless noted otherwise)	**LTSBO 8x8 LBU CF@VB**	**LTSBO 10x10 HBU CF@VB**
Station blackout [battery failure]	0 [8]	0 [8]
RCIC suction transferred to suppression pool due to high pool level	1.52	1.45
RCIC isolation due to high suppression pool temperatures	2.61	2.51
Reactor water level at TAF	4.45	4.21
First hydrogen production	5.29	4.96
First fuel-cladding gap release	5.35	5.01
First channel box failure	5.89	5.40
First core cell collapse due to time at temperature [cell]	6.27 [311]	5.70 [112]
First core support plate localized failure in supporting debris	7.48	6.81
SRV sticks open [open fraction]	10.29 [0.041]	9.97 [0.041]
Dryout of lower head	9.31	8.11
Ring 1 CRGT Column Collapse	15.11	13.68
Ring 2 CRGT Column Collapse	14.99	13.17
Ring 3 CRGT Column Collapse	14.74	13.30
Ring 4 CRGT Column Collapse	14.55	13.18
Ring 5 CRGT Column Collapse	15.12	13.84
Lower head failure	17.72	16.22
First hydrogen burn initiated in reactor pedestal	17.72	16.22
First hydrogen burn initiated in containment	17.72	16.22
Containment quasi-static overpressure failure	17.72	16.22
Drywell external quasi-static overpressure failure	17.72	16.22
Suppression pool water forced over weir wall into reactor pedestal	17.72	16.22
Hydrogen burns cease in drywell	17.73	16.23
Hydrogen burns cease in containment	17.73	16.23
Time Iodine release to environment exceeds 1%	25.14	16.24
Calculation terminated	68.3	56.8

C.14 Grand Gulf, Case 6B

Long-term Station Blackout. dc control power for RCIC is available for 8 hrs. However, automatic RCIC isolation occurs earlier as a result of high suppression pool temperature. One (1) SRV seizes in the open position before reactor vessel breach. Containment failure occurs as a consequence of static over-pressure late in time.

Event (Time in hours unless noted otherwise)	**LTSBO 8x8 LBU CF Late**	**LTSBO 10x10 HBU CF Late**
Station blackout [battery failure]	0 [8]	0 [8]
RCIC suction transferred to suppression pool due to high pool level	1.52	1.45
RCIC isolation due to high suppression pool temperatures	2.61	2.51
Reactor water level at TAF	4.45	4.21
First hydrogen production	5.29	4.96
First fuel-cladding gap release	5.35	5.01
First channel box failure	5.89	5.40
First core cell collapse due to time at temperature [cell]	6.27 [311]	5.70 [112]
First core support plate localized failure in supporting debris	7.48	6.81
SRV sticks open [open fraction]	10.29 [0.041]	9.97 [0.041]
Dryout of lower head	9.31	8.11
Ring 1 CRGT Column Collapse	15.11	13.68
Ring 2 CRGT Column Collapse	14.99	13.17
Ring 3 CRGT Column Collapse	14.74	13.30
Ring 4 CRGT Column Collapse	14.55	13.18
Ring 5 CRGT Column Collapse	15.12	13.84
Lower head failure	17.72	16.22
Containment quasi-static overpressure failure	57.10	33.09
Time Iodine release to environment exceeds 1%	57.17	50.08
Calculation terminated	168.	168.

C.15 Grand Gulf, Case 7

Anticipated Transient Without Scram (ATWS). MSIV closure is initiating event. Manual intervention is assumed unsuccessful (ADS not inhibited, low-pressure injection systems actuate and cycle on demand to maintain level). Containment fails due to over-pressure (steaming from saturated suppression pool).

Event (Time in hours unless noted otherwise)	**ATWS 8x8 LBU**	**ATWS 10x10 HBU**
Spurious MSIV closure leading to ATWS	0	0
RCIC initiated and recirculation pump trip	32 sec.	33 sec.
Reactor water level at TAF	2.1 min.	2.1 min.
RCIC suction transferred to suppression pool due to high pool level	2.1 min.	2.1 min.
RCIC isolation due to high suppression pool temperatures	3.9 min	3.9 min.
RHR initiated in suppression pool cooling mode	5 min.	5 min.
ADS actuation	13 min.	13 min.
Feedwater failure due to hotwell inventory depletion	0.41	0.34
High and low pressure core spray initiated to maintain core level at TAF	0.41	0.34
Containment overpressure failure	8.18	6.76
HPCS, LPCS, RHR and CRDHS fail due to containment failure	8.18	6.76
First hydrogen production	9.11	7.66
First fuel-cladding gap release	9.15	7.70
First channel box failure	9.80	8.24
First core cell collapse due to time at temperature [cell]	9.98 [110]	8.54 [113]
First core support plate localized failure in supporting debris	11.30	9.73
Ring 1 CRGT Column Collapse	11.33	9.76
Dryout of lower head	14.04	12.64
Ring 2 CRGT Column Collapse	17.89	9.83
Ring 3 CRGT Column Collapse	16.81	15.45
Ring 4 CRGT Column Collapse	17.55	15.85
Ring 5 CRGT Column Collapse	18.39	16.36
Lower head failure	21.41	19.29
Total In-vessel H_2 production (kg)	1365.	1232.
Time Iodine release to environment exceeds 1%	24.54	36.24
Calculation terminated	51.9	47.7

C.16 Grand Gulf, Case 8

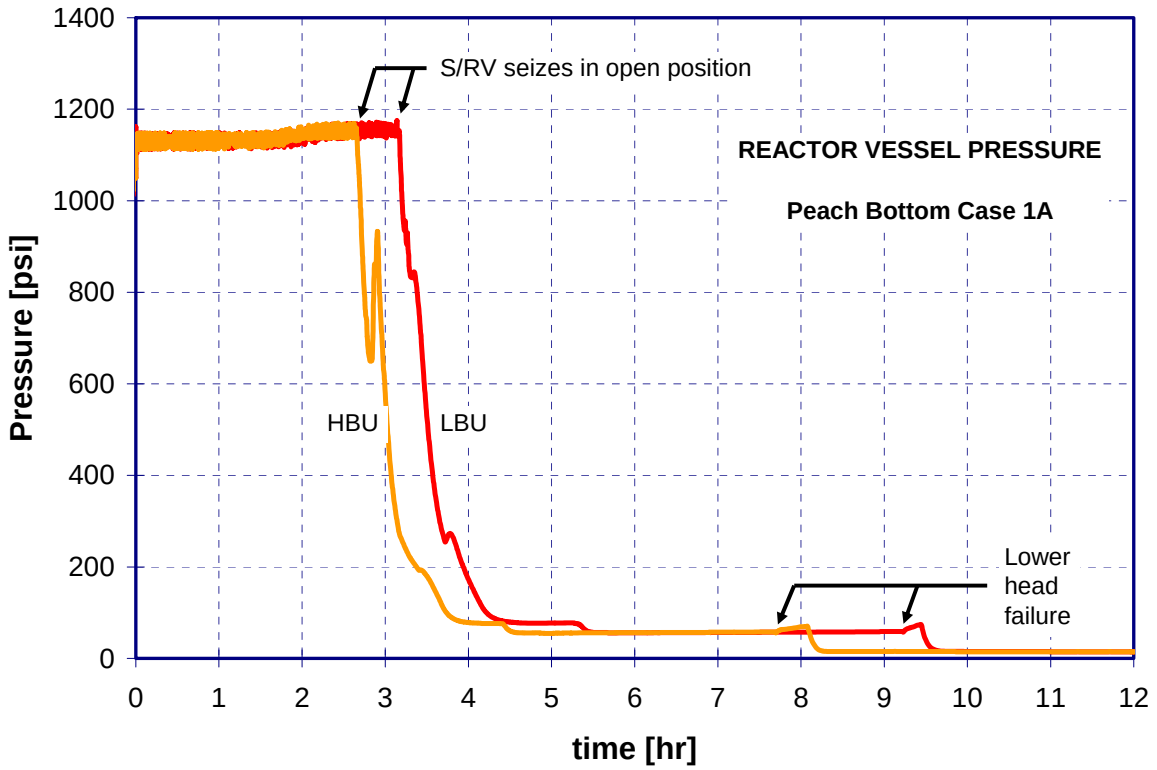
Large (Recirculation Suction) LOCA. RCIC actuates and operates until reactor vessel depressurizes below minimum turbine operating pressure. All forms of low-pressure injection fail or are unavailable.

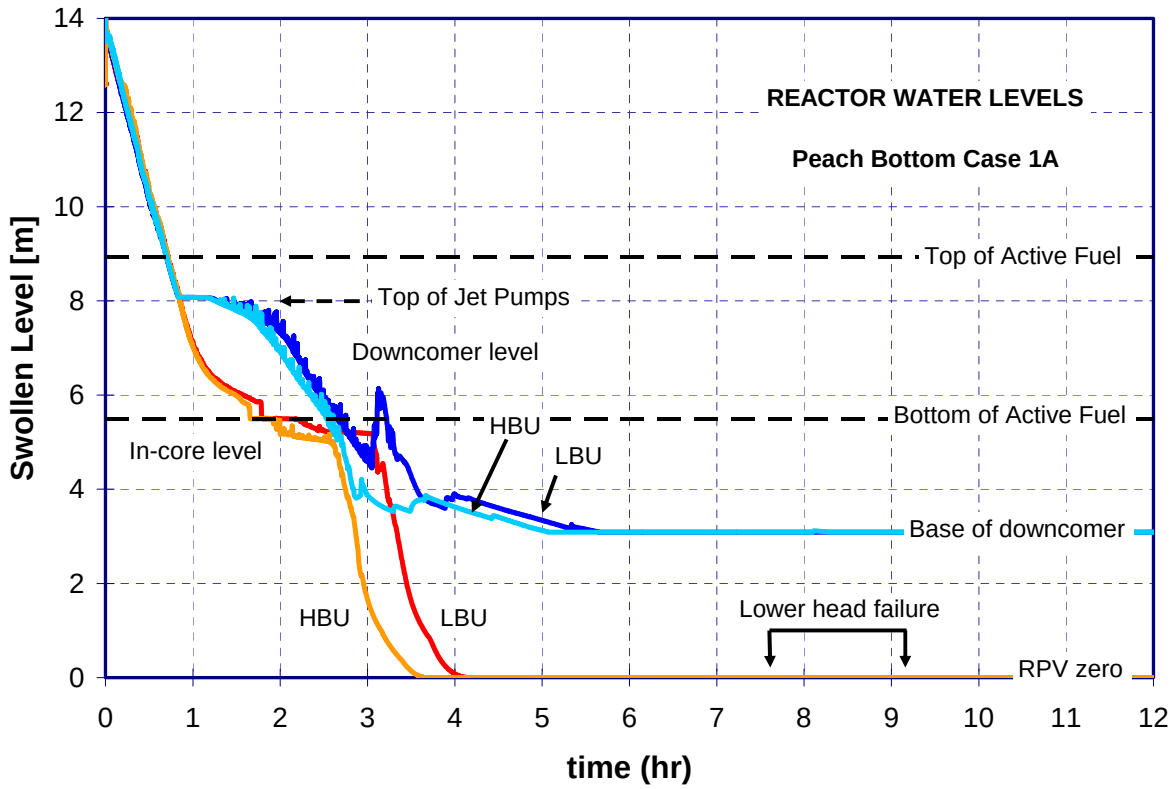
Event (Time in hours unless noted otherwise)	**Large LOCA 8x8 LBU**	**Large LOCA 10x10 HBU**
Station blackout and large LOCA in recirculation loop A suction piping	0	0
RCIC suction aligned to suppression pool – high suppression pool level	1.4 sec.	1.4 sec.
RCIC initiated	2.4 sec.	2.4 sec.
Reactor water level at TAF	5.3 sec.	5.6 sec.
RCIC isolation due to low steam line pressure	1.3 min.	1.3 min.
ADS actuation	1.8 min.	1.8 min.
First hydrogen production	8.3 min.	7.1 min.
First fuel-cladding gap release	9.2 min.	7.8 min.
First channel box failure	0.42	0.38
First core cell collapse due to time at temperature [cell]	0.62 [111]	0.52 [111]
First core support plate localized failure in supporting debris	1.62	1.43
Ring 1 CRGT Column Collapse	1.65	1.44
Ring 2 CRGT Column Collapse	1.63	1.47
Ring 3 CRGT Column Collapse	1.67	1.52
Dryout of lower head	2.25	2.07
Ring 4 CRGT Column Collapse	3.89	3.68
Ring 5 CRGT Column Collapse	4.63	4.79
Lower head failure	7.00	5.51
First hydrogen deflagration in containment	7.00	5.51
Containment overpressure failure	35.96	5.51
Time Iodine release to environment exceeds 1%	-	5.52
Calculation terminated (CAV rupture)	51.4	44.6

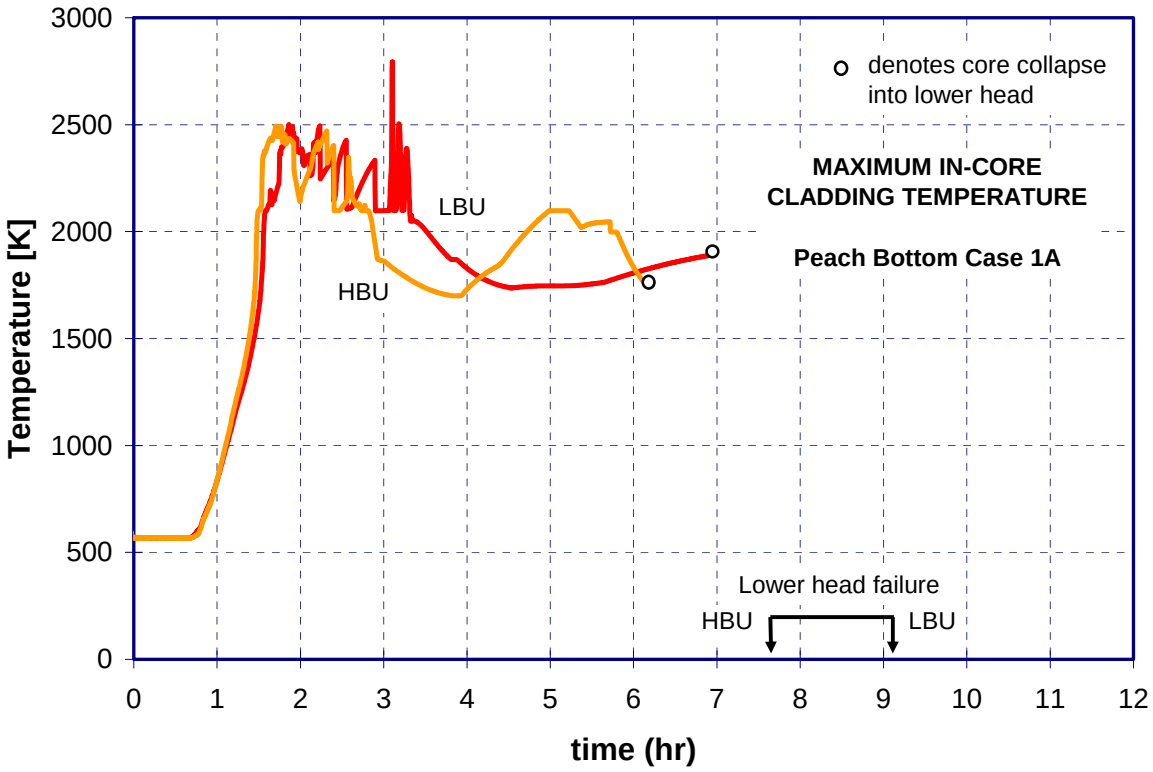
Appendix D. Accident Progression Signatures for Selected Accident Sequences

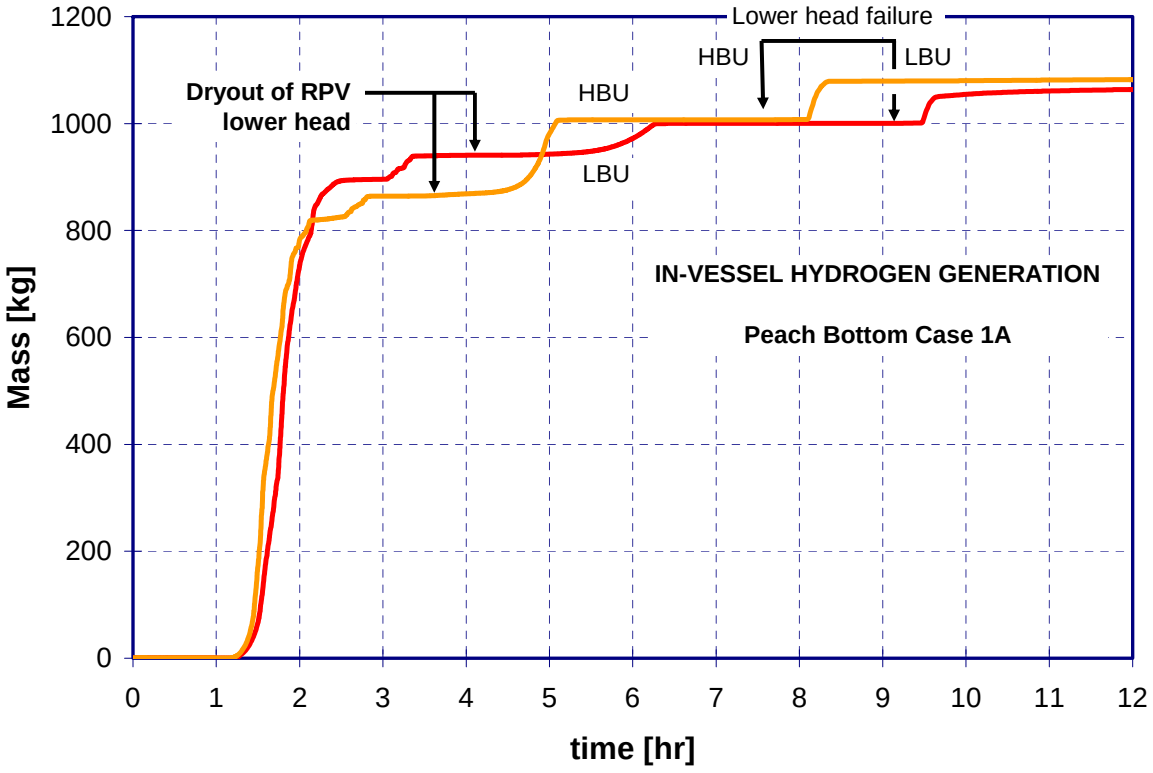
The following parameters are plotted from both the LBU and HBU calculations for a short-term station blackout sequence with containment failure at vessel breach in the BWR/4 Mark I (Peach Bottom) model.

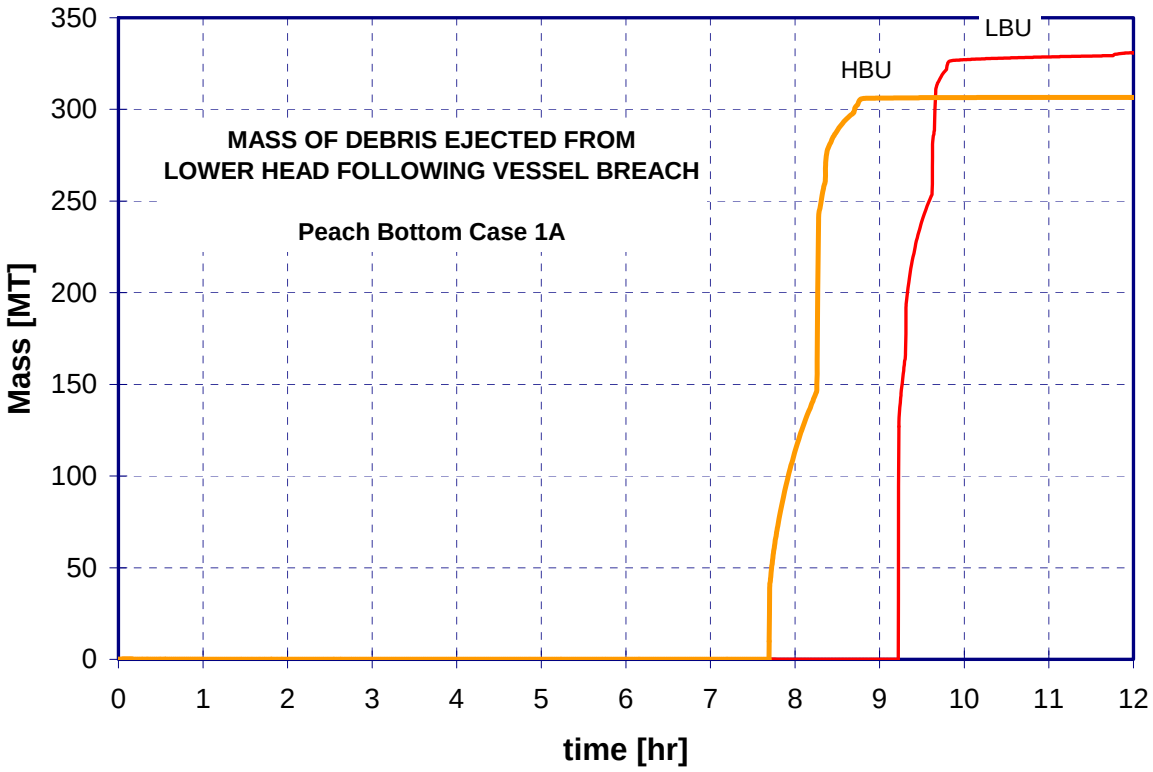
- Reactor vessel pressure (psia)
- Reactor vessel water levels – downcomer and in-core (m)
- Maximum fuel cladding temperature in the core (K)
- Total quantity of hydrogen generated by in-vessel oxidation (kg)
- Mass of core debris ejected from the RPV lower head following vessel breach (MT)
- Containment pressure (psia)
- Iodine released from fuel (fraction of initial inventory)
- Iodine released to containment (fraction of initial inventory)

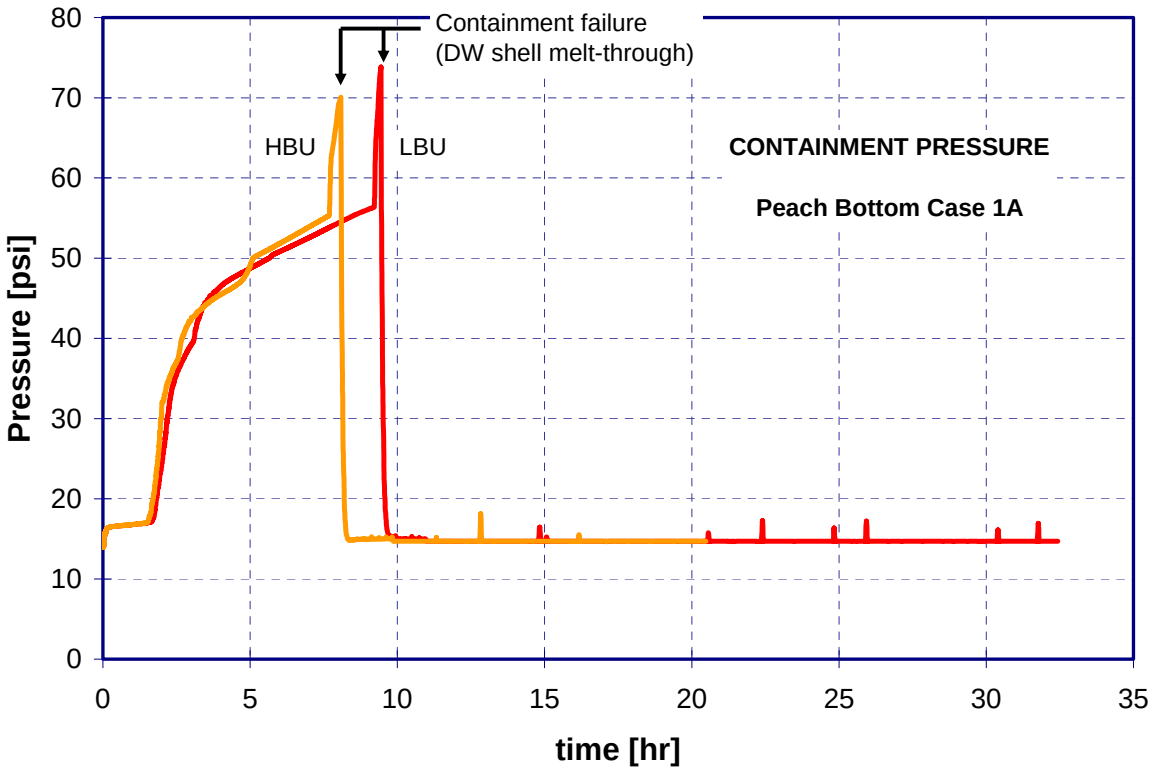

REACTOR VESSEL PRESSURE
Peach Bottom Case 1A
S/RV seizes in open position
HBU
LBU
Lower head failure
Pressure [psi]
time [hr]
0
200
400
600
800
1000
1200
1400
0
1
2
3
4
5
6
7
8
9
10
11
12

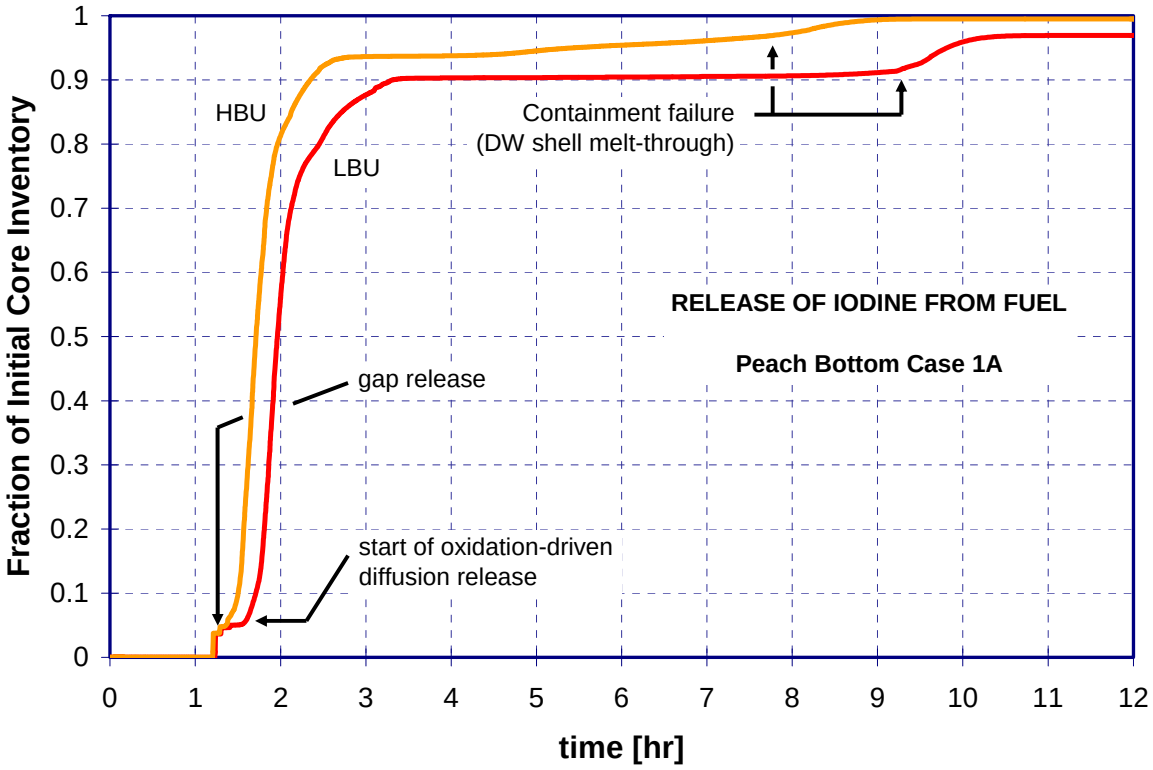

REACTOR WATER LEVELS
Peach Bottom Case 1A
Top of Active Fuel
Top of Jet Pumps
Downcomer level
Bottom of Active Fuel
In-core level
HBU
LBU
Base of downcomer
HBU
LBU
Lower head failure
RPV zero
Swollen Level [m]
time (hr)
0
2
4
6
8
10
12
14
0
1
2
3
4
5
6
7
8
9
10
11
12

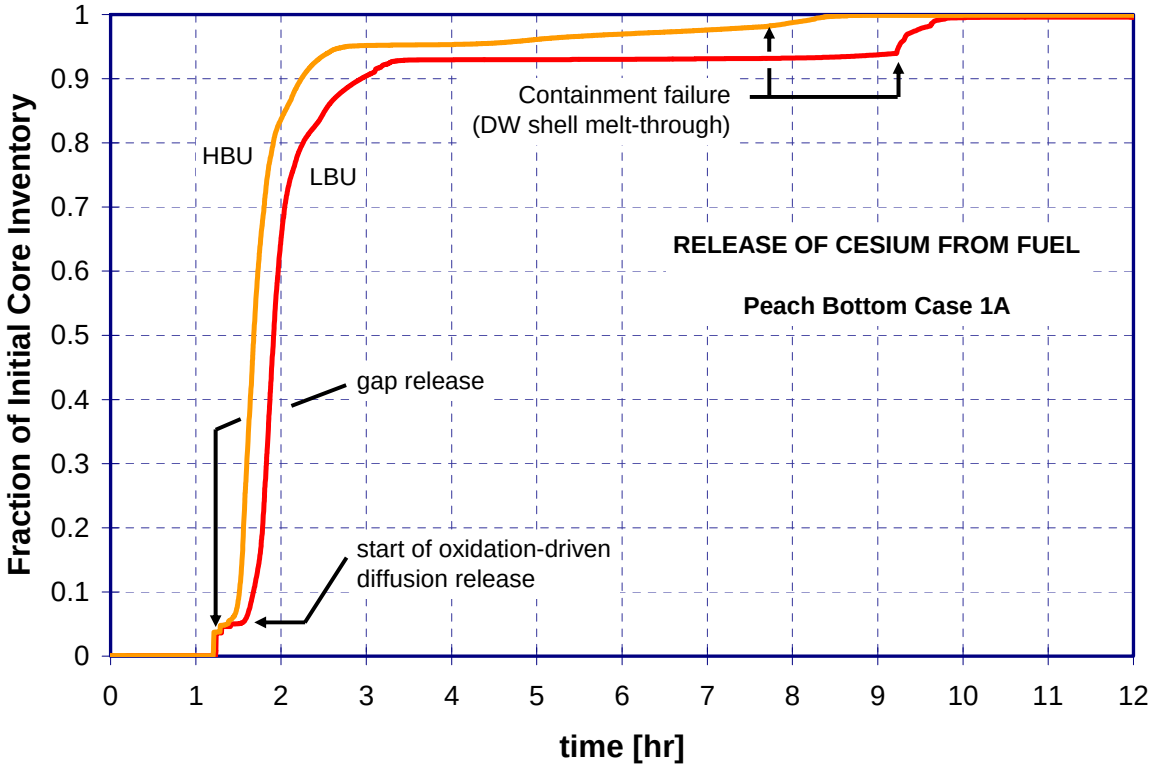

3000
2500
2000
1500
1000
500
0
Temperature [K]
0
1
2
3
4
5
6
7
8
9
10
11
12
time (hr)
o denotes core collapse into lower head
MAXIMUM IN-CORE CLADDING TEMPERATURE
Peach Bottom Case 1A
LBU
HBU
Lower head failure
HBU
LBU

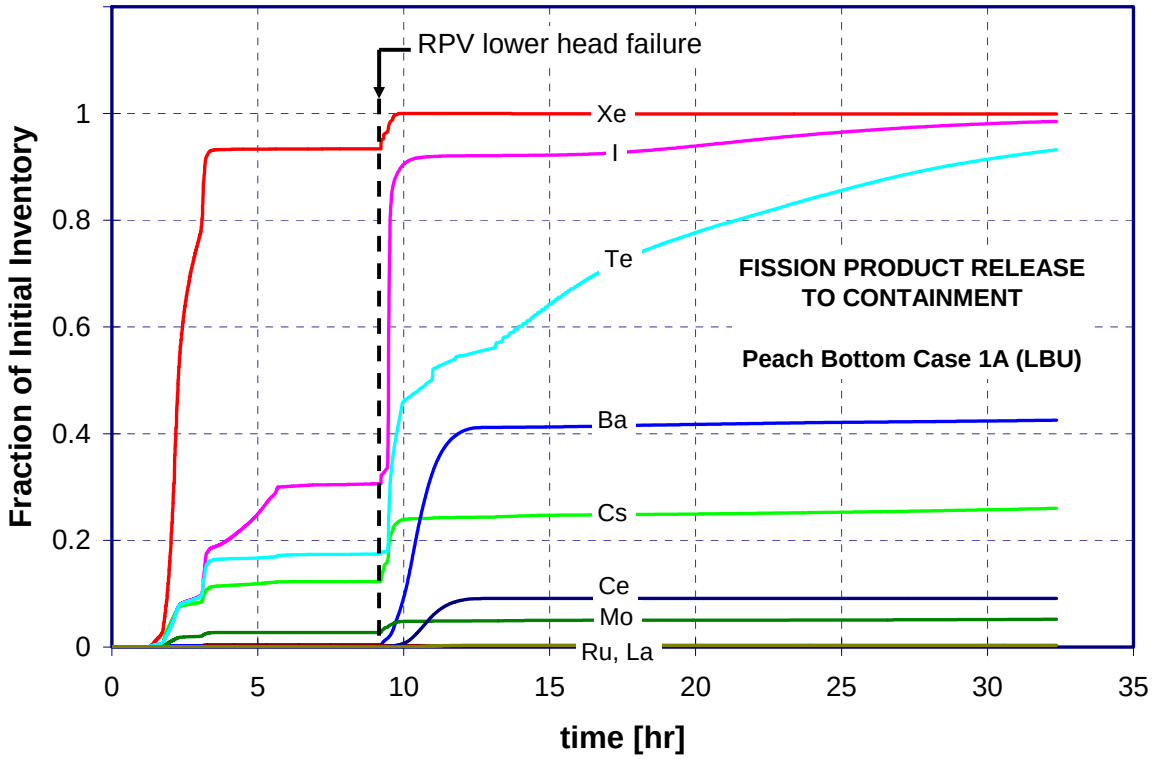

1200
1000
800
600
400
200
0
Mass [kg]
0
1
2
3
4
5
6
7
8
9
10
11
12
time [hr]
Lower head failure
HBU
LBU
Dryout of RPV lower head
HBU
LBU
IN-VESSEL HYDROGEN GENERATION
Peach Bottom Case 1A

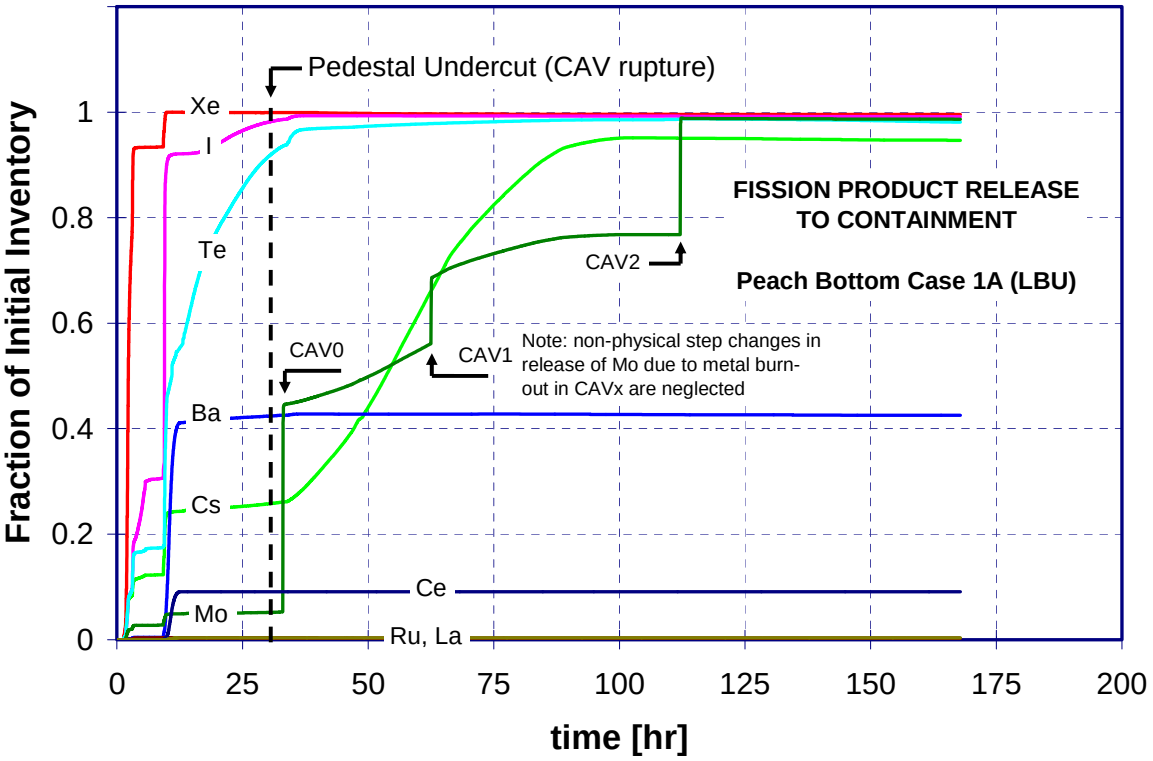

MASS OF DEBRIS EJECTED FROM
LOWER HEAD FOLLOWING VESSEL BREACH
Peach Bottom Case 1A
LBU
HBU
Mass [MT]
time [hr]
0
50
100
150
200
250
300
350
0
1
2
3
4
5
6
7
8
9
10
11
12

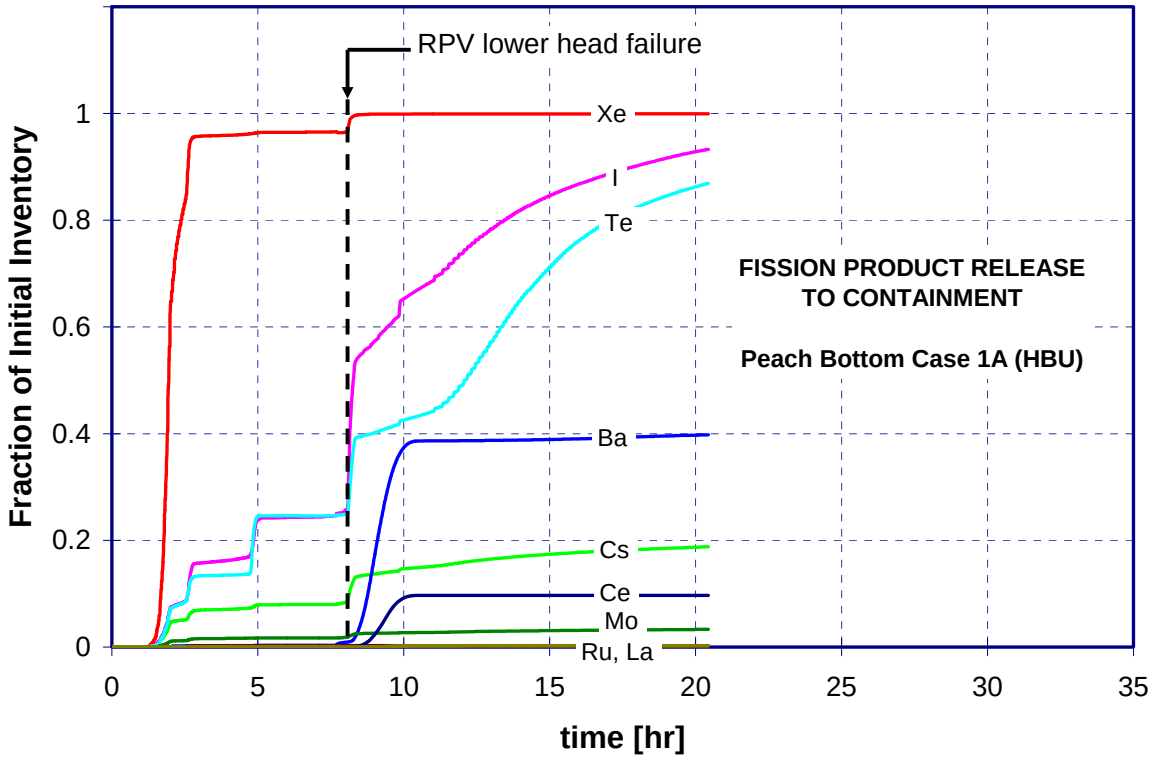

Containment failure
(DW shell melt-through)
HBU
LBU
CONTAINMENT PRESSURE
Peach Bottom Case 1A
Pressure [psi]
time [hr]
0
10
20
30
40
50
60
70
80
0
5
10
15
20
25
30
35

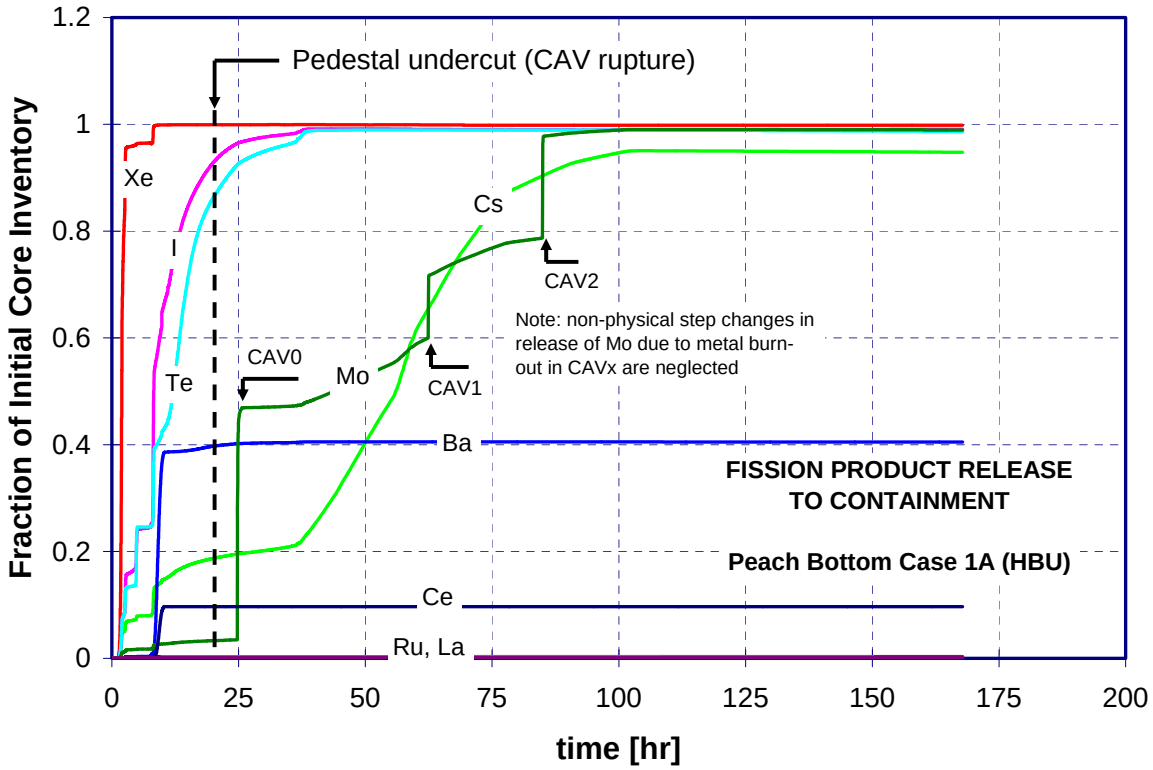

Fraction of Initial Core Inventory
time [hr]
HBU
LBU
Containment failure
(DW shell melt-through)
RELEASE OF IODINE FROM FUEL
Peach Bottom Case 1A
gap release
start of oxidation-driven
diffusion release


Fraction of Initial Core Inventory
time [hr]
HBU
LBU
Containment failure
(DW shell melt-through)
RELEASE OF CESIUM FROM FUEL
Peach Bottom Case 1A
gap release
start of oxidation-driven
diffusion release


RPV lower head failure
Xe
I
Te
Ba
Cs
Ce
Mo
Ru, La
FISSION PRODUCT RELEASE
TO CONTAINMENT
Peach Bottom Case 1A (LBU)
Fraction of Initial Inventory
time [hr]
0
0.2
0.4
0.6
0.8
1
0
5
10
15
20
25
30
35


Pedestal Undercut (CAV rupture)
Xe
I
Te
Ba
Cs
Ce
Mo
Ru, La
CAV0
CAV1
CAV2
Note: non-physical step changes in
release of Mo due to metal burn-
out in CAVx are neglected
FISSION PRODUCT RELEASE
TO CONTAINMENT
Peach Bottom Case 1A (LBU)
Fraction of Initial Inventory
time [hr]
0
0.2
0.4
0.6
0.8
1
0
25
50
75
100
125
150
175
200


RPV lower head failure
Xe
I
Te
Ba
Cs
Ce
Mo
Ru, La
FISSION PRODUCT RELEASE
TO CONTAINMENT
Peach Bottom Case 1A (HBU)
Fraction of Initial Inventory
time [hr]
0
0.2
0.4
0.6
0.8
1
0
5
10
15
20
25
30
35


Pedestal undercut (CAV rupture)
Xe
I
Te
Cs
CAV0
CAV1
CAV2
Mo
Ba
Ce
Ru, La
Note: non-physical step changes in
release of Mo due to metal burn-
out in CAVx are neglected
FISSION PRODUCT RELEASE
TO CONTAINMENT
Peach Bottom Case 1A (HBU)
Fraction of Initial Core Inventory
time [hr]
0
0.2
0.4
0.6
0.8
1
1.2
0
25
50
75
100
125
150
175
200

Distribution

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