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Accident Source Terms for Pressurized Water Reactors with High-Burnup Cores Calculated using MELCOR 1.8.5

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Abstract

In this study, risk-significant pressurized-water reactor severe accident sequences are examined using MELCOR 1.8.5 to explore the range of fission product releases to the reactor containment building. Advances in the understanding of fission product release and transport behavior and severe accident progression are used to render best estimate analyses of selected accident sequences. Particular emphasis is placed on estimating the effects of high fuel burnup in contrast with low burnup on fission product releases to the containment. Supporting this emphasis, recent data available on fission product release from high-burnup (HBU) fuel from the French VERCOR project are used in this study. The results of these analyses are treated as samples from a population of accident sequences in order to employ approximate order statistics characterization of the results. These trends and tendencies are then compared to the NUREG-1465 alternative source term prescription used today for regulatory applications.

In general, greater differences are observed between the state-of-the-art calculations for either HBU or low-burnup (LBU) fuel and the NUREG-1465 containment release fractions than exist between HBU and LBU release fractions. Current analyses suggest that retention of fission products within the vessel and the reactor coolant system (RCS) are greater than contemplated in the NUREG-1465 prescription, and that, overall, release fractions to the containment are therefore lower across the board in the present analyses than suggested in NUREG-1465. The decreased volatility of Cs_2MoO_4 compared to CsI or CsOH increases the predicted RCS retention of cesium, and as a result, cesium and iodine do not follow identical behaviors with respect to distribution among vessel, RCS, and containment. With respect to the regulatory alternative source term, greater differences are observed between the NUREG-1465 prescription and both HBU and LBU predictions than exist between HBU and LBU analyses. Additionally, current analyses suggest that the NUREG-1465 release fractions are conservative by about a factor of 2 in terms of release fractions and that release durations for in-vessel and late in-vessel release periods are in fact longer than the NUREG-1465 durations.

It is currently planned that a subsequent report will further characterize these results using more refined statistical methods, permitting a more precise reformulation of the NUREG-1465 alternative source term for both LBU and HBU fuels, with the most important finding being that the NUREG-1465 formula appears to embody significant conservatism compared to current best-estimate analyses.

ACKNOWLEDGEMENTS

This work was supported by the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

The authors would like to thank Dr. Ian Gauld and Dr. Germina Ilas, of Oak Ridge National Laboratory, for their contributions to this work. In addition to development of core fission product inventory and decay heat information for use in MELCOR models, their insights related to fuel management practices and resulting effects on spatial distribution of fission products in the core was instrumental in completion of our work.

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ACRONYMS

AFW	auxiliary feedwater
AST	alternative source term
BWR	boiling-water reactor
CDF	core damage frequency
CF	containment failure
CFR	Code of Federal Regulations
COR	Core (MELCOR Package)
Cs	cesium
CS	containment sprays
Cs_2MoO_4	cesium molybdate
CVJ	Control Volume Hydrodynamics (MELCOR Package)
DCH	direct containment heating
ECCS	emergency core cooling system
FC	fan coolers
FSAR	Final Safety Analysis Report
HBU	high burnup
INEL	Idaho National Engineering Laboratory
IPE	Individual Plant Examination
ISLOCA	interfacing system loss-of-coolant accident
Kr	krypton
LBU	low burnup
LEU	low-enriched uranium
LLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LWR	light water reactor
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
PCS	power conversion system
PWR	pressurized-water reactor

RCP	reactor coolant pump
RCS	reactor coolant system
SBLOCA	small-break loss-of-coolant accident
SBO	station blackout
SGTR	steam generator tube rupture
SRV	safety relief valve
STCP	Source Term Code Package
TD	turbine-driven
TID	Technical Information Document
TMI	Three Mile Island
UO_2	uranium dioxide
VF	vessel failure
Xe	xenon
Zr	zirconium

1 INTRODUCTION AND BACKGROUND

1.1 Regulatory Use of 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 [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 [2,3]. These specifications were derived from the 1962 report TID-14844 [4], which described fission product release based on very early studies involving heated, irradiated uranium dioxide (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 accidents 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 [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 particulates and substantial retention of these particulates 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 [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.

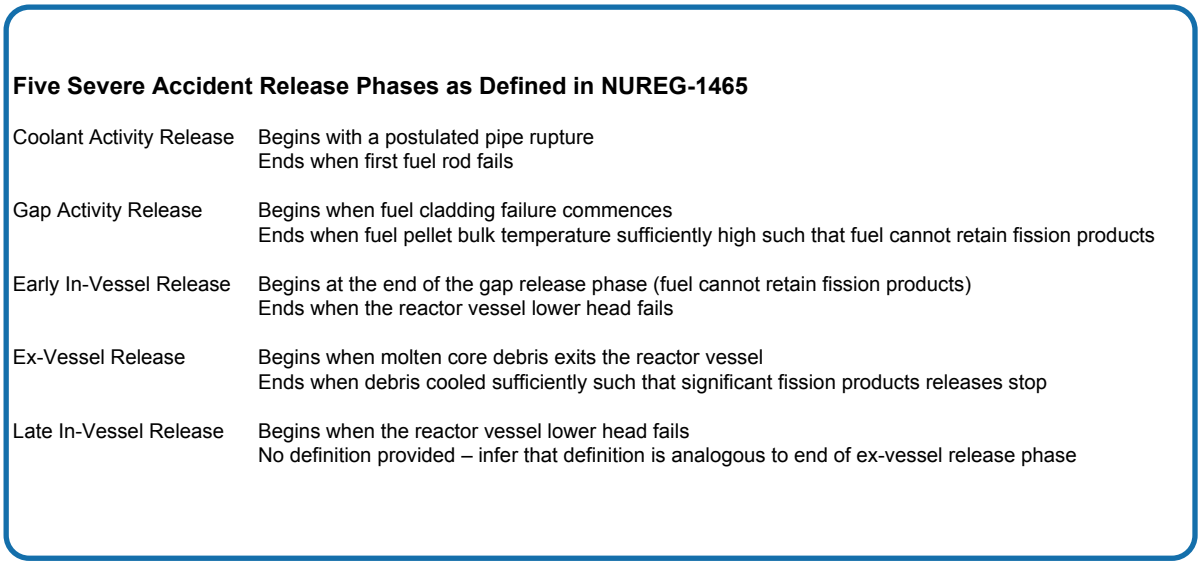

Figure 1. NUREG-1465 Severe Accident Release Phases.

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 pressurized-water reactors (PWRs) and boiling-water reactors (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), early versions of the MELCOR accident analysis code, and expert judgment.

The coolant activity release is the expulsion of mildly 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, fuel continues to heat and radionuclides diffuse from the fuel and vaporize. This phase is referred to as the "in-vessel" release phase in NUREG-1465. 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 liquefied fuel and clad penetrates the reactor vessel and cascades into the reactor cavity. Processes contributing to the ex-vessel release include the 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 PWRs, pressurization could be maintained throughout the degradation process. Releases associated with core debris interactions with concrete depend significantly on the amounts of metallic zirconium (Zr) 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 vaporization of deposited radionuclides into the natural circulation of gases through the ruptured RCS can occur. The re-vaporization from surfaces is slow but 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 PWRs 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 Pressurized-Water Reactors. *

	Gap Release***	In-vessel	Ex-vessel	Late In-vessel
Duration (hours)	0.5	1.3	2.0	10.0
Noble Gases**	0.05	0.95	0	0
Halogens	0.05	0.35	0.25	0.1
Alkali Metals	0.05	0.25	0.35	0.1
Tellurium Group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Lanthanides	0	0.0002	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 (xenon [Xe] and krypton [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 also released as a result of core degradation. The behavior of aerosol is especially affected by the nonradioactive aerosol produced in the accident as a result of the increase in agglomeration rate between individual aerosol particles. Certainly, this nonradioactive aerosol can include control rod materials, alloying agents from the cladding and the like. Most especially, 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 (LWRs). 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 UO_2 fuel taken to burnups typical of the time when the source term was developed. This burnup was usually less than 40 GWd/t. 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 [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 [8] is a noteworthy example, and widely used around the world by research and licensing agencies.

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 1-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 [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 inconsistent with the 5% assumed in the NUREG-1465 Source Term.[1] 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 [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.

[1] Very recent results from Phebus test FPT-3 may challenge this assumption as higher gaseous iodine was observed to be released to the containment. One proposed explanation is that speciation effects involving boron compounds so effectively tie up cesium that iodine speciation is biased towards increased elemental form.

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 aspect of industry evolution that is important 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 GWd/t. Today, fuels are being taken to higher burnups approaching the regulatory limit, which is 62 GWd/t average for the lead assembly. This increase in burnup makes advances in fuel design features important. Some of these advances include:

- It is anticipated that with changes in some plant features, including the steam generators, PWRs may in the future be able to operate at substantially higher powers. Further increases in fuel use will require improved fuel cladding. Interest is now focused on fuel cladding that increases corrosion resistance for the demands of increased burnup. Corrosion-resistant clad types include a proprietary niobium alloy of Zr called M5, Duplex cladding, optimized Zircaloy-4 cladding (e.g., ZIRLO), LTP2 or iron-lined Zircaloy cladding. All of these are being considered and/or currently used by fuel manufacturers to replace conventional Zircaloy alloys as cladding materials. The plants studied here are currently employing fuel designs with ZIRLO cladding (Surry) and M5 cladding (Sequoyah). The importance of advanced cladding materials is further discussed in Section 3.5.

- Higher U-235 enrichment, the extensive use of power-shaping features in core design (e.g., burnable poisons), and significant changes in fuel assembly mechanical design (e.g., number of rods/assembly) can have a significant impact on the maximum core fission product inventory, the spatial distribution of those fission products within the core, and the associated decay heat distribution within the core. These modern core design features have been incorporated into the MELCOR core models for HBU cores, as well as the ORIGEN decay heat and fission product inventory calculations, as discussed in Section 2.3.

2 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 PWRs 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.

The accidents selected for the analysis are biased to include those that contribute significantly to the overall risk posed by the operation of 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 loss-of-coolant accidents (LLOCAs) are not risk-significant for PWRs. 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 GWd/t cited in NUREG-1465 to the regulatory limit of 62 GWd/t. 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 [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 United States. 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 PWR reactors selected to form the basis for the HBU supplement to the NUREG-1465 Source Term are the Sequoyah Nuclear Plant and the Surry Power Station.

2.2.1 Sequoyah

The Sequoyah Nuclear Plant consists of two PWR units located in east Tennessee 18 miles north of Chattanooga, on the banks of Chickamauga Reservoir. Sequoyah Unit 1 began full commercial operation in 1981. Each Sequoyah unit is a Westinghouse four-loop nuclear steam supply system (NSSS) housed within an ice condenser containment. A photo of the Sequoyah site is shown in Figure 2. Each of the Sequoyah reactors currently operates at a nominal power of 3455 MWth. Previous to the power uprate that occurred in 2002, the reactors operated at a nominal power of 3411 MWth.

Figure removed for copyright reasons

Figure 2. Sequoyah Nuclear Plant.

The Sequoyah MELCOR model used for the HBU Source Term analyses was based on that used in the NRC's plant security assessment for plants with ice condenser containments [12]. A general description of the Sequoyah MELCOR model can be found in Appendix B.

2.2.2 Surry

The Surry Power Station consists of two PWR units located at Gravel Neck along the James River in Surry County, VA. Surry Unit 1 began commercial operation in December 1972. Each Surry unit is a Westinghouse three-loop NSSS housed within a sub-atmospheric large-dry containment. A photo of the Surry site is shown in Figure 3. Each of the Surry reactors currently operates at a nominal power of 2546 MWth. Previous to the power uprate that occurred in 1995, the reactors operated at a nominal power of 2441 MWth.

Figure removed for copyright reasons

Figure 3. Surry Power Station.

The Surry MELCOR model used for the HBU Source Term analyses was based on that used in the NRC's plant security assessment for plants with large-dry containments [13]. A general description of the Surry 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. For the PWR plants that are currently operating near the regulatory limit of 62 GWd/t, fuel

assembly and core design information was 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 PWRs are provided in the following paragraphs.

To provide the ability 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 low-burnup (LBU) version MELCOR model was also created for each plant. These models were based on older (readily available) cycle information for assembly/bundle design, core loading, and power/burnup histories. Again, specifics are provided below.

MELCOR COR input representing fuel assembly geometry, fuel mass, cladding mass, control poison mass, grid support material and mass and other important physical characteristics were developed based on plant-specific data. The fuel designs that formed the basis for the MELCOR COR models for the two PWR plants are listed in Table 3. Assembly design data are not included in this report, as some data were obtained from company proprietary fuel design reports. However, much of the data are available in the relevant plant Final Safety Analysis Reports (FSARs) [14,15].

Table 3. Core Assembly Designs Represented in Pressurized-Water-Reactor MELCOR Models.

Plant	Core Model	Fuel Assembly Design
Sequoyah	LBU	Westinghouse 17x17
	HBU	Mark-BW 17x17
Surry	LBU	Westinghouse 15x15
	HBU	SIF 15x15

In addition to fuel assembly design information, core axial and radial power profiles, as well as end-of-cycle fission product mass inventories and decay heat information was needed. Radial core power profiles for HBU cores were based on plant- and cycle-specific nuclear design reports obtained from the licensees. For each PWR plant, three different recent cycles were examined to ensure that significant cycle-to-cycle variations were not observed. The Sequoyah HBU radial power profile was based on information for Unit 2 Cycles 12 and 13, and Unit 1 Cycle 14 [16,17,18]. The resulting MELCOR profile is shown in Figure 4. The Surry HBU radial power profile was based on information for Unit 2 Cycles 16 through 18 [19,20,21]. The resulting MELCOR profile is shown in Figure 5.

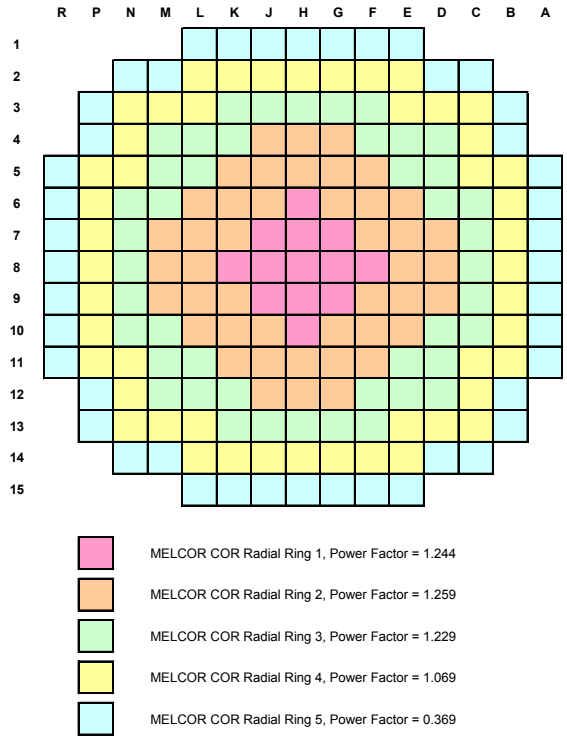

Figure 4. Radial Power Profile for a Sequoyah High-Burnup MELCOR Model.

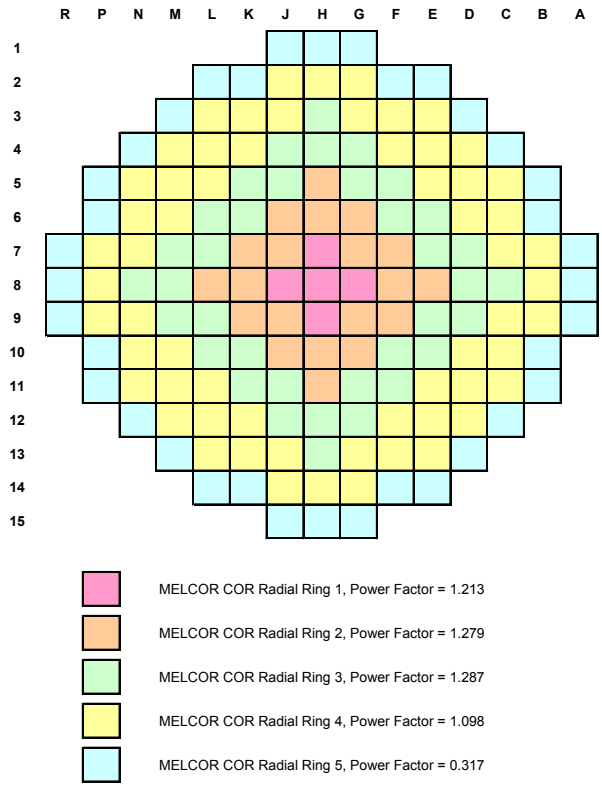

Figure 5. Radial Power Profile for a Surry High-Burnup MELCOR Model.

Radial power profiles for the LBU cores were not provided by the licensees. Instead, power profiles were obtained from LBU SCDAP/RELAP5 models developed at Idaho National Engineering Laboratory (INEL) for similar plants. These radial power profiles were adopted for use in the LBU MELCOR models. They are shown in Figure 6 and Figure 7 for Sequoyah and Surry, respectively.

Axial power profile data were not provided by the licensees, and it was indicated that the axial profiles are fairly consistent from plant to plant and cycle to cycle. Axial power profiles were obtained from SCDAP/RELAP5 models developed at INEL for similar plants, and these were compared to a generic PWR axial power profile obtained from NUREG-1754 [22]. Private communications with the licensees indicated that the axial profiles for PWRs did not vary significantly between LBU and HBU cores. In fact, the variation during a single cycle was thought to be as large as any difference between LBU and HBU cores. The power profiles used for the Sequoyah and Surry MELCOR models are shown in Figure 8.

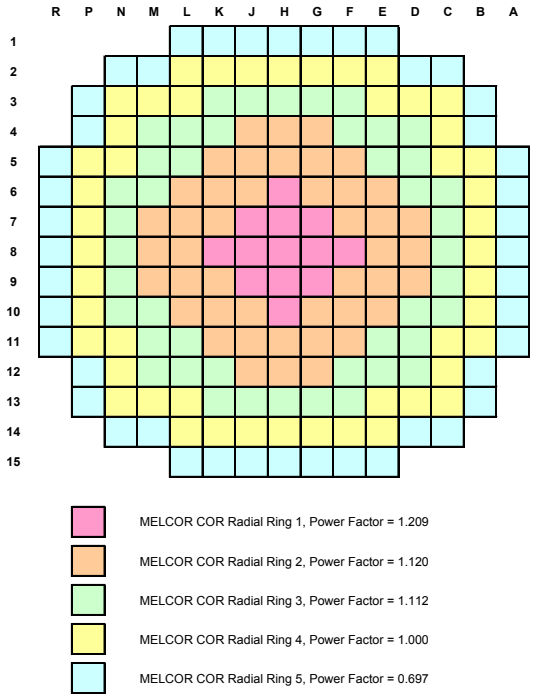

Figure 6. Radial Power Profile for a Surry Low-Burnup MELCOR Model.

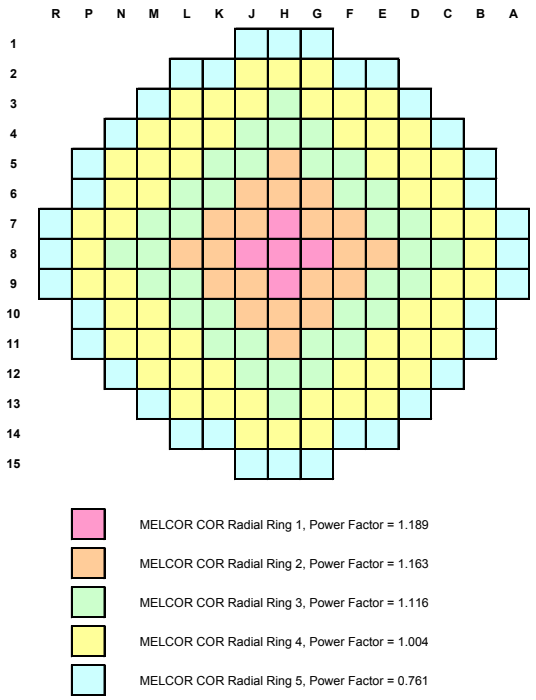

Figure 7. Radial Power Profile for a Surry Low-Burnup MELCOR Model.

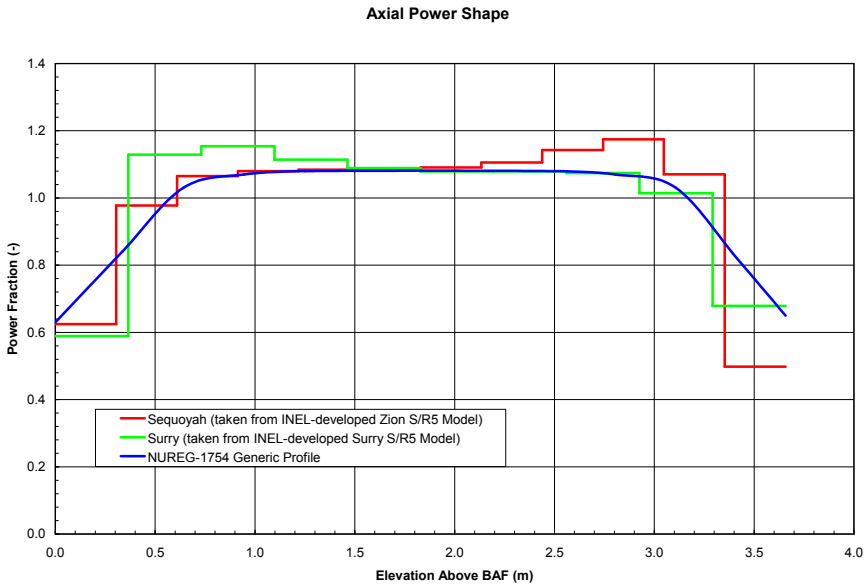

Figure 8. MELCOR Axial Power Profiles for Pressurized-Water Reactor Cores.

Information obtained from the licensees' documents used to create the power profiles, along with data from FSARs and other available information, was used as a basis to calculate fission product core inventories and associated decay heats. These calculations were performed by Oak Ridge National Laboratory (ORNL) for HBU and LBU cores for all four plants [23]. It should be

noted here that these 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 information that was based on plant-specific operational history. This is a significant advancement over old severe accident modeling techniques of using total core inventories and decay heat. The initial core fission product inventories used in the MELCOR model are shown in Table 4 and Table 5 for Sequoyah and Surry, respectively. Only whole core values are shown in these tables, although the spatial distributions from the ORNL report were used in the MELCOR input. Decay heat values for Sequoyah and Surry are listed in Table 6 and Table 7, respectively.

Comparison of the fission product inventory results show a significantly larger increase in the HBU inventory over the LBU inventory for Surry than for Sequoyah. While both HBU models were based on very recent cycles and were extrapolated (slightly) to a burnup of 59 GWd/t for the lead assembly, the burnups of the two LBU models are significantly different. The LBU core models represent a burnup of 24 GWd/t and 34 GWd/t for the lead assemblies in the Surry and Sequoyah models, respectively. This difference occurred for two reasons. First, there was limited detailed data readily available for early reactor cycles, so the ORIGEN calculations were performed based on the data available. Second, and more importantly, early cycles for the Surry reactor were generally run at lower burnups and a lower specific power than the early cycles for Sequoyah. Sequoyah specific powers were much more consistent over the life of the plant. This is further evidenced by the differences between the two plants when comparing the LBU time-dependent decay heat against the HBU values. The Surry decay heats are noticeably higher for the HBU core, while the Sequoyah values show little change from LBU to HBU. It is worth noting, however, that the HBU values for Sequoyah are approximately 5% higher in the long term than the LBU values. This difference is simply not observed when rounding to the nearest megawatt.

Table 4. Initial Core Inventories Calculated for Sequoyah.

Radionuclide Group	LBU Core (kg)	HBU Core (kg)	% Increase
Noble Gases	294.4	513.7	74%
Halogens	11.46	19.82	73%
Alkali Metals	164.0	288.9	76%
Tellurium	26.90	47.15	75%
Barium, Strontium Group	125.8	214.9	71%
Noble Metals	209.9	367.0	75%
Lanthanides	390.8	703.8	80%
Cerium Group	973.0	1444	48%

Table 5. Initial Core Inventories Calculated for Surry.

Radionuclide Group	LBU Core (kg)	HBU Core (kg)	% Increase
Noble Gases	186.3	410.1	120%
Halogens	7.4	15.7	112%
Alkali Metals	102.1	371.8	264%
Tellurium	17.0	37.4	120%
Barium, Strontium Group	78.8	169.1	115%
Noble Metals	131.9	145.4	10%

Lanthanides	242.9	561.4	131%
Cerium Group	624.8	1085	74%

Table 6. Initial Whole-Core Decay Heat Calculated for Sequoyah.

Time After Shutdown (hours)	LBU Core (MW)	HBU Core (MW)
0.0	221	217
0.0003	204	200
0.001	186	183
0.002	168	165
0.004	153	151
0.01	137	135
0.02	121	120
0.03	106	105
0.06	93	92
0.12	81	81
0.25	70	69
0.50	58	57
1	46	46
2	37	37
12	22	23
24	18	19
48	15	15
168	9	9

Table 7. Initial Whole-Core Decay Heat Calculated for Surry.

Time After Shutdown (hours)	LBU Core (MW)	HBU Core (MW)
0.0	152	179
0.0003	140	166
0.001	128	151
0.002	115	136
0.004	105	125
0.01	94	112
0.02	84	99
0.03	73	87
0.06	64	77
0.12	56	67
0.25	48	58
0.50	40	48
1	32	38
2	25	31
12	16	19
24	13	16
48	10	13
168	6	8

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 cesium release rate model that accounts for observed trends from the RT-6 test. The VERCORS RT-6 test [24,25,26] 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 9 shows the measured Cs release from the RT-6 experiment. These data are used to develop release rate parameters for a diffusional release model, as described in the following section.

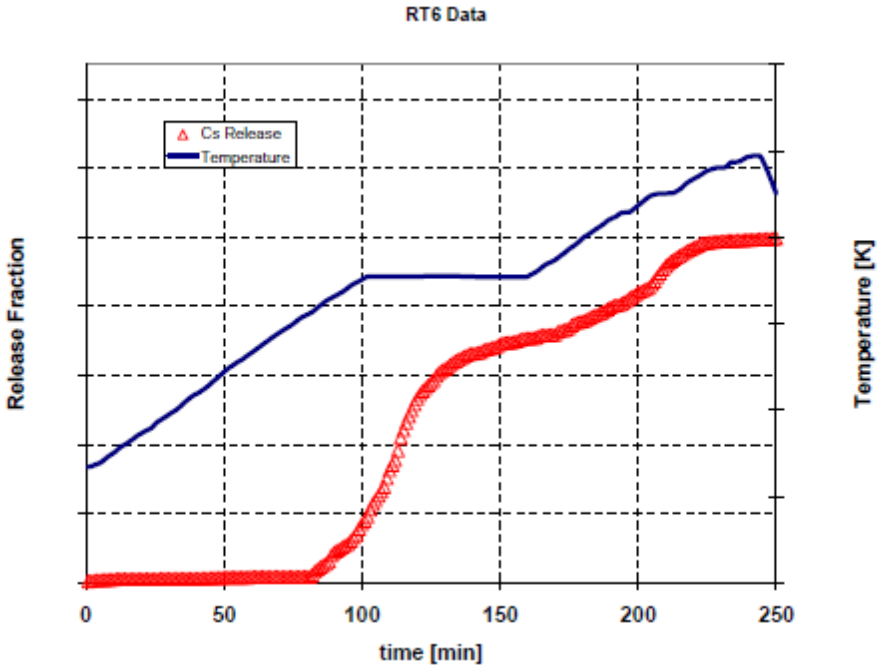

Figure 9. RT-6 Release of Cesium as a Function of Test Sample Temperature. (Units are suppressed to respect proprietary nature of the test results.)

2.4.1 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 overheated 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 species vapor pressure is considered before release to the local atmosphere. The effective diffusion coefficient for Cs in the fuel grain is given by

$$D = D_o \exp(-Q/RT) \quad \textbf{(1)}$$

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 burnup. The Cs release fraction at time *t* is calculated from an approximate solution of the diffusion equation for fuel grains of spherical geometry [27],

$$f = 6\sqrt{\frac{D't}{\pi}} - 3D't \quad for\ D't < 1/\pi^2 \quad \textbf{(2)}$$

$$f = 1 - \frac{6}{\pi^2}\exp(-\pi^2 D't) \quad for\ D't > 1/\pi^2 \quad \textbf{(3)}$$

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 [28]. In this process, Eqs. 4 and 5 are inverted to solve for the product Dt/a^2, as indicated below.

$$\frac{Dt}{a^2} = \frac{2}{\pi} - \frac{f}{3} - 2\sqrt{\frac{1}{\pi^2} - \frac{f}{3\pi}} \quad for\ f < 0.85 \quad \textbf{(4)}$$

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

where *f* is the release fraction.

Using the method described above (Eqs. 4 and 5), the experimental data may be cast in terms of the apparent instantaneous diffusion coefficient as shown in Figure 10. 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 10 and the release rate versus temperature measurements shown in Figure 9. The release prediction using the Booth formula for HBU fuel is shown compared to the measured RT-6 data in Figure 11 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 low-enriched uranium (LEU) fuel are summarized in Table 8.

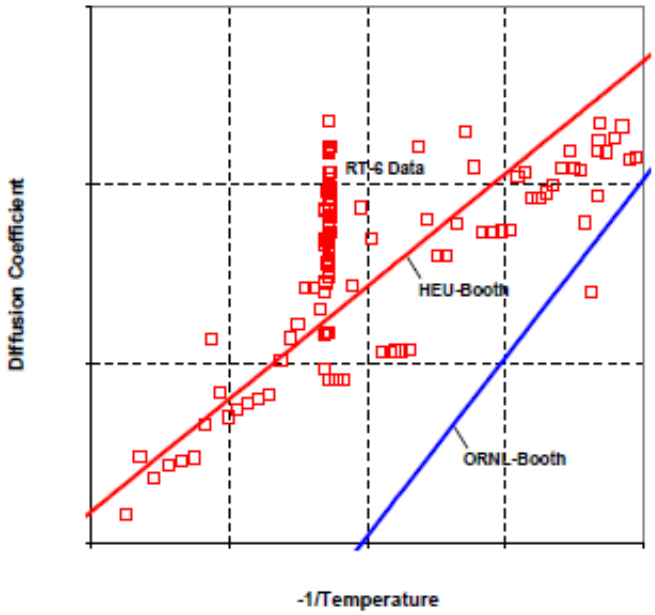

Figure 10. Instantaneous RT-6 Diffusion Coefficients and Booth Model Fits. (Units are suppressed to respect proprietary nature of the test results.)

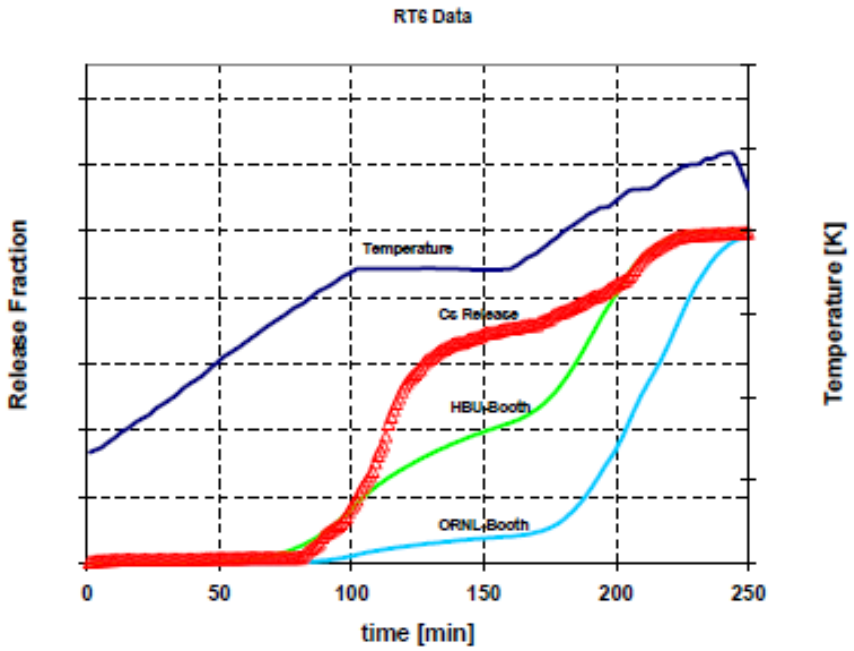

Figure 11. RT-6 Release Measurements Compared to Booth Model Predictions for High-Burnup and Low-Burnup Fuels. (Units are suppressed to respect proprietary nature of the test results.)

Table 8. Parameters for Diffusion Coefficient for High-Burnup and Low-Enriched Uranium Fuel.

	D_o [m^2/s]	Q [J/kg-mole]
LEU Fuel (ORNL-Booth)	1×10^{-6}	3.814×10^{5}
HBU Fuel (HBU-Booth)	2.3×10^{-9}	2.411×10^{5}
Grain Radius	6 μm	6μm

VERCORS tests involving HBU fuel will be reviewed for insights on melting point and other relevant issues [24,25,26]. Insights gained will be folded into this HBU source term development in future revisions to this report.

3 ASSUMPTIONS AND LIMITATIONS OF THE MELCOR ANALYSES

3.1 Scope Limitations

As mentioned several times herein, the mechanistic source term analyses performed using MELCOR does not explicitly include each risk-significant accident for all currently operating plants. Rather, the calculations performed are intended to be representative. Specifically, calculations are not performed for Combustion Engineering or Babcock & Wilcox plants. In addition, certain accident sequences, identified as having a small contribution to the core damage frequencies for the analyzed plants, were not considered (e.g., steam generator tube rupture [SGTR]). 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 assembly/bundle 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 an 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 25th 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 has been assumed here that HBU fuel properties that affect core degradation, such as fuel-clad interactions, fuel dissolution in residual metallic cladding and relocation temperature, are the

same as those of LBU fuel. It has further been assumed that the steam oxidation kinetics of the HBU cladding are adequately predicted by correlations derived for LBU Zircaloy-4 cladding. Available data seem to substantiate this assumption, at least at higher temperatures where the chemical kinetics of steam reaction with the cladding are rapid and most strongly affect fuel heating. As a consequence of these necessary assumptions (along with the similar decay heat profiles), core damage progression in LBU and HBU calculations are similar in most cases. Observed differences can be traced to differences in the decay heating rates. In particular, the HBU long-term decay heat (>24 hours) tends to be slightly higher in the HBU cores. While these differences are usually small (< 1 MW), the fractional difference in total decay heat is large enough to produce slightly faster long-term accident progression in the HBU cases. This is evidenced, primarily, by the timing of containment failure in cases where an early failure due to hydrogen combustion was not calculated.

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 options on HBU source terms [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 HBU fuel would involve "fuel foaming" rather than fuel candling as observed with fuel at lower burnup levels.

If future experimental work is performed, such as additional fuel melt experiments for HBU fuels or fuel melt experiments that include ZIRLO or M5 cladding, results from the work presented herein should be revisited with new insights in mind.

3.6 Modeling of Nonradioactive Aerosols

The effect of nonradioactive aerosol generation (e.g., control poison) 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. MELCOR models that model generation of nonradioactive aerosols from control poisons (Ag-In-Cd) have been applied in these calculations.

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 significantly different. MELCOR modeling improvement that would allow 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 direct containment heating [DCH] package) might be recommended. In

the calculations presented here, MELCOR distribution of fission product mass and decay heat was made according to the plant reference data fission product mass distribution. An informal sensitivity calculation was performed where the MELCOR distribution of fission products and decay heat was redistributed according to the reference plant decay heat distribution. Differences in accident progression and source term characteristics between these two calculations were small in comparison to differences between selected sequences. Therefore, this modeling limitation was determined to be acceptable.

3.8 Accident Progression Uncertainties

The MELCOR calculations performed for this work represent the current best practices in MELCOR severe accident modeling techniques, as well as best-estimate treatment of accident progression. No uncertainty analyses have been performed to determine whether there is a significant effect on results due to these uncertainties in accident progression. Key uncertainties that could affect release phase durations in particular may include mechanism for vessel lower head failure (i.e., penetration failure v. lower head creep); uncertainties in late-phase in-vessel melt progression, loop natural circulation phenomena, and resulting creep-rupture of PWR RCS structures; behavior of RCP seals under high-temperature conditions; and safety relief valve (SRV) behavior (e.g., stuck-open valves).

4 SELECTION OF REPRESENTATIVE ACCIDENT SEQUENCES

The effort to develop an alternative source term (AST) that is applicable for accidents involving HBU 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 HBU fuels [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. The accident sequences on which these source terms were based are listed in Table 9. A key for PWR accident sequence symbols is provided in Table 10.

Table 9. Pressurized-Water Reactor Accident Sequence Basis for Revised Source Term [6].

Plant	Sequence	Description
Surry (Westinghouse 3-loop PWR with large-dry, subatmospheric containment)	AG	Hot leg loss of coolant accident (LOCA), no containment sprays (CS), no fan coolers (FC)
	TMLB'	Loss of Offsite Power (LOOP), no Power Conversion System (PCS), no Auxiliary Feedwater (AFW)
	V	Interfacing systems LOCA (ISLOCA)
	S3B	Station blackout (SBO), reactor coolant pump (RCP) seal LOCA
	S2D-δ	Small-break LOCA (SBLOCA), no emergency core cooling system (ECCS), H_2 combustion
	S2D-β	SBLOCA, 6"-equivalent diameter hole in containment
Zion (Westinghouse 4-loop PWR with large-dry containment)	S2DCR	2"-equivalent diameter LOCA, no ECCS, no CS recirculation
	S2DCF1	RCP seal LOCA, no ECCS, no CS, no FC, early H_2 burn-induced containment failure (CF)
	S2DCF2	22DCF1, except late CF (overpressure or H_2 burn)
	TMLU	Transient, no power conversion system (PCS), no ECCS, no AFW, CF due to direct containment heating (DCH)
Oconee 3 (B&W PWR with large-dry containment)	TMLB'	SBO, no AFW
	S1DCF	3"-equivalent diameter LOCA, no AFW, no ECCS, no CS, no FC
Sequoyah (Westinghouse 4-loop PWR with ice condenser containment)	S3HF1	RCP seal LOCA, no ECCS, no CS recirculation, reactor cavity flooded
	S3HF2	S3HF1, hot leg creep-rupture before vessel failure (VF)
	3HF3	S3HF1, dry reactor cavity
	S3B	½"-equivalent diameter LOCA, SBO, no AFW
	TBA	SBO, hot leg creep-rupture before VF, H_2 burn-induced CF
	ACD	Hot leg LOCA, no ECCS, no CS
	S3B1	SBO, delayed RCP seal failure (4), turbine-driven AFW
	S3HF	RCP seal LOCA, no ECCS, no CS recirculation
	S3H	RCP seal LOCA, no ECCS recirculation

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 [29]. 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 11 presents identification of the risk significance, as discussed in NUREG-1560, for each of the accident sequences that provided the

basis for the NUREG-1465 PWR Source Term. Note that NUREG-1560 listed transients as risk-significant accidents for many plants. However, when failures of engineered safety features occur that may lead to core damage, the event progression characteristics of these accidents (and thus the estimated source term) are very similar to SBO accidents listed. Therefore, additional transient accidents are not added to the tables.

Table 10. Key to Pressurized-Water Reactor Accident Sequence Symbols [30].

Symbol	Description
A	Intermediate to large LOCA
B	Failure of electric power to engineered safety features
B'	Failure to recover either onsite or offsite electric power within 1.5 hours following an initiating transient which is a loss of offsite AC power
C	Failure of the containment spray injection system
D	Failure of the emergency core cooling injection system
F	Failure of the containment spray recirculation system
G	Failure of the containment heat removal system
H	Failure of the emergency core cooling recirculation system
K	Failure of the reactor protection system
L	Failure of the secondary system steam relief valves and the auxiliary feedwater system
M	Failure of the secondary system steam relief valves and the power conversion system
Q	Failure of the primary system safety relief valves to recluse after opening
R	Massive rupture of the reactor vessel
S1	A small LOCA with an equivalent diameter of about 2 to 6 inches
S2	A small LOCA with an equivalent diameter of about ½ to 2 inches
T	Transient event
V	LPIS check valve failure (containment bypass event)
α	Containment rupture due to a reactor vessel steam explosion
β	Containment failure resulting from inadequate isolation of containment openings and penetrations
γ	Containment failure due to hydrogen burning
δ	Containment failure due to overpressure
ε	Containment vessel melt-through

Many of the accident sequences listed in Table 11 were identified as important contributors to risk in the NUREG-1560 assessment. Table 12 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 LBU 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 12 for LBU fuel. This would provide a new baseline for source terms for LBU fuel, and would provide a point of comparison for the NUREG-1465 revised source term.

Table 11. Risk-Significance of NUREG-1465 Pressurized-Water Reactor Accident Sequences.

Plant	NUREG-1465 Sequence	Sequence Description	Risk Significance (NUREG-1560)	Other Comments
Surry	AG	Hot leg LOCA, no CS, no FC	Moderate	Large-break LOCAs currently thought to be minimally risk-significant. However, induced creep-rupture failure (e.g., during SBO) more important.
	TMLB'	LOOP, no PCS, no AFW	High	
	V	ISLOCA	Low	IPE identification of potential bypass path led to operator training to minimize risk.
	S3B	SBO, RCP seal LOCA	High	
	S2D-δ	SBLOCA, no ECCS, H_2 combustion	High	
	S2D-β	SBLOCA, 6" initial CF	Not discussed	
Zion	S2DCR	2" LOCA, no ECCS, no CS recirc	High	
	S2DCF1	RCP seal LOCA, no ECCS, no CS, no FC, early CF	Low	Early failure unlikely for large dry containment. Vessel pressure reduced by LOCA, prevents HPME failure at VF.
	S2DCF2	S2DCF1, late CF	High	Transient causes loss of pump seal cooling.
	TMLU	Transient, no PCS, no ECCS, no AFW, CF @ VF	High	Current thinking is CF @ VF less likely. Primary system likely depressurized before VF.
Oconee 3	TMLB'	SBO, no AFW	High	
	S1DCF	3" LOCA, no AFW, no ECCS, no CS, no FC	Moderate	
Sequoyah	S3HF1	RCP seal LOCA, no ECCS, no CS recirc, cavity flooded	High	Transient causes loss of pump seal cooling.
	S3HF2	S3HF1, hot leg creep-rupture before VF	Not discussed	Transient causes loss of pump seal cooling.
	S3HF3	S3HF1, dry cavity	Not discussed	Some large-dry and ice condenser designs limit flow of water to cavity.
	S3B	½"-equivalent diameter LOCA, SBO, no AFW	Low	
	TBA	SBO, hot leg creep-rupture before VF, H_2 burn-induced CF	High	
	ACD	Hot leg LOCA, no ECCS, no CS	Moderate	
	S3B1	SBO, delayed RCP seal failure (4), turbine-driven AFW	High	TD-AFW likely fails upon battery depletion.
	S3HF	RCP seal LOCA, no ECCS, no CS recirc	High	Transient causes loss of pump seal cooling.
	S3H	RCP seal LOCA, no ECCS recirc	High	Transient causes loss of pump seal cooling.

Table 12. MELCOR Calculation Matrix for Pressurized-Water Reactor High Burnup Accident Sequences.

Case	MELCOR Model	Init.	ECCS	AFW	CS	FC	Cavity Flood	RCP Seal LOCA	Ctmt Failure
1A	Surry	SBO	No	No	No	No			
1B	Surry	SBLOCA	No	Yes	Yes	Yes			
1C	Surry	LLOCA	Inj.	Yes	Inj.	No			
1D	Surry	SBO	No	No	No	No		No	
1E	Surry	ISLOCA	Yes	Yes	Yes	Yes			Bypass
1F	Surry	SBLOCA	No	Yes	Yes	Yes			@ VF
2A	Zion / IP3	SBLOCA	No	Yes	Inj.	Yes			
2B	Zion / IP3	RCP Seal	No	Yes	No	No		1 Loop	Late
2C	Zion / IP3	Trans	No	No	Yes	Yes			@ VF
2D	Zion / IP3	RCP Seal	No	Yes	No	No		1 Loop	Early
3A	TMI	SBO	No	No	No	No			
3B	TMI	MLOCA	No	No	No	No			
4A	Sequoyah	RCP Seal	No	Yes	Inj.	Yes	Yes	1 Loop	
4B	Sequoyah	RCP Seal	No	Yes	Inj.	Yes	No	1 Loop	
4C	Sequoyah	RCP Seal	Inj.	Yes	Yes	Yes		1 Loop	
4D	Sequoyah	SBO	No	Steam	No	No			
4E	Sequoyah	SBO	No	No	No	No			Early
4F	Sequoyah	LLOCA	No	Yes	No	Yes			
4G	Sequoyah	SBLOCA	No	No	No	No			

Notes: Blank cells indicate that no specification will be made and MELCOR will calculate results.
IP3 = Indian Point 3 Nuclear Power Plant.
Inj. = System available in injection mode only.

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

4. Repeat MELCOR calculations for entire suite of accident sequences listed in Table 12 for HBU 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.

Resource limitations did not allow development of HBU and LBU core models for each of the MELCOR models identified in Table 11 and Table 12. However, it was judged that a reasonably representative set of accident analyses could be performed by covering the sequences listed for Surry and Sequoyah. This is primarily due to similarities in accident sequences across plant types. Therefore, with two exceptions, this study examines the accidents listed for Surry and Sequoyah in Table 12. First, Case 1C was one of the accident sequences that formed the basis for the NUREG-1465 Source Term. However, while NUREG-1560 indicates “moderate” risk significance for this sequence, core damage is not expected to occur unless containment steam overpressure results in ECCS failure. Previous analyses of the AG severe accident suggest that core damage would not occur for at least 60 hours after the initial LOCA event. Therefore, this accident is screened from consideration. Case 1E (ISLOCA) was identified as risk-significant for

Surry. The AST of NUREG-1465 addresses containment response to postulated accidents and the resulting containment source term. The ISLOCA event, by definition, results in containment bypass. Protection from containment bypass events is considered separately in NRC regulations; therefore, no MELCOR calculation was performed for Case 1E. The technical basis for the development of an HBU supplement to the NUREG-1465 Source Term is provided by MELCOR calculations for Cases 1A through 1D, Case 1F, and Cases 4A through 4G.

For Sequoyah, an LBU and HBU 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 fuel type and burnup level. For Surry LBU calculations, Cases 1A and 1B, an LBU core model was developed based on plant operation data from Cycles 1 through 3. Specific power histories for these plant "startup" cycles were not typical of most LBU cycles; this can be seen in the lower decay heat values developed for the Surry LBU model. Thus, accidents simulated with the Surry LBU model tended to be more slowly developing than one would realistically expect.

5 ACCIDENT SOURCE TERMS CALCULATED USING MELCOR

Based on the accident sequence selection process described in Section 4, the following accidents were considered in the development of the HBU supplement to the NUREG-1465 Source Term for PWRs:

- Surry, SBO, no AFW (Case 1A)
- Surry, SLOCA, no ECCS, AFW operates (Case 1B)
- Surry, LLOCA, ECCS in injection model only, no containment heat removal (Case 1C)
- Surry, SBO, no AFW and no RCP seal failure (Case 1D)
- Surry, SLOCA, no ECCS, AFW operates, early containment failure (Case 1F)
- Sequoyah, RCP seal LOCA, no ECCS, AFW operates, with cavity flooding (Case 4A, for extension to other plants)
- Sequoyah, RCP seal LOCA, no ECCS, AFW operates (Case 4B)
- Sequoyah, RCP seal LOCA, ECCS in injection mode only, AFW operates (Case 4C)
- Sequoyah, SBO, AFW for approximately one hour (Case 4D)
- Sequoyah, SBO, no AFW (Case 4E)
- Sequoyah, SBLOCA (Case 4F)
- Sequoyah, LLOCA (Case 4G)

While the accidents considered are not exhaustive, they are considered representative of risk-significant accidents for PWRs. For example, the Sequoyah accidents considered account for 96% of the total core damage frequency (CDF) described in the IPE [31]. Only two accident sequences identified in the Sequoyah IPE as having "appreciable frequency" were not modeled here. The first was a steam generator tube rupture, representing approximately 1% of the CDF. The other was an ISLOCA, representing 0.1% of the CDF. Note that bypass sequences, such as SGTR, were identified as being low-risk contributors in NUREG-1560. Most other sequences that were smaller contributors to the CDF that are not treated here were slight variations of the sequences listed above, such as a RCP seal LOCA with failure of AFW. Results for release timing and magnitude for 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 5.1.6 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 12.

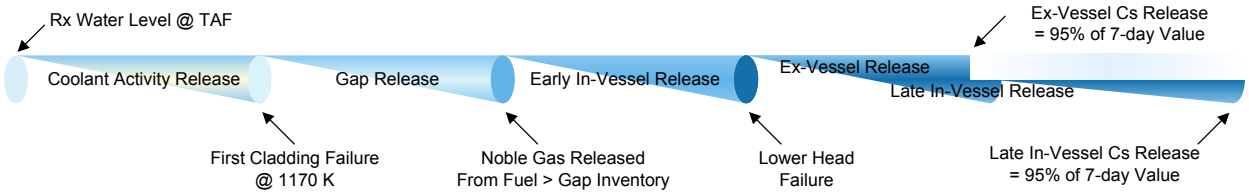

Figure 12. Release Phase Timing Definitions – Tie to Calculated MELCOR Results.

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 13. Since a shortened period for the onset of release would be conservative, the 25th percentile of the inferred distribution was taken to be representative.

Table 13. Time of Onset of Radionuclide Release for Pressurized-Water Reactors.

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Sequoyah	4A	RCP Seal LOCA, AFW, Cavity Flooded	5.8833	5.8667
	4B	RCP Seal LOCA, AFW, Cavity Dry	5.8834	5.8667
	4C	RCP Seal LOCA, AFW, ECCS Inj	9.1333	9.1000
	4D	SBO, TD-AFW (approx. 1 hr)	8.9000	8.8000
	4E	SBO, no AFW	2.1167	2.1000
	4F	LLOCA	0.0003	0.0002
	4G	SLOCA	1.7167	1.6667
Surry	1A	SBO, no AFW, Induced RCP Seal Fail	1.9833	1.4833
	1B	SLOCA, no ECCS, Late CF	2.4500	2.6334
	1C	LLOCA, ECCS in injection model only	0.0333	0.0273
	1D	SBO, no AFW, no RCP Seal Failure	1.9667	1.5000
	1F	SLOCA, no ECCS, CF@VF	2.4500	2.6334

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 14.

Table 14. Duration of Coolant Release Phase for Pressurized-Water Reactors.

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Sequoyah	4A	RCP Seal LOCA, AFW, Cavity Flooded	1.1500	1.1500
	4B	RCP Seal LOCA, AFW, Cavity Dry	1.1500	1.1333
	4C	RCP Seal LOCA, AFW, ECCS Inj	1.5167	1.4000
	4D	SBO, TD-AFW (approx. 1 hr)	1.4500	1.3333
	4E	SBO, no AFW	1.0000	0.9833
	4F	LLOCA	0.2164	0.1998
	4G	SLOCA	0.7667	0.7666
Surry	1A	SBO, no AFW, Induced RCP Seal Fail	0.7667	0.4667
	1B	SLOCA, no ECCS, Late CF	0.7000	0.6667
	1C	LLOCA, ECCS in injection model only	8.9667	6.8227
	1D	SBO, no AFW, no RCP Seal Failure	0.7500	0.4667
	1F	SLOCA, no ECCS, CF@VF	0.7000	0.6667

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 in-vessel release has always been difficult to draw. 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 12, 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 15. 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 15. Duration of Gap Release Phase for Pressurized-Water Reactors.

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Sequoyah	4A	RCP Seal LOCA, AFW, Cavity Flooded	1.0667	0.3833
	4B	RCP Seal LOCA, AFW, Cavity Dry	1.0500	0.3667
	4C	RCP Seal LOCA, AFW, ECCS Inj	1.2833	0.7167
	4D	SBO, TD-AFW (approx. 1 hr)	0.3333	0.2167
	4E	SBO, no AFW	0.3500	0.2833
	4F	LLOCA	0.0500	0.0833
	4G	SLOCA	0.2500	0.1667
Surry	1A	SBO, no AFW, Induced RCP Seal Fail	0.3833	0.2500
	1B	SLOCA, no ECCS, Late CF	0.2500	0.1833
	1C	LLOCA, ECCS in injection model only	0.2167	0.1333
	1D	SBO, no AFW, no RCP Seal Failure	0.2833	0.2167
	1F	SLOCA, no ECCS, CF@VF	0.2500	0.1833

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 reactor 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 16.

Table 16. Duration of In-Vessel Release Phase for Pressurized-Water Reactors.

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Sequoyah	4A	RCP Seal LOCA, AFW, Cavity Flooded	25.9754	17.7746
	4B	RCP Seal LOCA, AFW, Cavity Dry	6.0572	7.1124
	4C	RCP Seal LOCA, AFW, ECCS Inj	7.2166	4.9333
	4D	SBO, TD-AFW (approx. 1 hr)	3.4680	3.0713
	4E	SBO, no AFW	3.8462	2.5793
	4F	LLOCA	2.1785	2.6519
	4G	SLOCA	4.4334	2.4182
Surry	1A	SBO, no AFW, Induced RCP Seal Fail	7.1899	3.9626
	1B	SLOCA, no ECCS, Late CF	7.9061	18.1849
	1C	LLOCA, ECCS in injection model only	3.6706	3.4890
	1D	SBO, no AFW, no RCP Seal Failure	4.9265	3.9033
	1F	SLOCA, no ECCS, CF@VF	7.9061	18.1849

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 nonradioactive aerosols. The release of radionuclides proceeds at a very slow rate once the residual metallic Zr 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 Cs in the core debris has been released to the containment atmosphere. Ex-vessel release durations are shown in Table 17.

Table 17. Duration of Ex-Vessel Release Phase for Pressurized-Water Reactors.

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Sequoyah	4A	RCP Seal LOCA, AFW, Cavity Flooded	0.0913	0.0744
	4B	RCP Seal LOCA, AFW, Cavity Dry	4.0762	4.9877
	4C	RCP Seal LOCA, AFW, ECCS Inj	5.2715	5.9441
	4D	SBO, TD-AFW (approx. 1 hr)	2.1653	1.8287
	4E	SBO, no AFW	0.7427	2.6873
	4F	LLOCA	3.8382	1.6880
	4G	SLOCA	2.4499	3.7652
Surry	1A	SBO, no AFW, Induced RCP Seal Fail	138.0100	10.6041
	1B	SLOCA, no ECCS, Late CF	31.1941	9.3320
	1C	LLOCA, ECCS in injection model only	83.2794	6.9611
	1D	SBO, no AFW, no RCP Seal Failure	140.0736	4.4801
	1F	SLOCA, no ECCS, CF@VF	0.0106	8.8320

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. 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 18.

Table 18. Duration of Late In-Vessel Release Phase for Pressurized-Water Reactors.

Reactor	Case	Accident	LBU Core (hr)	HBU Core (hr)
Sequoyah	4A	RCP Seal LOCA, AFW, Cavity Flooded	48.4248	69.6588
	4B	RCP Seal LOCA, AFW, Cavity Dry	144.0262	152.0210
	4C	RCP Seal LOCA, AFW, ECCS Inj	143.8502	127.0167
	4D	SBO, TD-AFW (approx. 1 hr)	142.8486	139.9120
	4E	SBO, no AFW	154.9093	158.2207
	4F	LLOCA	136.0548	141.5648
	4G	SLOCA	152.5000	159.1485
Surry	1A	SBO, no AFW, Induced RCP Seal Fail	153.0101	153.8374
	1B	SLOCA, no ECCS, Late CF	3.0106	120.3319
	1C	LLOCA, ECCS in injection model only	35.2794	151.6943
	1D	SBO, no AFW, no RCP Seal Failure	156.0735	157.5801
	1F	SLOCA, no ECCS, CF@VF	0.0115	129.8319

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 19 and Table 20 for accident sequences involving LBU and HBU cores, respectively. Results are shown only for the four most volatile radionuclide groups. Meaningful results are not calculated for the other radionuclide groups. These results emphasize that the source term developed here is for release into the containment.

5.2.2 In-Vessel Release

Calculated in-vessel release fractions for the various groups of radionuclides are shown in Table 21 and Table 22 for accident sequences involving LBU and HBU cores, respectively.

5.2.3 Ex-Vessel Release

Ex-vessel release fractions are shown in Table 23 and Table 24 for accident sequences involving a LBU and HBU core, respectively.

5.2.4 Late In-Vessel Release

Late in-vessel release fractions are shown in Table 25 and Table 26 for accident sequences involving a LBU and HBU core, respectively.

Table 19. Gap Release Fractions for Pressurized-Water Reactors with a Low-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	1.9270E-02	2.0181E-03	6.4842E-04	3.3165E-03	3.2994E-03	1.4181E-07	4.8967E-05	2.2724E-12	2.2741E-12
	4B	1.9331E-02	2.8207E-03	9.1800E-04	4.5041E-03	4.6349E-03	1.2553E-07	4.3354E-05	2.0076E-12	2.0090E-12
	4C	1.9493E-02	2.1633E-03	6.9856E-04	3.4065E-03	3.5359E-03	1.2660E-07	4.3765E-05	2.0226E-12	2.0236E-12
	4D	2.3732E-02	6.7822E-03	2.2305E-03	1.0457E-02	1.0905E-02	8.3697E-08	2.8749E-05	1.3330E-12	1.3342E-12
	4E	2.3711E-02	8.1864E-03	2.7108E-03	1.2439E-02	1.3176E-02	3.9865E-08	1.3703E-05	6.3456E-13	6.3518E-13
	4F	3.7297E-02	1.8561E-02	6.1144E-03	2.9700E-02	3.0530E-02	8.1062E-08	2.8384E-05	1.2990E-12	1.2990E-12
	4G	1.7838E-02	6.7685E-03	2.2318E-03	1.0422E-02	1.0822E-02	6.8428E-08	2.3493E-05	1.0887E-12	1.0898E-12
Surry	1A	2.1593E-02	6.8036E-03	1.3693E-03	6.3199E-03	6.6356E-03	1.9546E-08	1.9358E-06	3.1135E-13	3.1157E-13
	1B	2.4811E-02	5.6292E-03	1.1279E-03	5.3973E-03	5.4966E-03	5.2507E-08	5.2349E-06	8.4159E-13	8.4117E-13
	1C	4.1192E-02	3.3169E-02	6.2489E-03	3.0992E-02	3.2333E-02	2.0158E-06	5.9398E-04	1.0004E-10	9.9827E-11
	1D	7.1446E-03	3.0730E-04	5.6540E-05	5.9950E-04	2.8790E-04	1.3069E-09	1.2993E-07	2.0811E-14	2.0827E-14
	1F	2.4811E-02	5.6292E-03	1.1279E-03	5.3973E-03	5.4966E-03	5.2507E-08	5.2349E-06	8.4159E-13	8.4117E-13

Table 20. Gap Release Fractions for Pressurized-Water Reactors with a High-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	7.6388E-03	1.1635E-03	3.4961E-04	1.9499E-03	1.8050E-03	2.7534E-07	9.5714E-05	4.3404E-12	4.3587E-12
	4B	7.6110E-03	1.8111E-03	5.7258E-04	2.8400E-03	2.9162E-03	2.3170E-07	8.0974E-05	3.6546E-12	3.6692E-12
	4C	9.0762E-03	1.5106E-03	4.5882E-04	2.4450E-03	2.3672E-03	3.2827E-07	1.1436E-04	5.1690E-12	5.1929E-12
	4D	1.6667E-02	4.5789E-03	1.4324E-03	6.8957E-03	7.2254E-03	7.3469E-07	2.5615E-04	1.1540E-11	1.1604E-11
	4E	2.1324E-02	6.3481E-03	1.9583E-03	9.7476E-03	9.9347E-03	1.2047E-06	4.2046E-04	1.8947E-11	1.9044E-11
	4F	3.8719E-02	2.0403E-02	5.0635E-03	2.8396E-02	2.8380E-02	1.0635E-05	4.8076E-03	2.2069E-10	2.2136E-10
	4G	1.3726E-02	5.0235E-03	1.5765E-03	7.4895E-03	7.9805E-03	7.7962E-07	2.7178E-04	1.2243E-11	1.2312E-11
Surry	1A	1.6211E-02	2.6999E-03	5.0931E-04	2.5528E-03	2.6362E-03	4.6322E-07	4.5807E-05	7.3753E-12	7.3754E-12
	1B	2.1375E-02	2.7496E-03	5.2958E-04	2.5849E-03	2.7363E-03	3.3155E-07	3.2851E-05	5.3004E-12	5.2916E-12
	1C	4.4089E-02	3.3592E-02	5.4599E-03	3.2121E-02	3.1321E-02	9.0766E-06	1.6846E-03	2.8355E-10	2.8200E-10
	1D	1.0963E-02	8.4164E-05	1.3669E-05	1.7440E-04	7.2220E-05	2.4880E-08	2.4562E-06	3.9575E-13	3.9592E-13
	1F	2.1375E-02	2.7496E-03	5.2958E-04	2.5849E-03	2.7363E-03	3.3155E-07	3.2851E-05	5.3004E-12	5.2916E-12

Table 21. In-Vessel Release Fractions for Pressurized-Water Reactors with a Low-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	8.7064E-01	7.7200E-02	1.2363E-03	9.5542E-02	8.5996E-02	6.6404E-04	6.6477E-02	2.0509E-08	2.0167E-08
	4B	6.5498E-01	1.2580E-01	2.4033E-03	1.2128E-01	1.3449E-01	5.0978E-04	1.0821E-01	1.2048E-08	1.2072E-08
	4C	6.6716E-01	1.8959E-01	3.0484E-03	1.8672E-01	1.8691E-01	1.3394E-03	1.6336E-01	2.8028E-08	2.8065E-08
	4D	7.4982E-01	8.3134E-02	1.0634E-03	1.0910E-01	8.8139E-02	3.3891E-04	7.1201E-02	9.6757E-09	9.9534E-09
	4E	8.0032E-01	1.9467E-01	2.4995E-03	5.8573E-01	2.3547E-01	6.6365E-03	1.5568E-01	1.2503E-07	1.2504E-07
	4F	9.2762E-01	6.4112E-01	3.5698E-03	6.2756E-01	6.5168E-01	1.1358E-02	5.7570E-01	2.2171E-07	2.2581E-07
	4G	7.8835E-01	2.1093E-01	2.3227E-03	5.3846E-01	2.5036E-01	5.6064E-03	1.7323E-01	1.2722E-07	1.2517E-07
Surry	1A	8.9855E-01	1.9170E-01	1.9854E-03	4.4713E-01	2.1971E-01	3.0209E-03	4.3395E-02	8.2657E-08	8.2783E-08
	1B	8.3540E-01	3.4885E-01	5.9087E-03	3.9021E-01	3.7563E-01	9.1139E-03	8.5531E-02	1.9454E-07	1.9487E-07
	1C	9.2316E-01	5.6628E-01	1.1088E-02	7.7008E-01	7.3110E-01	3.1887E-02	1.4703E-01	8.5673E-07	8.6976E-07
	1D	9.3938E-01	5.7164E-01	4.2902E-03	6.9225E-01	6.9093E-01	1.3621E-02	1.4603E-01	2.7030E-07	2.7072E-07
	1F	8.3540E-01	3.4885E-01	5.9087E-03	3.9021E-01	3.7563E-01	9.1139E-03	8.5531E-02	1.9454E-07	1.9487E-07

Table 22. In-Vessel Release Fractions for Pressurized-Water Reactors with a High-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	8.9930E-01	8.5353E-02	1.6170E-03	1.4005E-01	1.0106E-01	1.3761E-03	7.2820E-02	2.9518E-08	2.9602E-08
	4B	9.5045E-01	3.7553E-01	5.6692E-03	4.5464E-01	4.7890E-01	4.3247E-03	3.3311E-01	1.0169E-07	1.0176E-07
	4C	5.6379E-01	1.9231E-02	5.0382E-04	2.2238E-02	1.7818E-02	1.2846E-04	1.6220E-02	2.6480E-09	2.6673E-09
	4D	9.3796E-01	6.4757E-02	1.1232E-03	1.0351E-01	8.6343E-02	6.4108E-04	5.5616E-02	2.6914E-08	2.7068E-08
	4E	9.2544E-01	6.1139E-02	1.2439E-03	1.5152E-01	8.9158E-02	4.0209E-04	4.9548E-02	1.8048E-08	1.8324E-08
	4F	9.4479E-01	6.4743E-01	9.0880E-03	6.7345E-01	6.8637E-01	2.4586E-02	5.8870E-01	5.7963E-07	5.7547E-07
	4G	8.9548E-01	7.4589E-02	1.3393E-03	1.4086E-01	1.0390E-01	7.9192E-04	6.3114E-02	3.2390E-08	3.2513E-08
Surry	1A	9.7804E-01	2.5578E-01	7.1380E-03	4.7446E-01	3.7174E-01	2.8011E-02	6.7738E-02	5.4610E-07	5.5039E-07
	1B	9.4960E-01	2.8018E-01	5.2408E-03	4.4462E-01	3.4942E-01	7.1163E-03	6.6402E-02	1.6128E-07	1.6178E-07
	1C	9.4471E-01	6.9093E-01	1.4303E-02	8.1586E-01	7.9418E-01	3.5068E-02	1.8234E-01	8.9773E-07	8.9887E-07
	1D	9.7672E-01	4.1416E-01	7.3361E-03	5.2816E-01	5.3538E-01	2.0319E-02	1.0728E-01	4.5869E-07	4.5546E-07
	1F	9.4960E-01	2.8018E-01	5.2408E-03	4.4462E-01	3.4942E-01	7.1163E-03	6.6402E-02	1.6128E-07	1.6178E-07

Table 23. Ex-Vessel Release Fractions for Pressurized-Water Reactors with a Low-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	4.0198E-04	6.7002E-06	5.0286E-05	1.0497E-06	3.3813E-07	1.5899E-13	6.2529E-06	2.3210E-06	8.1418E-08
	4B	8.9312E-02	9.3580E-02	1.9135E-03	7.5214E-02	1.4695E-02	2.8857E-12	8.4804E-02	2.8791E-05	9.7656E-06
	4C	2.7652E-01	2.9002E-01	8.0336E-04	1.1236E-01	2.3026E-02	1.1293E-12	2.6786E-01	1.3555E-05	6.1838E-06
	4D	1.9429E-01	2.0068E-01	1.5039E-03	1.2340E-01	2.3804E-02	1.8581E-12	1.8345E-01	2.2024E-05	8.1111E-06
	4E	1.6224E-01	1.7384E-01	1.0775E-01	1.8812E-01	8.8057E-02	2.5265E-09	1.5552E-01	1.4156E-02	4.3807E-04
	4F	2.2538E-03	2.9571E-03	4.5044E-02	3.9016E-03	2.4756E-03	1.1833E-09	2.6162E-03	3.5064E-03	1.3481E-04
	4G	1.7060E-01	1.8194E-01	3.6318E-02	1.7999E-01	8.4084E-02	4.5419E-09	1.6347E-01	8.7412E-03	3.2878E-04
Surry	1A	0.0000E+00	6.5894E-03	1.5998E-03	8.7431E-02	1.1327E-01	2.5752E-06	7.2972E-01	5.9278E-05	4.5160E-05
	1B	1.3708E-01	1.1115E-02	1.4357E-03	1.4748E-01	5.2192E-02	6.8872E-12	5.7837E-09	3.6078E-05	2.4485E-05
	1C	1.5353E-02	6.8474E-04	4.4692E-04	9.0855E-03	4.6747E-03	8.5732E-13	7.5178E-08	1.4868E-05	1.2685E-05
	1D	3.5881E-02	4.0710E-03	1.4864E-03	5.4016E-02	7.2123E-02	1.4294E-06	7.5195E-01	4.3246E-05	3.6123E-05
	1F	6.8400E-03	8.8179E-06	4.9341E-06	1.1700E-04	8.4839E-06	5.1630E-16	1.5377E-16	7.6314E-08	3.4542E-08

Table 24. Ex-Vessel Release Fractions for Pressurized-Water Reactors with a High-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	8.5192E-03	3.9909E-05	1.9779E-05	1.5998E-06	4.7693E-07	3.2879E-12	3.8075E-05	2.9139E-06	8.2730E-08
	4B	2.0733E-02	2.6503E-02	3.8746E-03	2.3772E-02	4.1909E-03	2.0722E-12	2.4340E-02	5.8200E-05	9.9632E-06
	4C	1.1082E-01	1.2464E-01	1.4627E-03	5.0301E-02	8.0479E-03	4.6639E-13	1.1702E-01	1.8023E-05	5.3237E-06
	4D	2.2948E-02	3.0237E-02	8.9244E-03	3.4739E-02	9.2629E-03	1.2433E-11	2.7453E-02	1.8249E-04	1.5813E-05
	4E	3.2449E-02	4.4721E-02	9.1346E-03	4.4197E-02	9.6682E-03	3.5711E-11	4.0902E-02	4.0402E-04	1.7454E-05
	4F	9.5391E-04	2.0343E-03	5.0489E-01	3.5530E-03	4.2485E-03	1.8279E-06	1.7965E-03	4.2212E-01	2.0791E-02
	4G	4.6548E-02	6.1656E-02	3.6453E-03	2.3921E-02	4.8846E-03	3.4311E-12	5.7930E-02	7.9710E-05	5.2440E-06
Surry	1A	2.8717E-03	3.9992E-03	4.4402E-03	4.8531E-03	1.5361E-03	9.7651E-11	1.0269E-03	5.4327E-04	2.1866E-05
	1B	3.5056E-03	1.0495E-02	3.3898E-03	2.4198E-03	2.7179E-04	9.0821E-12	2.9001E-03	2.4436E-04	8.3278E-06
	1C	7.0611E-04	7.7865E-04	2.8320E-03	2.3634E-04	3.6817E-05	1.5381E-09	2.1404E-04	5.6059E-04	1.9814E-05
	1D	7.2006E-03	1.2254E-02	2.2574E-03	9.6888E-03	2.1329E-03	2.3138E-11	3.2497E-03	2.1853E-04	1.1991E-05
	1F	3.4982E-03	1.0712E-02	3.8148E-03	4.7586E-03	6.3904E-04	9.6442E-12	2.9147E-03	2.5291E-04	1.2676E-05

Table 25. Late In-Vessel Release Fractions for Pressurized-Water Reactors with a Low-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	0.0000E+00	2.0550E-02	*	3.2949E-04	1.4325E-02	0.0000E+00	1.7790E-02	1.5970E-03	2.6272E-02
	4B	0.0000E+00	*	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	3.4902E-01	0.0000E+00	0.000E+00
	4C	0.0000E+00	*	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	4.3181E-01	0.0000E+00	0.000E+00
	4D	0.0000E+00	3.8827E-02	*	5.1142E-01	2.2442E-01	0.0000E+00	1.9387E-01	0.0000E+00	2.8619E-02
	4E	0.0000E+00	1.9950E-02	0.0000E+00	1.3522E-01	2.2566E-01	0.0000E+00	1.5787E-01	0.0000E+00	5.5186E-03
	4F	0.0000E+00	*	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	2.4624E-01	0.0000E+00	0.000E+00
	4G	0.0000E+00	7.4652E-02	0.0000E+00	1.7721E-01	2.2562E-01	0.0000E+00	2.2097E-01	0.0000E+00	6.4748E-03
Surry	1A	0.0000E+00	7.6338E-02	8.3035E-01	3.4041E-01	0.0000E+00	0.0000E+00	1.0676E-02	1.2768E-03	2.8156E-02
	1B	0.0000E+00	7.1335E-02	5.4251E-01	1.8708E-01	1.5358E-01	5.1635E-03	1.4624E-02	6.5290E-04	2.4153E-02
	1C	0.0000E+00	9.0775E-03	0.0000E+00	1.0981E-01	3.2706E-02	1.6507E-03	0.0000E+00	0.0000E+00	0.0000E+00
	1D	0.0000E+00	1.6401E-02	0.0000E+00	1.0611E-02	0.0000E+00	1.9284E-02	0.0000E+00	0.0000E+00	1.9692E-03
	1F	0.0000E+00	3.4163E-04	0.0000E+00	1.0799E-03	1.0138E-03	1.5013E-03	2.0860E-04	0.0000E+00	2.8475E-06

*: Table entry was removed since the reported value was either above 1.0 or below 0.0. Such a release fraction is non-physical.

Table 26. Late In-Vessel Release Fractions for Pressurized-Water Reactors with a High-Burnup Core.

Plant	Case	Noble Gases [Xe]	Alkali Metals [Cs]	Alkaline Earths [Ba]	Halogens [I]	Chalcogens [Te]	Platinoids [Ru]	Early Transition Elements [Mo]	Tetravalents [Ce]	Trivalents [La]
Sequoyah	4A	0.0000E+00	4.6075E-02	*	2.9040E-01	2.3439E-01	0.0000E+00	3.2330E-02	1.5171E-03	2.4039E-02
	4B	0.0000E+00	1.5266E-03	*	2.1961E-02	0.0000E+00	0.0000E+00	2.4685E-02	7.3829E-04	2.7549E-02
	4C	0.0000E+00	4.5419E-03	*	0.0000E+00	0.0000E+00	3.1222E-01	2.8152E-01	6.0235E-04	3.3309E-02
	4D	0.0000E+00	1.2923E-01	0.0000E+00	7.8253E-01	7.1777E-01	0.0000E+00	1.1076E-01	0.0000E+00	2.1901E-03
	4E	0.0000E+00	1.3407E-01	0.0000E+00	6.8719E-01	6.5972E-01	0.0000E+00	1.2849E-01	0.0000E+00	3.1186E-03
	4F	2.7889E-03	1.7012E-02	0.0000E+00	8.1982E-02	5.1088E-02	0.0000E+00	1.0771E-02	0.0000E+00	0.0000E+00
	4G	0.0000E+00	2.6784E-01	0.0000E+00	7.0502E-01	6.4832E-01	0.0000E+00	2.7773E-01	0.0000E+00	3.3934E-03
Surry	1A	0.0000E+00	1.3145E-01	0.0000E+00	4.5718E-01	8.4053E-02	2.4659E-03	1.7186E-02	0.0000E+00	0.0000E+00
	1B	1.9602E-02	1.5860E-03	9.2874E-01	7.2001E-03	0.0000E+00	3.7693E-03	2.9568E-03	1.3299E-03	2.6825E-02
	1C	0.0000E+00	1.4620E-02	0.0000E+00	4.3672E-02	3.9915E-02	5.4285E-03	1.5727E-03	0.0000E+00	0.0000E+00
	1D	0.0000E+00	1.1130E-01	0.0000E+00	2.3223E-01	6.9931E-02	8.5047E-03	2.1977E-02	0.0000E+00	0.0000E+00
	1F	2.2248E-02	9.7629E-03	9.1372E-01	1.0361E-02	8.4906E-03	0.0000E+00	3.8185E-03	1.3660E-03	2.6609E-02

*: Table entry was removed since the reported value was either above 1.0 or below 0.0. Such a release fraction is non-physical.

6 DISCUSSION AND SUMMARY

The following sections display the tabular data for LBU and HBU release predictions presented in the previous section, in a ranked order in the form of fraction of observations that are less than or equal to the value on the x-axis. If the observations were actually random samples from a distribution of equally probable values (x-axis), then the following figures would approximate the cumulative distribution function for the population of accidents. We have of course restricted our population of accidents to those that are more risk-significant as opposed to simply more likely, so the distribution sampled, so to speak, is a subset of the full range on potential severe accidents. Nonetheless, within this restricted population, we have populated the selection with those risk significant accidents that are considered to be more likely. So, the display of results as an approximation of the cumulative distribution provide us with a means of characterizing the tendencies of the results in terms of median trends and span of variations. The distributions implied by this portrayal of the data are of course dependent on how fairly we have populated the collection of scenarios analyzed. For comparative purposes, the NUREG-1465 values are shown on the "distributions." A subsequent report will explore the use of order statistics to explore more fully the trends implied by these observations.

6.1 Timing and Duration of NUREG-1465 Phases

Shown in Figure 13 are the results for the predicted onset of gap release for the cases analyzed for both LBU and HBU fuel. Recall that the NUREG-1465 gap release phase was assumed to start essentially at the start of the transient, based on the expected trend of the design base LOCA. Only one sequence was calculated for a LLOCA, which is identifiable in Figure 13 at the far left of the x-axis, where approximately 90% of the observations indicated gap release start time greater than 30 seconds.

Depending on the true likelihood of a LLOCA, this 5% likelihood that start of gap release is less than or equal to 30 seconds may be overestimated by our implied weighting of the population with the LLOCA in contrast with other sequences, arguably an artifact of our small sample of the true distribution. Eighty percent of our observations produced start of gap release time greater than 1 hr, 25% between 1 hr and 2 hr, and 55% greater than 2 hr.

Figure 14 shows the distribution of predicted gap release durations as defined in this study. Recall that the gap release in this study was declared to be over when the net released fraction of fission product exceeded the amount considered to reside in the fuel rod gap (~3 to 5%). As defined, it is an artificial quantity since the reactor core does not respond coherently owing to large spatial (radial) variations in thermal response. The duration of the gap release phase in actuality defines the start of the early in-vessel release phase as the observed increase beyond 5% of released fission products is due to thermally driven release from rapidly heating fuel at a time when other fuel assemblies in the core have not yet started "their" gap release phase. Hence, almost all observations produced gap release durations less than the NUREG-1465 value of 30 minutes, even though the NUREG-1465 estimate of ½ hour for all fuel assemblies to experience gap release might actually be realistic. The very definition of a gap release phase is clearly problematic in its use with various regulatory applications.

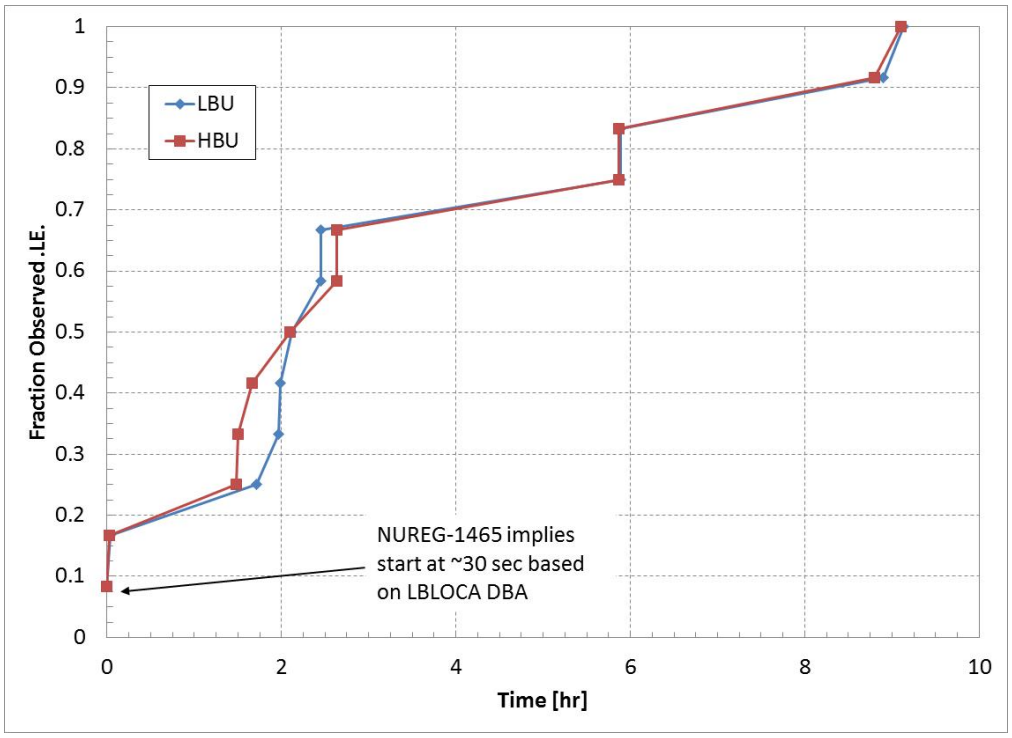

Figure 13. Onset of "Gap" Release from MELCOR Calculations.

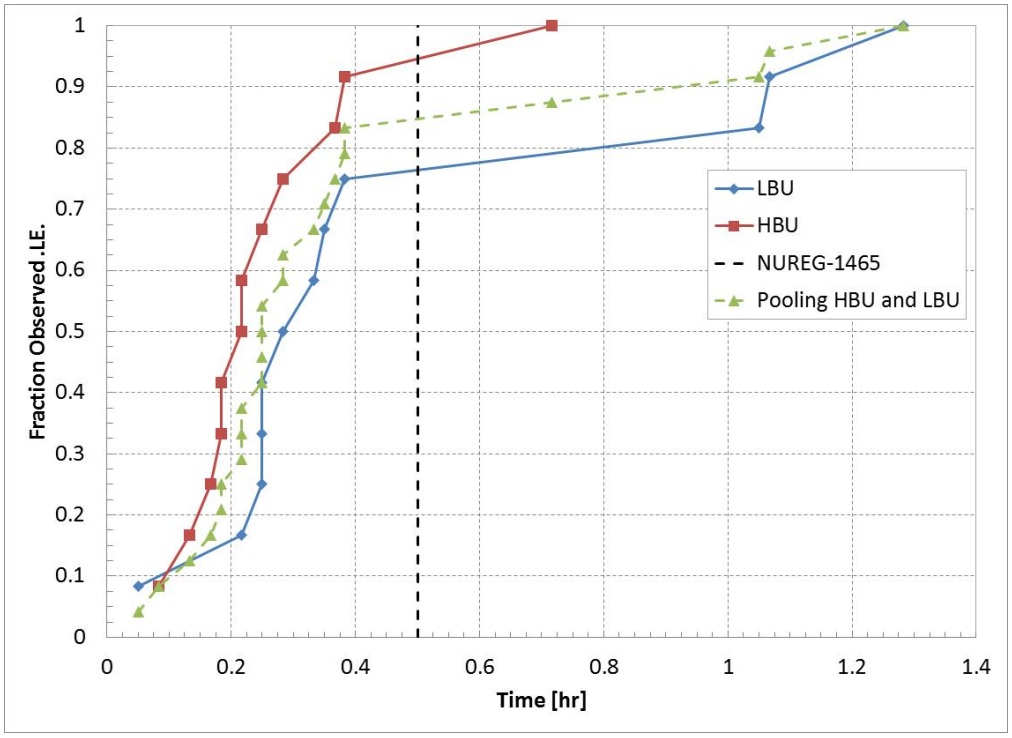

Figure 14. Duration of Gap Release Phase as Defined for MELCOR Analyses Compared with NUREG-1465.

The duration of the in-vessel release phase in the MELCOR analyses was defined to be the span of time between end of "gap release" (as defined in this study) and vessel lower head failure. The distribution of results is shown in Figure 15 and all observations indicated in-vessel release durations (by this definition) exceeded the NUREG-1465 value of 1.3 hr by at least a factor of 2. Fifty percent of the observations were in the 2- to 3-hr range and 50% were greater than about 3 hr.

The duration of the in-vessel release phase is sensitive to factors affecting the lower head failure time, which include, for example, mitigative water injection to the vessel, prior RCS depressurization through hot leg failure, or ex-vessel cooling by cavity flooding. Notably, there would appear to be an effect of fuel burnup on the duration of the in-vessel release duration with HBU tending to reduce the phase duration – the median value for HBU is about 4 hr compared to a median value of about 5 hr for LBU fuel. This would seem to be consistent with the greater decay heat level associated with the HBU fuel, leading to faster water boil-off and greater heat loads to the lower head. While the HBU and LBU results seem to be different in a statistically significant way, clearly the larger difference is between both the HBU and LBU results and the NUREG-1465 prescription. This difference (NUREG-1465 and HBU/LBU) may have implications on regulatory applications of the AST as applied to new reactor design certifications and current reactor license amendments (e.g., main stream isolation valve leakage requirements). Note also that the AST is also applied to non-regulatory applications such as the RASCAL emergency response tool and therefore may imply incorrect trends assumed for severe accident behavior in this context.

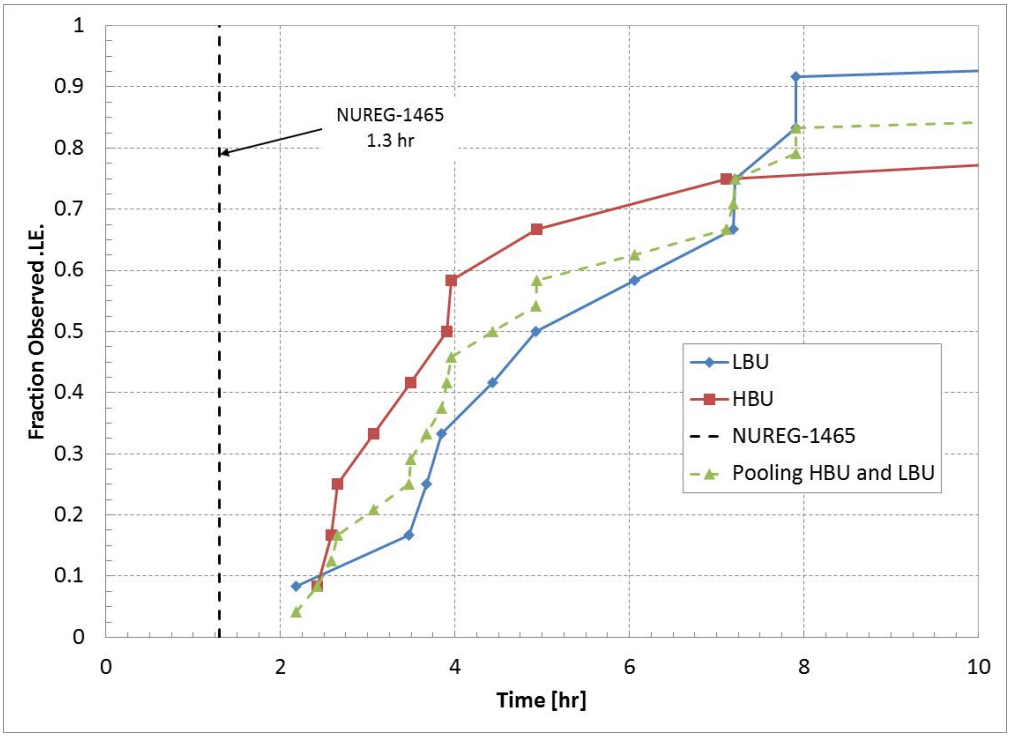

Figure 15. Duration of In-Vessel Release Phase as Defined for MELCOR Analyses Compared with NUREG-1465.

Results for the ex-vessel phase duration are shown in Figure 16. One case examined produced a near zero ex-vessel duration owing to the cavity being flooded, thereby cooling the debris and terminating the release phase. About 30% of the observations were in the neighborhood of the NUREG-1465 ex-vessel duration of 2 hr, and half of the observations were in excess of about 4 hr, some extended to 10 hr or more. The wide range of plausible ex-vessel durations rendered by the MELCOR calculations makes it difficult to specify a characteristic value; however, in general it can be noted that regulatory applications generally have not made use of the ex-vessel release phase duration. Finally, there seem to be no appreciable differences between HBU and LBU in the duration of the ex-vessel release phase.

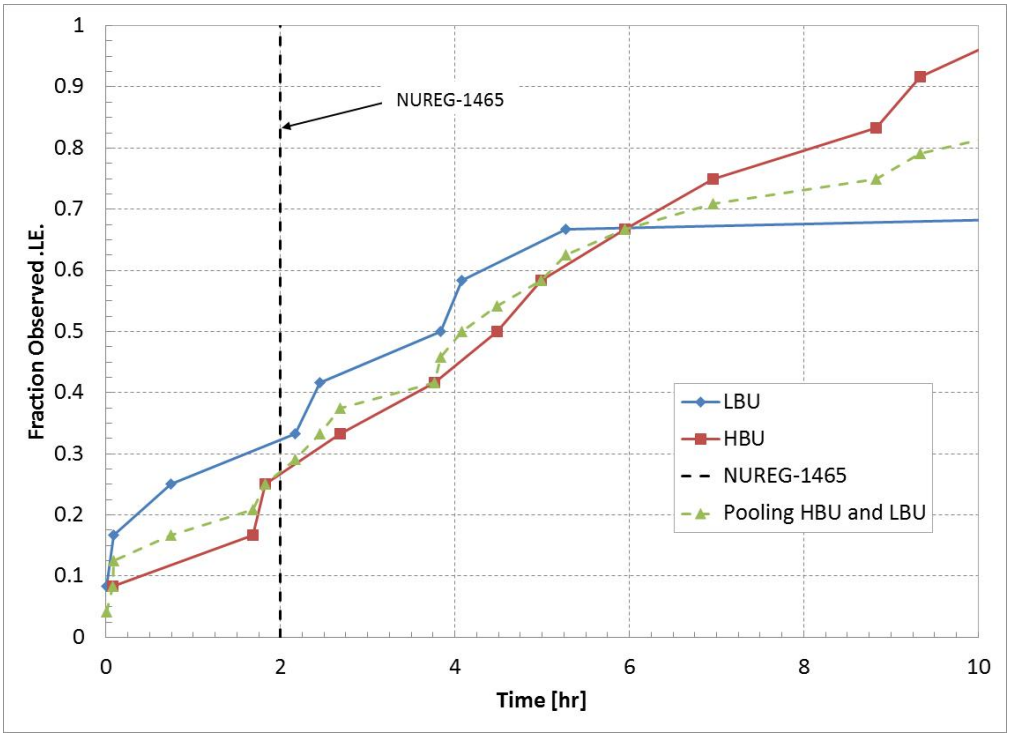

Figure 16. Duration of Ex-Vessel Release Phase as Defined for MELCOR Analyses Compared with NUREG-1465.

Shown in Figure 17 is the duration of the late in-vessel release phase. In general, late in-vessel release periods are long whenever significant RCS deposition has occurred. Factors acting against this tendency would be large LOCA events where deposition in RCS is not favored or arrested accidents where release in general is low. When RCS deposition has been significant, the re-vaporization phase can be significantly protracted, owing partly to the lower volatility of the Cs molybdate forms. When re-vaporization from the RCS is significant, the duration of the late in-vessel release period in general is significantly longer than the NUREG-1465 estimate of 10 hr.

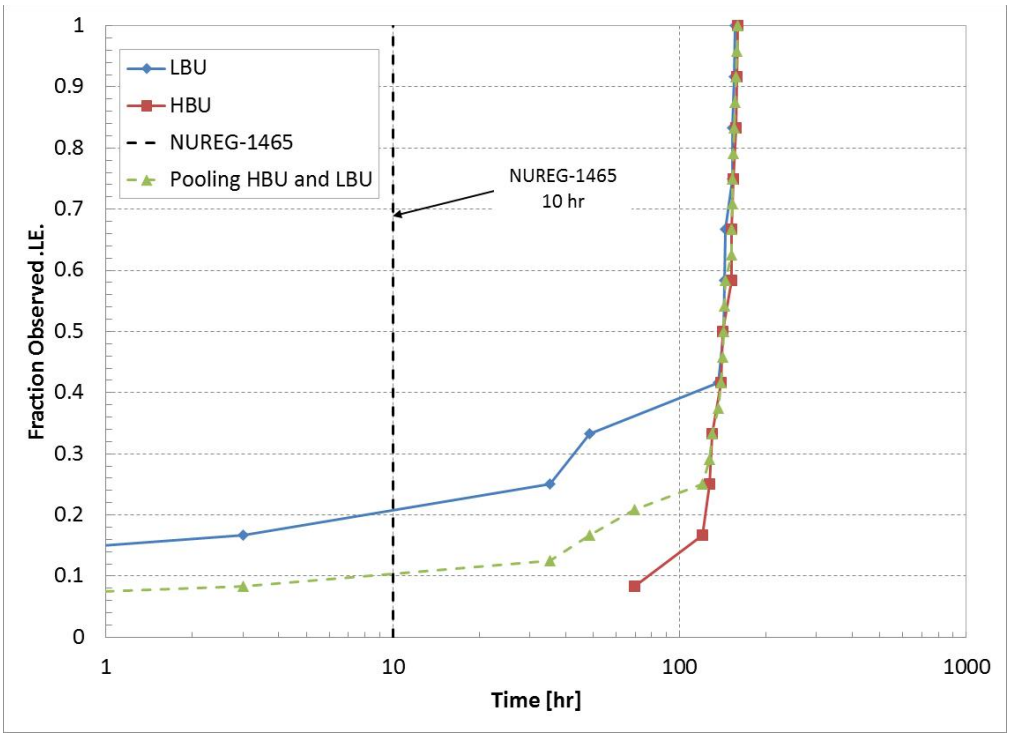

Figure 17. Duration of Late In-Vessel Release Phase as Defined for MELCOR Analyses Compared with NUREG-1465.

6.2 Release Fractions during NUREG-1465 Accident Phases

The following section reviews the fractional release tendencies for iodine and Cs during the various phases of the NUREG-1465 Source Term. We forego any discussion of the gap release phase owing to its simple nature and begin with the early in-vessel release of iodine, shown in Figure 18. In general the release of iodine (CsI) during the early in-vessel is calculated to be about half that prescribed in NUREG-1465. Forty percent of the observations were less than or on the order of 20% release, compared to the NUREG-1465 value of 35% release to containment.

The Cs release to the containment during this phase is about the same as iodine, and shows similar trends as seen in Figure 19. The lower fractions released to the containment for Cs are attributable to the greater retention in the RCS owing to the lower volatility of the dominant speciation of Cs, namely Cs_2MoO_4. Some additional clarification of this behavior is found by inspection of Figure 20 and Figure 21 for one of the sequences examined, showing the distribution of released CsI and Cs_2MoO_4 among the vessel and RCS, containment, and environment. Note that release of Cs from the fuel is relatively complete but the distribution of the release shows significant retention within the reactor vessel and the RCS. The effect of the differing volatility of CsI versus Cs_2MoO_4 is clearly seen in the large increase in the containment CsI at vessel breach (Figure 20), releasing the more volatile CsI, in contrast to the general sequestering of deposited Cs_2MoO_4 within the vessel and RCS (Figure 21).

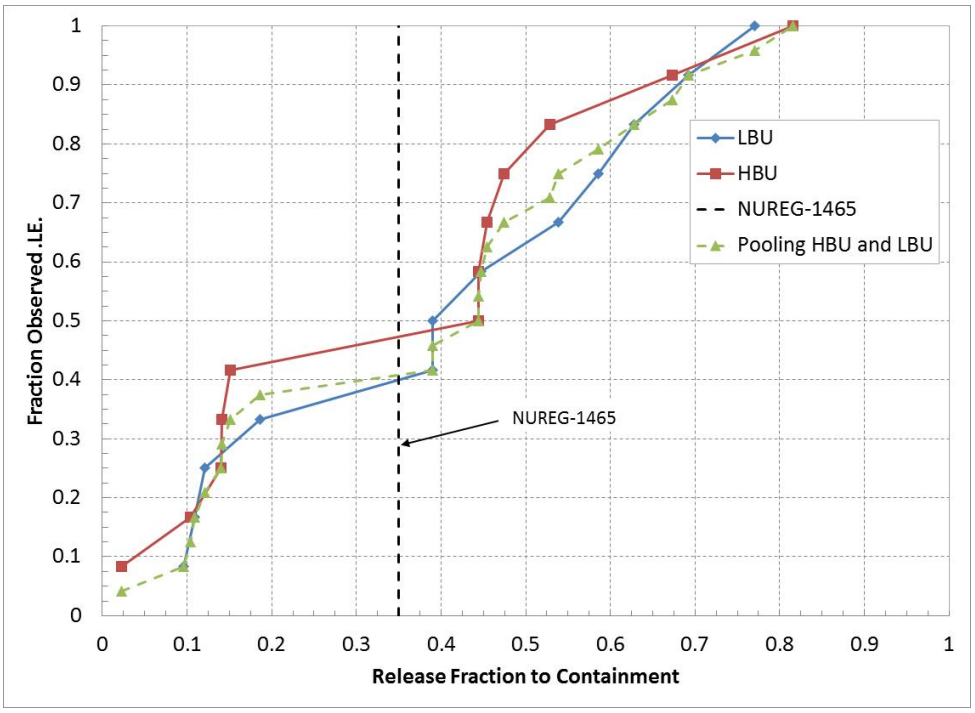

Figure 18. Calculated Release to Containment of In-Vessel Release Phase Iodine for MELCOR Analyses Compared with NUREG-1465. Constitutes principally iodine associated with CsI.

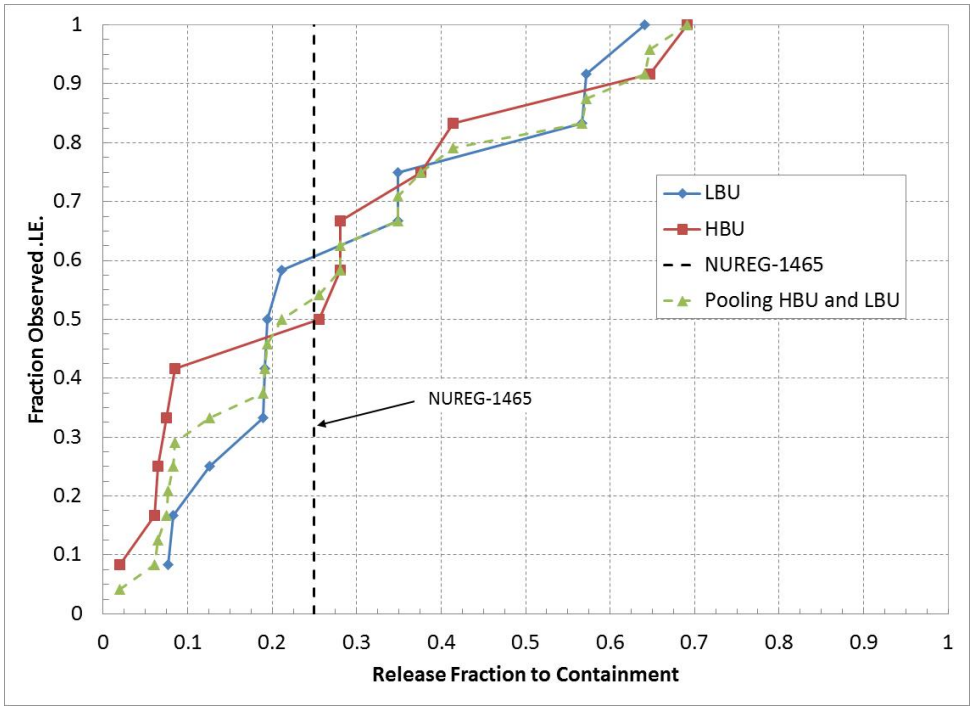

Figure 19. Calculated Release to Containment of In-Vessel Release Phase Cesium for MELCOR Analyses Compared with NUREG-1465. Constitutes Cs associated with CsI, Cs_2MoO_4 and CsOH.

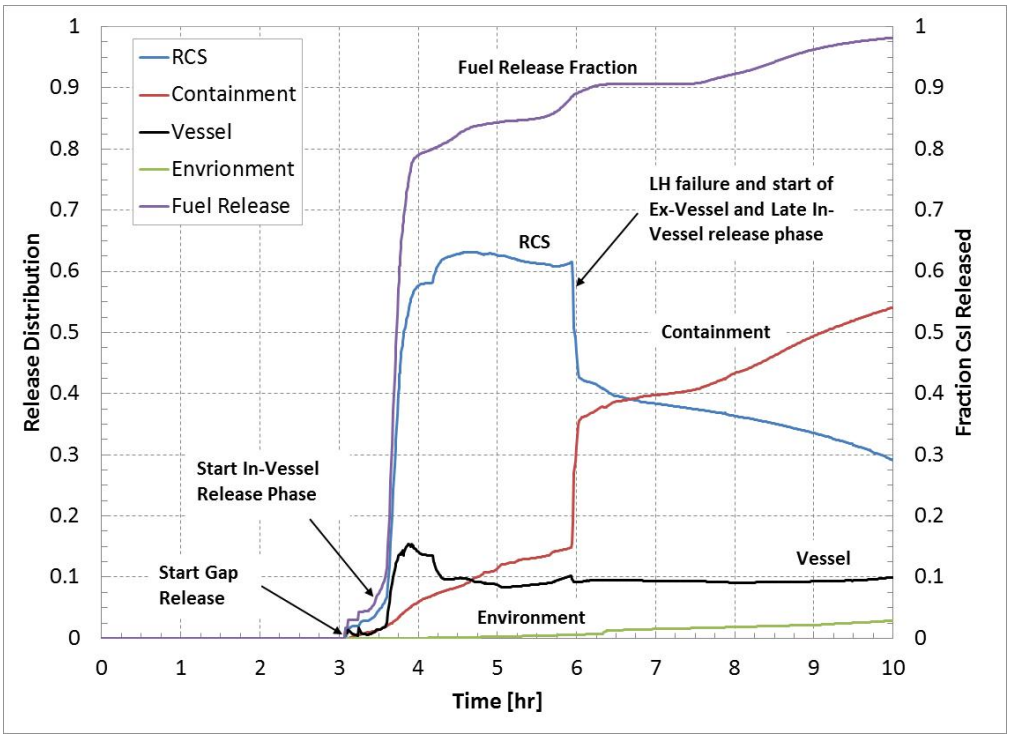

Figure 20. Sequoyah SBO (No AFW): Distribution of Released CsI.

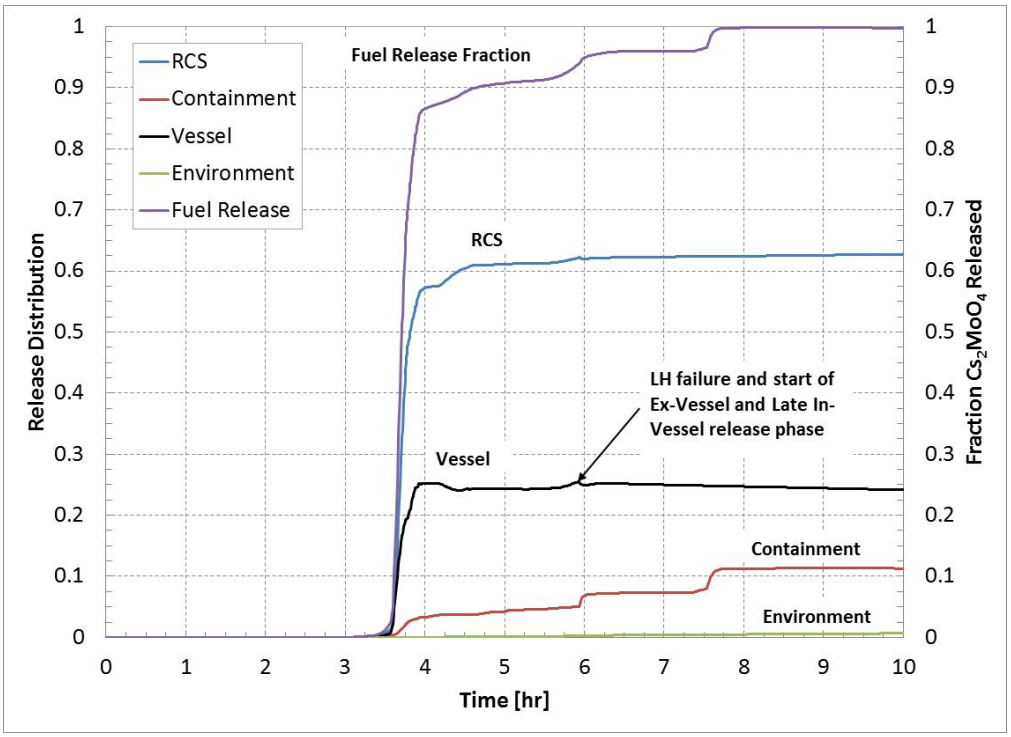

Figure 21. Sequoyah SBO (No AFW): Distribution of Released Cs_2MoO_4.

High release of the volatile Cs and iodine species from the fuel combined with fairly high retention of these species within the vessel and RCS result in lower fractions available for release during the ex-vessel release period. These trends are evident in Figure 22 and Figure 23 showing release to containment of iodine (CsI) and Cs (Cs_2MoO_4/CsOH) respectively. For both iodine and Cs releases during the ex-vessel release phase, calculated releases to containment are generally less than the NUREG-1465 release fraction by about a factor of 2, more or less.

Ongoing simultaneously with the ex-vessel release phase is the late in-vessel release phase, which arises from late releases from fuel assemblies still retained in the reactor vessel after lower head failure and re-vaporization of fission products that have deposited within the vessel internals and the RCS. Trends for iodine (CsI) and Cs (Cs_2MoO_4) for late in-vessel release to the containment are shown in Figure 24 and Figure 25, respectively. Again the differing volatility of CsI and Cs_2MoO_4 are evident. For iodine behavior the late in-vessel release fractions are significantly greater than the NUREG 1465 fraction. This is due to the greater calculated RCS deposition during the early in-vessel phase (not fully credited in NUREG-1465) and its subsequent re-vaporization during the late in-vessel release period. In contrast, Cs_2MoO_4 is not as easily re-vaporized from the RCS, hence the late in-vessel release fraction for Cs is considerably less than for iodine, and generally in fair agreement with the NUREG-1465 prescription.

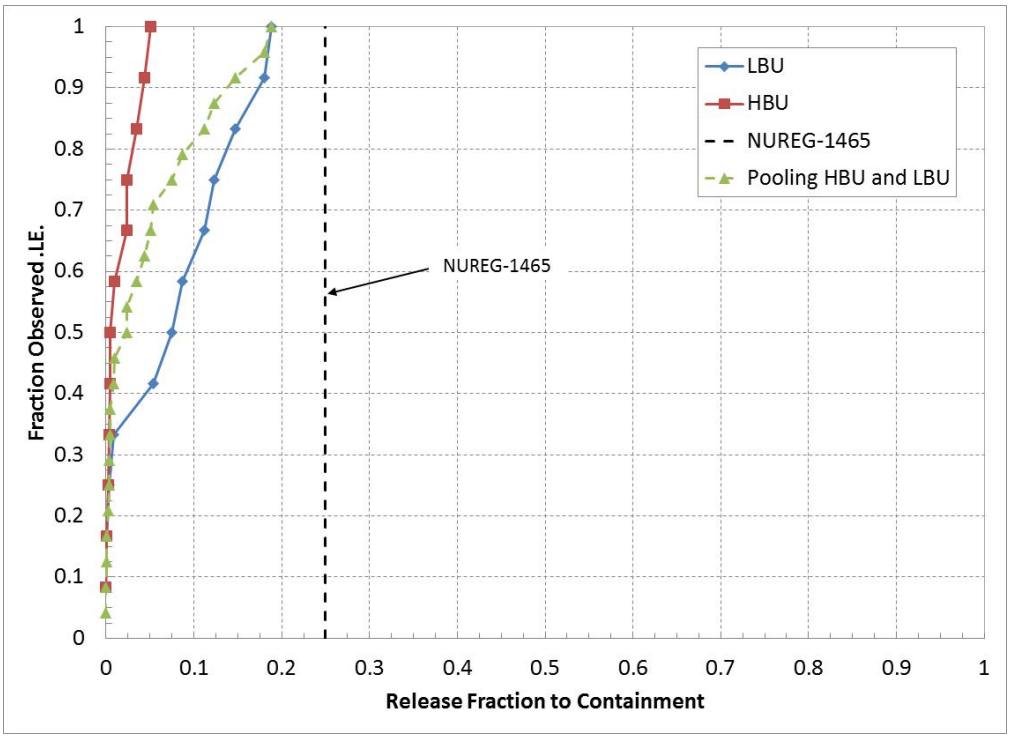

Figure 22. Calculated Release to Containment of Ex-Vessel Release Phase Iodine for MELCOR Analyses Compared with NUREG-1465.

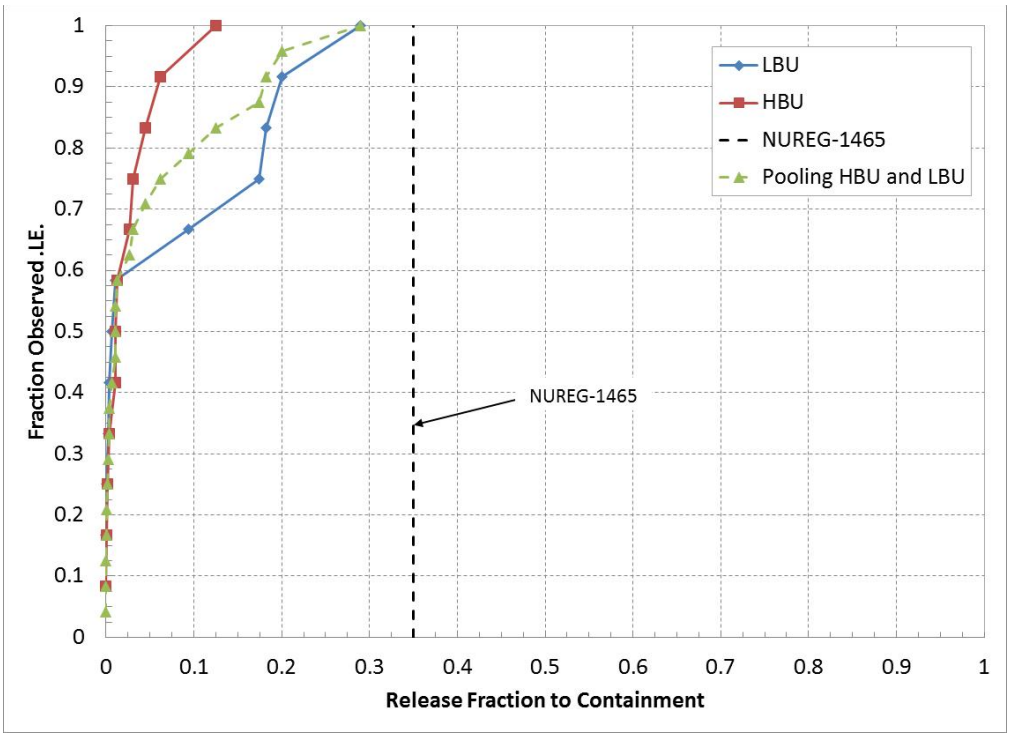

Figure 23. Calculated Release to Containment of Ex-Vessel Release Phase Cesium for MELCOR Analyses Compared with NUREG-1465.

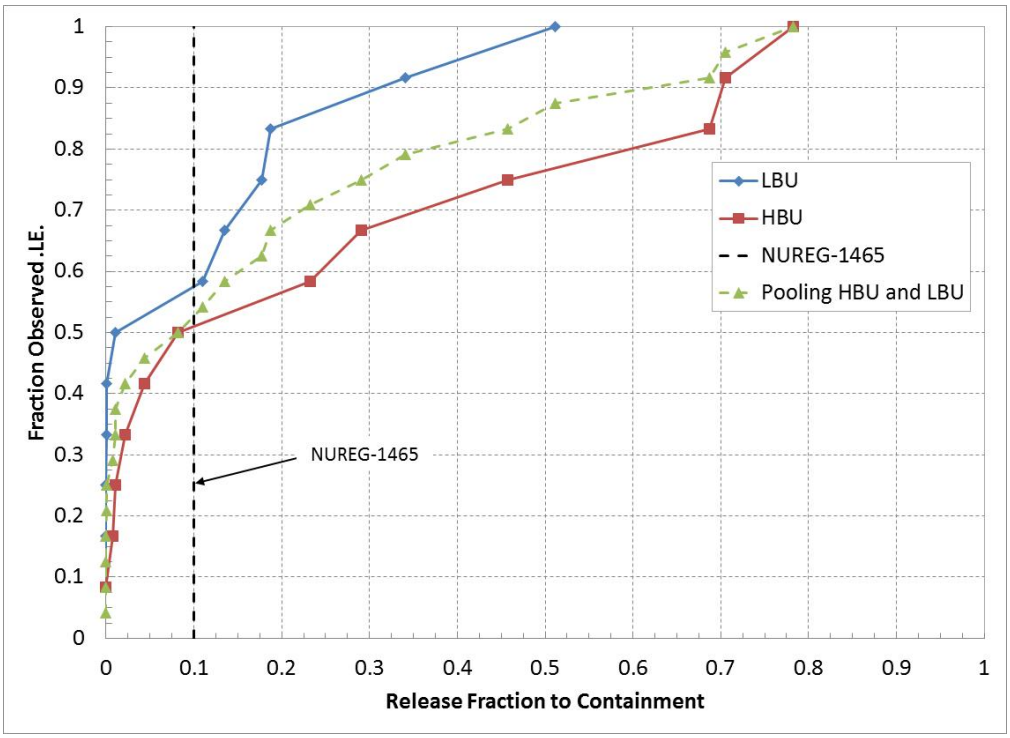

Figure 24. Calculated Release to Containment of Late In-Vessel Release Phase Iodine for MELCOR Analyses Compared with NUREG-1465. Constitutes principally iodine associated with CsI and revaporized elemental iodine from chemisorbed CsI.

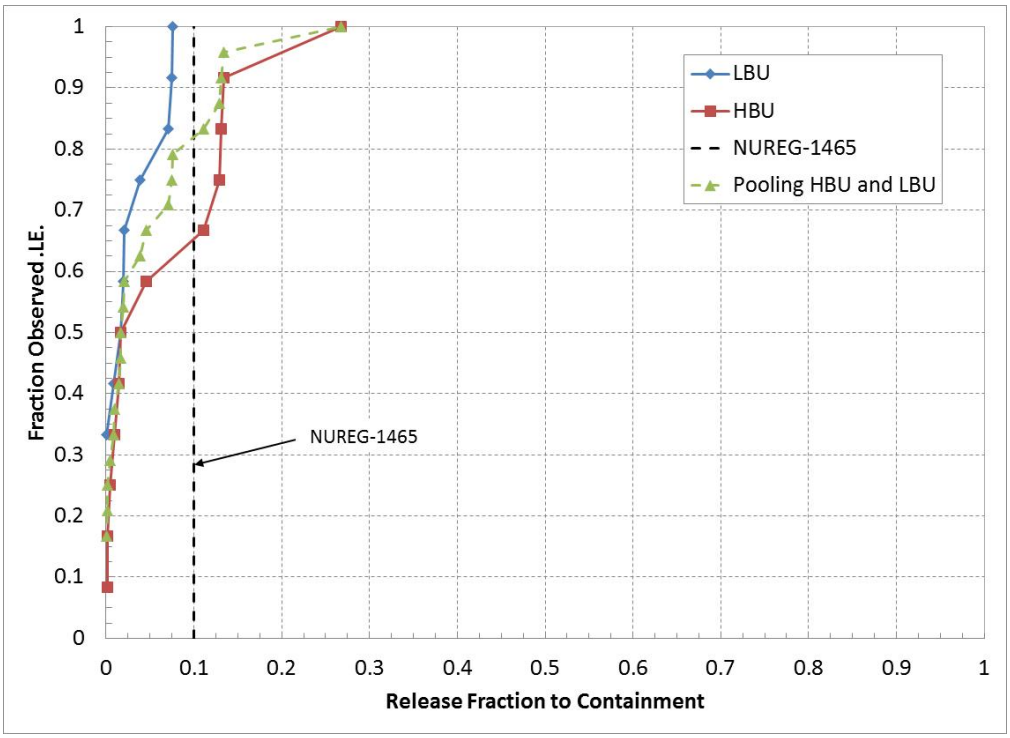

Figure 25. Calculated Release to Containment of Late In-Vessel Release Phase Cesium for MELCOR Analyses Compared with NUREG-1465. Constitutes Cs from fuel release and from revaporization of deposited CsI, Cs_2MoO_4 and CsOH.

6.3 Conclusions

In general, greater differences are observed between the state-of-the-art calculations for either HBU or LBU fuel and the NUREG-1465 containment release fractions than exist between HBU and LBU release fractions. Current analyses suggest that retention of fission products within the vessel and the RCS are greater than contemplated in the NUREG-1465 prescription, and that, overall, release fractions to the containment are therefore lower across the board in the present analyses than suggested in NUREG-1465. The decreased volatility of Cs_2MoO_4 compared to CsI or CsOH increases the predicted RCS retention of Cs, and as a result, Cs and iodine do not follow identical behaviors with respect to distribution among vessel, RCS, and containment. With respect to the regulatory AST, greater differences are observed between the NUREG-1465 prescription and both HBU and LBU predictions than exist between HBU and LBU analyses. Additionally, current analyses suggest that the NUREG-1465 release fractions are conservative by about a factor of 2 in terms of release fractions and that release durations for in-vessel and late in-vessel release periods are in fact longer than the NUREG-1465 durations.

It is currently planned that a subsequent report will further characterize these results using more refined statistical methods, permitting a more precise reformulation of the NUREG-1465 AST for both LBU and HBU fuels, with the most important finding being that the NUREG-1465 formula appears to embody significant conservatism compared to current best-estimate analyses.

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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
HScccc400 & HScccc600	CPFPL CPFAL	None CPFPL	0.9 0.9	Minimum value of CVH pool fraction such that heat transfer is calculated to Pool/Atmosphere 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.
HScccc401 HScccc601	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 [Gauntt, June 2003]
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.

Reference

R. Seigel and J.R. Howell, *Thermal Radiation Heat Transfer*, 4th ed., Taylor and Francis, New York, 2001.

APPENDIX B. DESCRIPTION OF MELCOR MODELS USED

B.1 Sequoyah MELCOR Model

The general MELCOR Control Volume Hydrodynamics (CVH) nodalization scheme used for the Sequoyah model is shown in Figures B-1 through B-3. The MELCOR Core Package (COR) model used for this work is described in Section 2.3. Updates to the model, including changes to MELCOR default values and reactor severe accident modeling best-practices are listed in Appendix A.

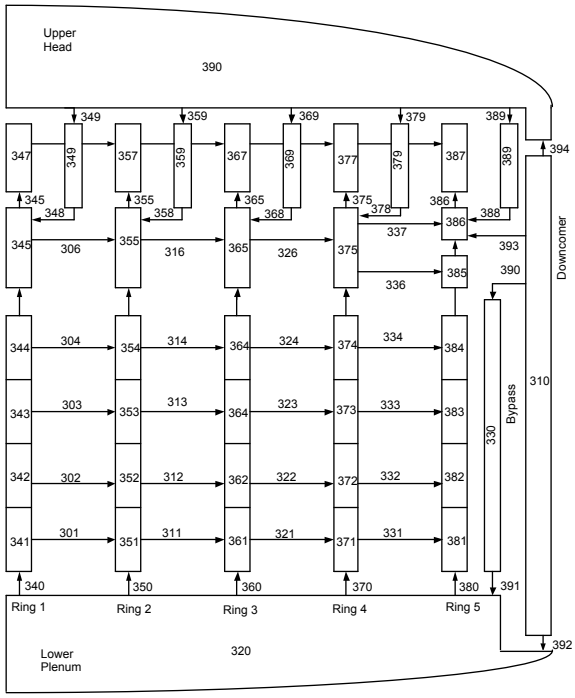

Figure B-1 Reactor Vessel MELCOR CVH Nodalization Used in the Sequoyah Model

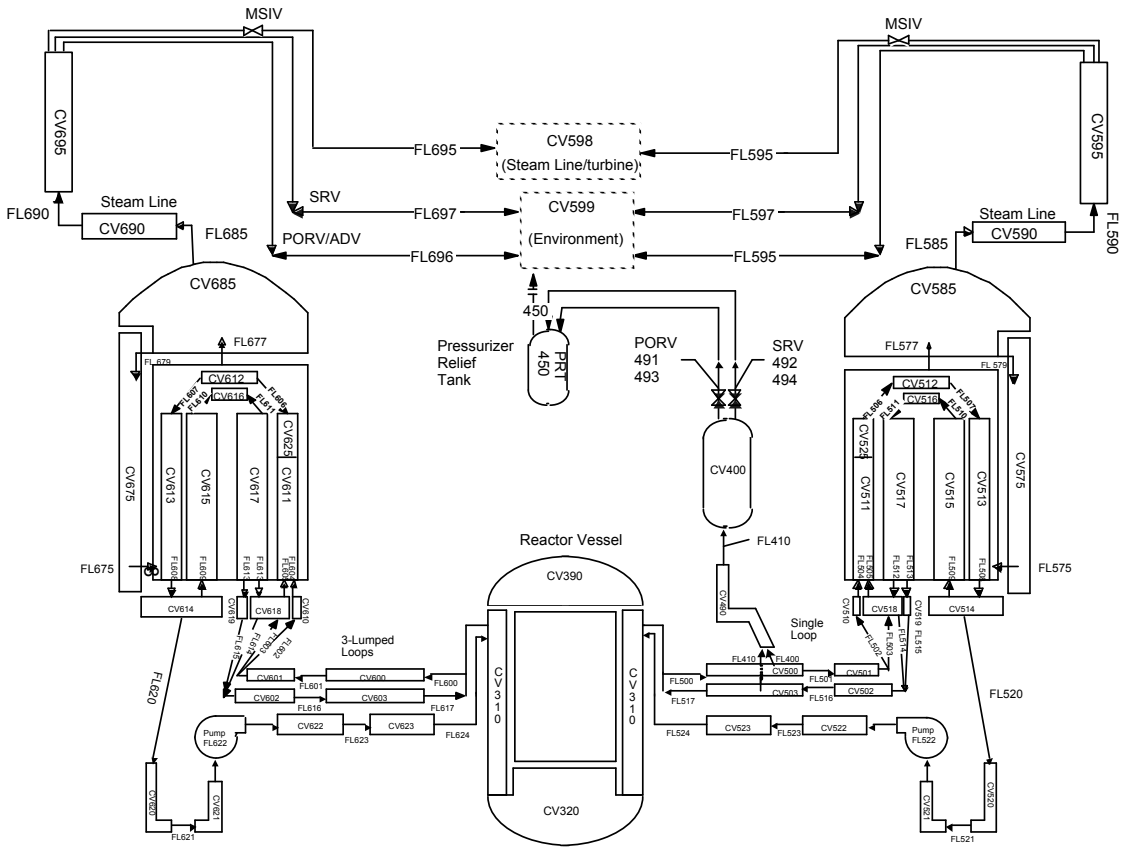

Figure B-2 Nodalization of Westinghouse 4-loop PWR RCS Used in the Sequoyah MELCOR Model

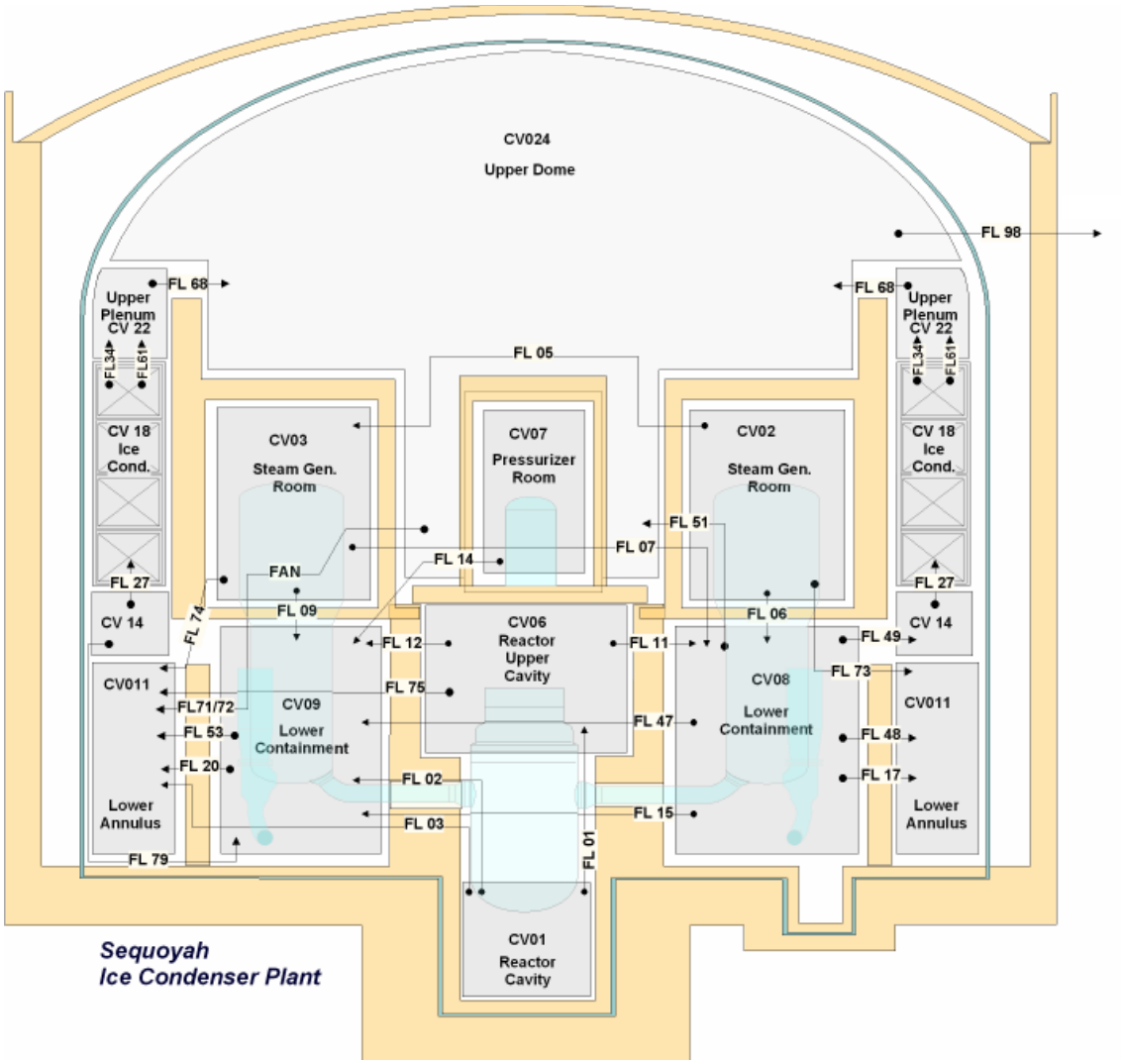

Figure B-3 Nodalization of Ice Condenser Containment Used in the Sequoyah MELCOR Model

B.2 Surry MELCOR Model

The general MELCOR CVH nodalization scheme used for the Surry model is nearly identical to that used for Sequoyah, except that the NSSS is a 3 loop system with each loop modeled independently. The containment MELCOR CVH nodalization scheme for Surry is shown in Figure B-4. Further details regarding the model can be found in the security assessment report. The MELCOR Core (COR) model used for this work is described in Section 2.3. Updates to the model, including changes to MELCOR default values and reactor severe accident modeling best-practices are listed in Appendix A.

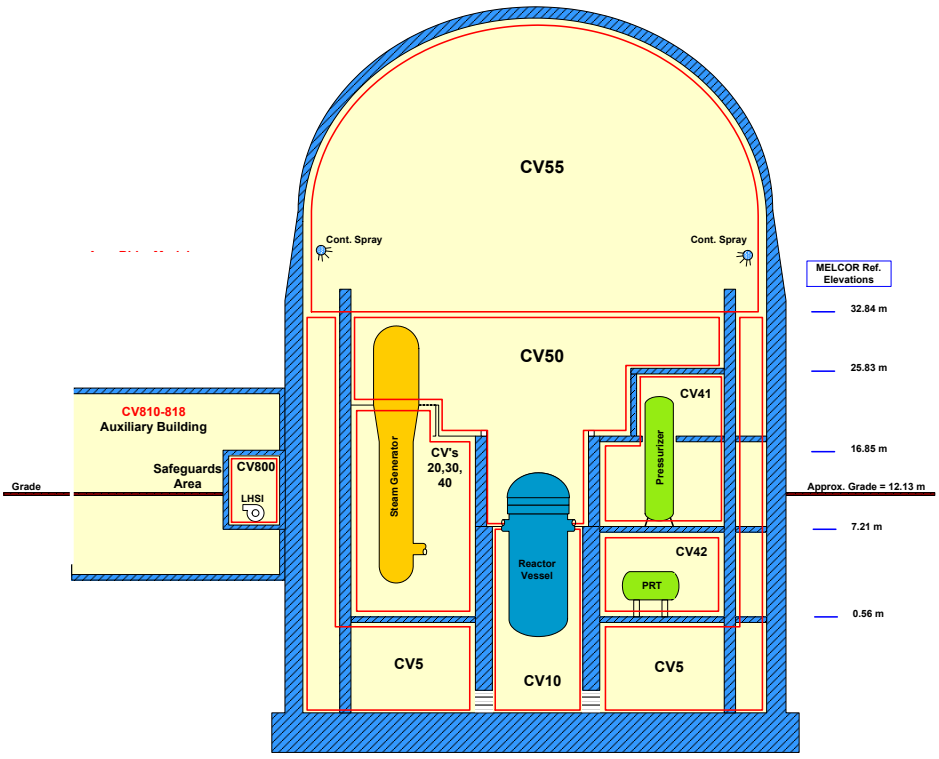

Figure B-4 Nodalization of Subatmospheric Containment Used in the Surry MELCOR Model

APPENDIX C. KEY EVENT TIMING TABLES FOR MELCOR ACCIDENT SEQUENCES

C.1 Sequoyah, Case 4A

Reactor Coolant Pump (RCP) Seal Loss-of-Coolant Accident (LOCA), No Emergency Core Cooling System (ECCS), Auxiliary Feedwater (AFW) Operates, Containment Sprays Operate in Injection Mode Only, Cavity Flooded at Vessel Failure

Event	LBU [hr]	HBU [hr]
Pump seals fail in one coolant loop, coolant leaks <= 170 gpm	0.0	0.0
Reactor trip (SCRAM)	0.31	0.31
Main feedwater trips, AFW signal received	0.31	0.31
ECCS signal, ECCS fails to start	0.31	0.31
Steam generator reaches safety setpoint, safety-relief valves (SRVs) begin to cycle	0.38	0.32
AFW starts after 60-second delay	0.33	0.33
Reactor coolant pump trip on high void	0.85	0.84
Containment sprays initiated	3.3	3.3
RWST inventory falls below 10%, spray recirculation signal	3.9	3.9
Containment sprays switchover to recirculation mode fails	3.9	3.9
Rx water level below TAF (begin “coolant activity release” phase)	5.8	5.6
Start of fuel cladding failures (begin “gap release” phase)	7.0	7.0
Release from fuel begins (begin “early release” phase)	8.2	7.4
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	8.2	8.1
Control rod poison debris first drips through core support plate	8.1	8.0
Core support plate fails	8.3	8.2
First relocation of fuel (UO_2) and clad material (Zr) to lower head	8.3	8.3
Debris quench begins, steam cooling of particulate debris above support plate	8.3	8.2
Diffuser plate fails	8.5	8.4
First hydrogen burn in containment	9.4	9.1
Containment design pressure of 12 psig (burn spike)	9.4	9.1
Lower plenum dry, rapid debris reheat in lower head begins	10.4	10.5
Lower core plate fails	11.4	9.3
Large-scale debris relocation to lower head	11.4	9.3
Accumulator injection starts	14.0	14.4
Accumulators empty	17.7	21.6
Vessel failure (begin “late release” phase)	34.1	25.2
Core debris relocation to cavity begins	34.1	25.2
Ex-Vessel cesium (Cs) release 95% complete	34.2	25.2
Containment design pressure of 12 psig (quasi-steady)	53.5	41.5
Late in-vessel Cs release 95% complete	58.2	95.3
Containment failure	80.5	65.8
Calculation terminated	168.0	168.0

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.2 Sequoyah, Case 4B

RCP Seal LOCA, No ECCS, AFW Operates, Containment Sprays Operate in Injection Mode Only, Cavity Dry at Vessel Failure

Event	LBU [hr]	HBU [hr]
Pump seals fail in one coolant loop, coolant leaks <= 170 gpm	0.0	0.0
Reactor trip (SCRAM)	0.31	0.31
Main feedwater trips, AFW signal received	0.31	0.31
ECCS signal, ECCS fails to start	0.31	0.31
Steam generator reaches safety setpoint, safety-relief valves (SRVs) begin to cycle	0.38	0.32
AFW starts after 60-second delay	0.33	0.33
Reactor coolant pump trip on high void	0.85	0.85
Containment sprays initiated	3.5	3.5
RWST inventory falls below 10%, spray recirculation signal	4.0	4.0
Containment sprays switchover to recirculation mode fails	4.0	4.0
Rx water level below TAF (begin “coolant activity release” phase)	5.6	5.8
Start of fuel cladding failures (begin “gap release” phase)	7.0	7.0
Release from fuel begins (begin “early release” phase)	8.2	7.4
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	8.2	8.2
Core support plate fails (Ring 1)	8.4	8.3
First relocation of fuel (UO_2) and clad material (Zr) to lower head	8.4	8.3
Debris quench begins, steam cooling of particulate debris above support plate	8.4	8.3
Diffuser plate fails (Ring 1)	8.5	8.4
First hydrogen burn in containment	9.7	9.4
Containment design pressure of 12 psig (burn spike)	12.9	9.4
Lower plenum dry, rapid debris reheat in lower head begins	10.2	10.2
Lower core plate fails (Ring 1)	12.1	9.1
Large-scale debris relocation to lower head	12.1	9.1
Accumulator injection starts	13.2	13.2
Vessel failure (begin “late release” phase)	13.5	13.2
Accumulators empty	13.6	13.3
Core debris relocation to cavity begins	14.1	14.5
Late in-vessel Cs release 95% complete	19.6	14.4
Ex-Vessel Cs release 95% complete	18.2	19.5
Containment design pressure of 12 psig (quasi-steady)	30.3	27.8
Containment failure	92.8	84.2
Calculation terminated	168.0	168.0

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.3 Sequoyah, Case 4C

RCP Seal LOCA, ECCS Operates in Injection Mode, AFW Operates, Containment Sprays Operate in Injection Mode, Cavity Dry at Vessel Failure

Event	LBU [hr]	HBU [hr]
Pump seals fail in one coolant loop, coolant leaks <= 170 gpm	0.0	0.0
Reactor trip (SCRAM)	0.31	0.31
Main feedwater trips, AFW signal received	0.31	0.31
ECCS signal, ECCS maintains coolant inventory	0.31	0.31
Steam generator reaches safety setpoint, safety-relief valves (SRVs) begin to cycle	0.38	0.32
AFW starts after 60-second delay	0.33	0.33
Containment sprays initiated	2.4	2.4
RWST inventory falls below 10%, spray recirculation signal	2.9	2.9
ECCS/sprays switchover to recirculation mode fails	2.9	2.9
Reactor coolant pump trip on high void	4.0	4.0
Rx water level below TAF (begin “coolant activity release” phase)	9.0	9.0
Start of fuel cladding failures (begin “gap release” phase)	10.6	10.5
Release from fuel begins (begin “early release” phase)	12.2	11.2
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	12.3	12.1
Control rod poison debris first drips through core support plate	12.1	11.9
Core support plate fails	12.4	12.2
First relocation of fuel (UO_2) and clad material (Zr) to lower head	12.4	12.3
Debris quench begins, steam cooling of particulate debris above support plate	12.4	12.2
Diffuser plate fails	12.4	12.3
First hydrogen burn in containment	12.8	13.2
Containment design pressure of 12 psig (burn spike)	12.8	13.2
Lower plenum dry, rapid debris reheat in lower head begins	14.2	14.0
Lower core plate fails (Ring 1)	14.5	13.4
Large-scale debris relocation to lower head	14.5	13.4
Vessel failure (begin “late release” phase)	18.5	16.2
Accumulator injection starts	18.1	16.2
Accumulators empty	18.6	16.3
Core debris relocation to cavity begins	19.1	16.2
Late in-vessel Cs release 95% complete	--	--
Ex-Vessel Cs release 95% complete	24.4	22.1
Containment design pressure of 12 psig (quasi-steady)	34.0	32.0
Containment failure	96.0	91.2
Calculation terminated	168.0	168.0

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.4 Sequoyah, Case 4D

Station Blackout (SBO), No ECCS, Turbine-driven AFW Operates at Max Flow until Loss of Steam, No Containment Sprays, Cavity Dry at Vessel Failure

Event	LBU [hr]	HBU [hr]
Loss of all off-site and on-site power	0.0	0.0
Reactor trip (SCRAM)	0.0	0.0
Reactor coolant pump (RCP) trip on loss of power	0.0	0.0
Main feedwater trips on loss of power, AFW signal received	0.0	0.0
RCP seal cooling lost due to component cooling water system failure	0.0	0.0
RCP seals begin leak (approx. 21 gpm/loop)	0.0	0.0
Steam generator reaches safety setpoint, safety-relief valves (SRVs) begin to cycle	0.01	0.01
Turbine-driven AFW starts after 60-second delay	0.02	0.02
Steam generators fill, AFW lost on loss of steam	0.91	0.92
Pressurizer SRVs begin to cycle	8.5	8.6
Pressurizer relief tank (PRT) rupture disks fail	8.6	8.5
RCP seals fail	8.8	8.7
Rx water level below TAF (begin “coolant activity release” phase)	8.9	8.8
Start of fuel cladding failures (begin “gap release” phase)	10.3	10.1
Release from fuel begins (begin “early release” phase)	10.9	10.3
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	10.9	10.7
Control rod poison debris first drips through core support plate	10.8	10.6
First hydrogen burn in containment	11.0	10.8
Accumulator injection starts	11.0	10.9
Core support plate fails	11.1	10.9
Diffuser plate fails	11.1	10.9
First relocation of fuel (UO_2) and clad material (Zr) to lower head	11.1	10.9
Debris quench begins, steam cooling of particulate debris above support plate	11.1	10.9
Lower core plate fails	11.5	11.1
Large-scale debris relocation to lower head	11.5	11.1
Lower plenum dry, rapid debris reheat in lower head begins	11.9	11.5
Containment design pressure of 12 psig (burn-spike)	14.2	13.7
Accumulators empty	14.2	13.4
Vessel failure (begin “late release” phase)	14.2	13.4
Core debris relocation to cavity begins	14.2	13.4
Ex-Vessel Cs release 95% complete	16.3	15.2
Containment design pressure of 12 psig (quasi-steady)	26.8	24.2
Containment failure	84.6	87.3
Late in-vessel Cs release 95% complete	46.7	152.8
Calculation terminated (HBU case terminated on CORCON error)	168.0	156.6

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.5 Sequoyah, Case 4E

SBO, No ECCS, No AFW, No Containment Sprays, Cavity Dry at Vessel Failure, Early Containment Failure

Event	LBU [hr]	HBU [hr]
Loss of all off-site and on-site power	0.0	0.0
Reactor trip (SCRAM)	0.0	0.0
Reactor coolant pump (RCP) trip on loss of power	0.0	0.0
Main feedwater trips on loss of power, AFW signal received	0.0	0.0
RCP seal cooling lost due to component cooling water system failure	0.0	0.0
RCP seals begin leak (approx. 21 gpm/loop)	0.0	0.0
Steam generator reaches safety setpoint, safety-relief valves (SRVs) begin to cycle	0.01	0.01
AFW fails to start	--	0.02
Pressurizer SRVs begin to cycle	1.5	1.9
Pressurizer relief tank (PRT) rupture disks fail	1.8	1.8
RCP seals fail	1.8	1.8
Rx water level below TAF (begin “coolant activity release” phase)	2.1	2.0
Start of fuel cladding failures (begin “gap release” phase)	3.1	3.1
Release from fuel begins (begin “early release” phase)	3.7	3.4
Control rod poison debris first drips through core support plate	3.6	3.6
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	3.7	3.7
First hydrogen burn in containment	3.8	3.7
Containment design pressure of 12 psig (burn spike)	10.0	6.3
Core support plate fails	3.8	3.8
First relocation of fuel (UO_2) and clad material (Zr) to lower head	3.8	3.8
Debris quench begins, steam cooling of particulate debris above support plate	3.8	3.8
Diffuser plate fails	3.8	3.8
Lower core plate fails	4.1	4.0
Large-scale debris relocation to lower head	4.1	4.0
Lower plenum dry, rapid debris reheat in lower head begins	7.2	4.2
Accumulator injection starts	4.6	4.6
Vessel failure (begin “late release” phase)	7.3	5.9
Core debris relocation to cavity begins	7.3	5.9
Accumulators empty	5.4	6.0
Containment failure	12.9	6.0
Ex-Vessel Cs release 95% complete	8.0	8.6
Late in-vessel Cs release 95% complete	163.9	163.7
Calculation terminated	168.0	168.0

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.6 Sequoyah, Case 4F

Large-break LOCA, No ECCS, AFW Operates, No Containment Sprays, Cavity Dry at Vessel Failure

Event	LBU [hr]	HBU [hr]
Large-break LOCA occurs in cold leg	0.0	0.0
Reactor trip (SCRAM)	0.0+	0.0+
Rx water level below TAF (begin "coolant activity release" phase)	0.0+	0.0+
ECCS signal, ECCS fails to start	0.0+	0.0+
Containment design pressure of 12 psig (blowdown spike)	0.0+	0.0+
Containment spray signal, sprays fail to start	0.0+	0.0+
Main feedwater trips, AFW signal received	0.0+	0.0+
RCP trip on high void	0.0+	0.0+
Accumulator injection starts	0.002	0.002
Accumulators empty	0.015	0.015
AFW starts after 60-second delay	0.02	0.02
Start of fuel cladding failures (begin "gap release" phase)	0.21	0.19
Release from fuel begins (begin "early release" phase)	0.37	0.28
First hydrogen burn in containment	0.56	0.76
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	0.42	0.45
Control rod poison debris first drips through core support plate	0.42	0.43
Diffuser plate fails	0.42	0.73
Core support plate fails	0.74	0.71
First relocation of fuel (UO_2) and clad material (Zr) to lower head	0.74	0.72
Lower core plate fails	0.78	0.74
Large scale debris relocation to lower head, debris quench	0.78	0.74
Lower plenum dry, rapid debris reheat in lower head begins	1.9	2.1
Steam generator reaches safety setpoint, SRVs begin to cycle	4.3	11.2
Vessel failure (begin "late release" phase)	2.4	2.9
Core debris relocation to cavity begins	2.4	2.9
Ex-Vessel Cs release 95% complete	6.3	4.6
Containment design pressure of 12 psig (quasi-steady)	12.8	11.7
Containment failure	41.8	37.0
Late in-vessel Cs release 95% complete	106.2	122.7
Calculation terminated	168.0	168.0

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.7 Sequoyah, Case 4G

Small-break LOCA in Cold Leg, No ECCS, No AFW, No Containment Sprays, Cavity Dry at Vessel Failure

Event	LBU [hr]	HBU [hr]
Small-break LOCA in cold leg	0.0	0.0
Reactor trip (SCRAM)	0.07	0.07
Main feedwater trips, AFW signal received	0.07	0.07
ECCS signal, ECCS fails to start	0.07	0.07
Steam generator reaches safety setpoint, SRVs begin to cycle	0.10	0.08
RCP trip on high void	0.28	0.28
Rx water level below TAF (begin “coolant activity release” phase)	1.7	1.6
Start of fuel cladding failures (begin “gap release” phase)	2.5	2.4
Release from fuel begins (begin “early release” phase)	2.9	2.6
Containment design pressure of 12 psig (quasi-steady)	2.6	2.6
Control rod poison debris first drips through core support plate	2.8	2.8
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	2.9	2.9
Core support plate fails	3.0	3.0
First relocation of fuel (UO_2) and clad material (Zr) to lower head	3.1	3.0
Debris quench begins, steam cooling of particulate debris above support plate	3.0	3.0
Diffuser plate fails	3.1	3.0
First hydrogen burn in containment	3.4	3.3
Lower core plate fails	3.7	3.3
Large-scale debris relocation to lower head	3.7	3.3
Lower plenum dry, rapid debris reheat in lower head begins	4.6	3.6
Accumulator injection starts	4.2	4.1
Vessel failure (begin “late release” phase)	7.2	5.0
Core debris relocation to cavity begins	7.2	5.0
Accumulators empty	5.0	5.0
Ex-Vessel Cs release 95% complete	9.6	8.8
Containment failure	61.7	58.9
Late in-vessel Cs release 95% complete	--	163.8
Calculation terminated	168.0	168.0

LBU: Core model approximates Sequoyah end-of-cycle 3, Westinghouse 17x17 fuel assemblies, peak assembly burnup = 34 GWd/MTU
HBU: Core model approximates Sequoyah end-of-cycle 12 with 10% increased burnup, Mark-BW fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.8 Surry, Case 1A

SBO, No ECCS, No AFW, No Containment Sprays, Late Containment Failure

Event	LBU [hr]	HBU [hr]
Loss of all off-site and on-site power	0.0	0.0
Reactor trip (SCRAM)	0.0	0.0
RCP trip on loss of power	0.0	0.0
Main feedwater trips on loss of power, AFW signal received	0.0	0.0
RCP seal cooling lost due to component cooling water system failure	0.0	0.0
RCP seals begin to leak (approx. 21 gpm/loop)	0.0	0.0
AFW fails to start	0.02	0.02
Steam generator reaches safety setpoint, SRVs begin to cycle	0.03	0.03
Pressurizer SRVs begin to cycle	1.1	0.82
PRT rupture disks fail	1.5	1.1
Rx water level below TAF (begin “coolant activity release” phase)	1.5	1.1
RCP seals fail	1.9	1.4
Start of fuel cladding failures (begin “gap release” phase)	2.8	1.9
Release from fuel begins (begin “early release” phase)	4.9	2.2
Accumulator injection starts	4.7	3.2
Control rod poison debris first drips through core support plate	4.9	2.5
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	5.1	2.8
Core support plate fails	5.1	2.8
First relocation of fuel (UO_2) and clad material (Zr) to lower head	7.8	3.9
Debris quench begins, steam cooling of particulate debris above support plate	5.1	2.8
Diffuser plate fails	5.9	2.8
Hot leg nozzle fails due to creep rupture	6.1	3.2
Accumulators empty	6.1	3.3
Lower core plate fails	7.8	5.5
Large-scale debris relocation to lower head	7.8	5.5
Vessel failure (begin “late release” phase)	10.3	6.2
Core debris relocation to cavity begins	10.3	6.2
Containment design pressure of 45 psig	19.8	13.5
Ex-Vessel Cs release 95% complete	148.3	16.8
Late in-vessel Cs release 95% complete	--	160.0
Containment failure	47.0	32.8
First hydrogen burn in auxiliary building	47.0	32.8
Calculation terminated	168.0	168.0

LBU: Core model approximates Surry end-of-cycle 2, Westinghouse 15x15 fuel assemblies, peak assembly burnup = 24 GWd/MTU
HBU: Core model approximates Surry end-of-cycle 17, SIF 15x15 fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.9 Surry, Case 1B

Small-break LOCA, No ECCS, AFW Operates as Designed, Containment Sprays Operate as Designed, Cavity Wet at Vessel Failure, Late Containment Failure

Event	LBU [hr]	HBU [hr]
Small-break LOCA in cold leg	0.0	0.0
Reactor trip (SCRAM)	0.03	0.03
Main feedwater trips, AFW signal received	0.03	0.03
ECCS signal received, ECCS fails to start	0.03	0.03
AFW starts after 60-second delay	0.05	0.05
Rx water level below TAF (begin "coolant activity release" phase)	1.1	1.1
RCPs trip on high void	1.3	1.3
Start of fuel cladding failures (begin "gap release" phase)	3.1	3.3
Release from fuel begins (begin "early release" phase)	3.5	3.5
Inconel debris first drips through core support plate	3.8	3.8
Accumulator injection starts	3.6	3.9
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	3.8	3.9
Core support plate fails	3.8	3.9
First relocation of fuel (UO_2) and clad material (Zr) to lower head	6.4	4.6
Debris quench begins, steam cooling of particulate debris above support plate	3.8	3.9
Diffuser plate fails	5.5	4.6
First hydrogen burn in containment	5.1	4.6
Containment sprays initiated on high containment pressure signal	5.1	4.6
Lower core plate fails	8.7	5.6
Large-scale debris relocation to lower head	8.7	5.6
RWST falls below 50,000 gal., containment sprays switch to recirculation mode	6.7	6.2
Lower plenum dry, debris re-heat in lower head begins	9.1	6.0
Vessel failure (begin "late release" phase)	11.3	21.7
Accumulators empty	11.3	18.8
Core debris relocation to cavity begins	11.3	21.7
Late in-vessel Cs release 95% complete	15.5	--
Ex-Vessel Cs release 95% complete	42.5	31.0
Containment failure	133.1	104.7
Calculation terminated	168.0	168.0

LBU: Core model approximates Surry end-of-cycle 2, Westinghouse 15x15 fuel assemblies, peak assembly burnup = 24 GWd/MTU
HBU: Core model approximates Surry end-of-cycle 17, SIF 15x15 fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.10 Surry, Case 1C

Small-break LLOCA, ECCS in Injection Model Only, No Containment Heat Removal, Containment Sprays in Injection Mode Only, No Containment Heat Removal, Cavity Wet at Vessel Failure, Late Containment Failure

Event	LBU [hr]	HBU [hr]
Large-break LOCA occurs in hot leg	0.0	0.0
Reactor trip (SCRAM)	0.002	0.002
Rx water level below TAF (begin "coolant activity release" phase)	0.01	0.01
ECCS signal	0.002	0.002
Containment design pressure of 12 psig (blowdown spike)	0.01	0.01
Containment spray signal	0.01	0.01
Main feedwater trips, AFW signal received	0.002	0.002
RCP trip on high void	0.01	0.01
Accumulator injection starts	0.03	0.03
Accumulators empty	0.04	0.04
AFW starts	0.002	0.002
RWST empty, recirculation fails	0.64	0.63
Start of fuel cladding failures (begin "gap release" phase)	9.0	6.8
Release from fuel begins (begin "early release" phase)	9.4	7.0
First hydrogen burn in containment	--	--
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	10.1	7.5
Core support plate fails	10.2	7.9
Diffuser plate fails	10.9	8.0
First relocation of fuel (UO_2) and clad material (Zr) to lower head	10.9	10.1
Lower core plate fails	12.0	10.1
Large scale debris relocation to lower head, debris quench	10.2	7.9
Lower plenum dry, rapid debris re-heat in lower head begins	12.4	10.2
Vessel failure (begin "late release" phase)	12.9	10.5
Core debris relocation to cavity begins	12.9	10.5
Ex-Vessel Cs release 95% complete	96.2	17.4
Containment design pressure of 45 psig	14.8	10.7
Late in-vessel Cs release 95% complete	13.2	166.3
Containment failure	--	--
Calculation terminated	168.0	168.0

LBU: Core model approximates Surry end-of-cycle 2, Westinghouse 15x15 fuel assemblies, peak assembly burnup = 24 GWd/MTU
HBU: Core model approximates Surry end-of-cycle 17, SIF 15x15 fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.11 Surry, Case 1D

SBO, No RCP Seal Failure, Late Containment Failure

Event	LBU [hr]	HBU [hr]
Loss of all off-site and on-site power	0.0	0.0
Reactor trip (SCRAM)	0.0	0.0
RCPs trip on loss of power	0.0	0.0
Main feedwater trips on loss of power, AFW signal received	0.0	0.0
ECCS signal received, ECCS fails to start	0.0	0.0
AFW fails to start after 60-second delay	0.02	0.02
Rx water level below TAF (begin “coolant activity release” phase)	1.6	1.2
Start of fuel cladding failures (begin “gap release” phase)	2.7	2.0
Release from fuel begins (begin “early release” phase)	3.4	2.2
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	3.7	2.6
Inconel debris first drips through core support plate	3.2	2.4
Core support plate fails	3.6	2.7
Debris quench begins, steam cooling of particulate debris above support plate	3.6	2.7
Hot leg nozzle fails due to creep rupture	3.7	2.8
Accumulator injection starts	3.7	2.8
First hydrogen burn in containment	3.7	2.8
Accumulators empty	3.7	2.8
Diffuser plate fails	4.1	3.6
First relocation of fuel (UO_2) and clad material (Zr) to lower head	5.0	3.6
Lower core plate fails	6.3	4.4
Large-scale debris relocation to lower head	6.3	4.4
Lower plenum dry	6.7	4.6
Vessel failure (begin “late release” phase)	7.9	6.1
Core debris relocation to cavity begins	7.9	6.1
Ex-Vessel Cs release 95% complete	148.0	10.6
Late in-vessel Cs release 95% complete	8.3	163.5
Containment failure	42.8	31.5
Calculation terminated	168.0	168.0

LBU: Core model approximates Surry end-of-cycle 2, Westinghouse 15x15 fuel assemblies, peak assembly burnup = 24 GWd/MTU
HBU: Core model approximates Surry end-of-cycle 17, SIF 15x15 fuel assemblies, peak assembly burnup = 59 GWd/MTU

C.12 Surry, Case 1F

Small-break LOCA, No ECCS, AFW Operates as Designed, Containment Sprays Operate as Designed, Cavity Wet at Vessel Failure, Early Containment Failure

Event	LBU [hr]	HBU [hr]
Small-break LOCA in cold leg	0.0	0.0
Reactor trip (SCRAM)	0.03	0.03
Main feedwater trips, AFW signal received	0.03	0.03
ECCS signal received, ECCS fails to start	0.03	0.03
AFW starts after 60-second delay	0.03	0.03
Rx water level below TAF (begin “coolant activity release” phase)	1.1	1.1
RCPs trip on high void	1.3	1.3
Start of fuel cladding failures (begin “gap release” phase)	3.1	3.3
Release from fuel begins (begin “early release” phase)	3.5	3.5
Inconel debris first drips through core support plate	3.5	3.8
Accumulator injection starts	3.6	3.9
First relocation of fuel (UO_2) and clad material (Zr) to core support plate	3.7	3.9
Core support plate fails	3.8	3.9
First relocation of fuel (UO_2) and clad material (Zr) to lower head	6.4	4.6
Debris quench begins, steam cooling of particulate debris above support plate	3.8	3.9
Diffuser plate fails	5.5	4.6
First hydrogen burn in containment	5.1	4.6
Containment sprays initiated on high containment pressure signal	5.1	4.6
Lower core plate fails	8.7	5.6
Large-scale debris relocation to lower head	8.7	5.6
RWST falls below 50,000 gal., containment sprays switch to recirculation mode	6.7	6.2
Lower plenum dry, debris re-heat in lower head begins	9.1	6.0
Accumulators empty	11.3	18.8
Vessel failure (begin “late release” phase)	11.3	21.7
Core debris relocation to cavity begins	11.3	21.7
Containment failure	11.3	
Ex-Vessel Cs release 95% complete	26.7	30.5
Late in-vessel Cs release 95% complete	12.3	70.5
Calculation terminated	168.0	168.0

LBU: Core model approximates Surry end-of-cycle 2, Westinghouse 15x15 fuel assemblies, peak assembly burnup = 24 GWd/MTU
HBU: Core model approximates Surry end-of-cycle 17, SIF 15x15 fuel assemblies, peak assembly burnup = 59 GWd/MTU

APPENDIX D. ACCIDENT PROGRESSION SIGNATURES FOR SELECTED ACCIDENT SEQUENCES

The following parameters are plotted from both the low burnup and high burnup calculations for a station blackout sequence with late containment failure in the Westinghouse three-loop pressurized-water reactor with a sub-atmospheric containment (Surry) model:

- Reactor vessel pressure (psia)
- Reactor vessel water levels – downcomer and in-shroud (m)
- Steam generator pressure (psia)
- Steam generator water level (m)
- Fuel cladding temperature in the core (K)
- Total quantity of hydrogen generated by in-vessel oxidation (kg)

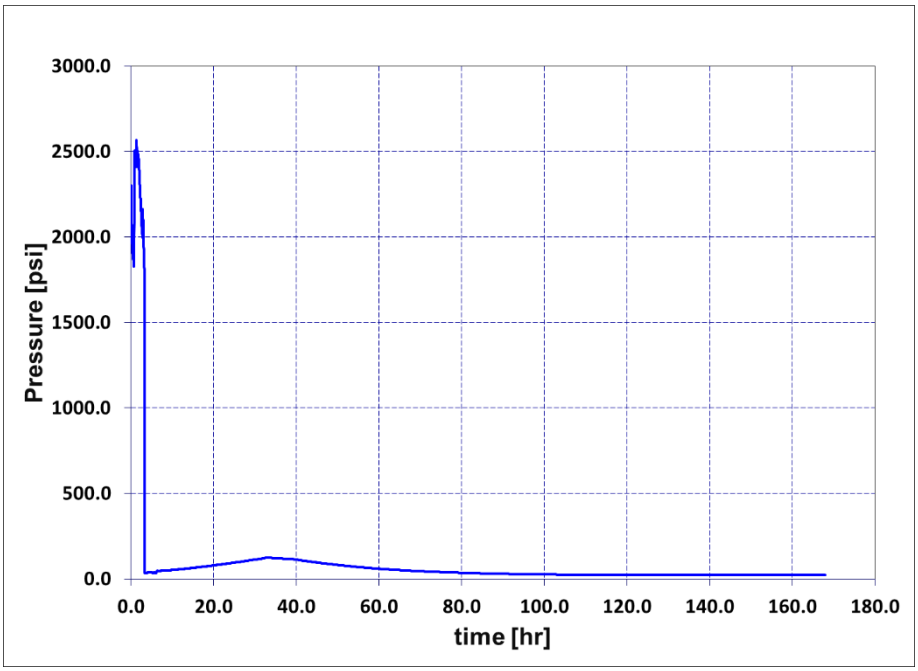

Figure D-1 Reactor vessel pressure; Surry case 1A; HBU.

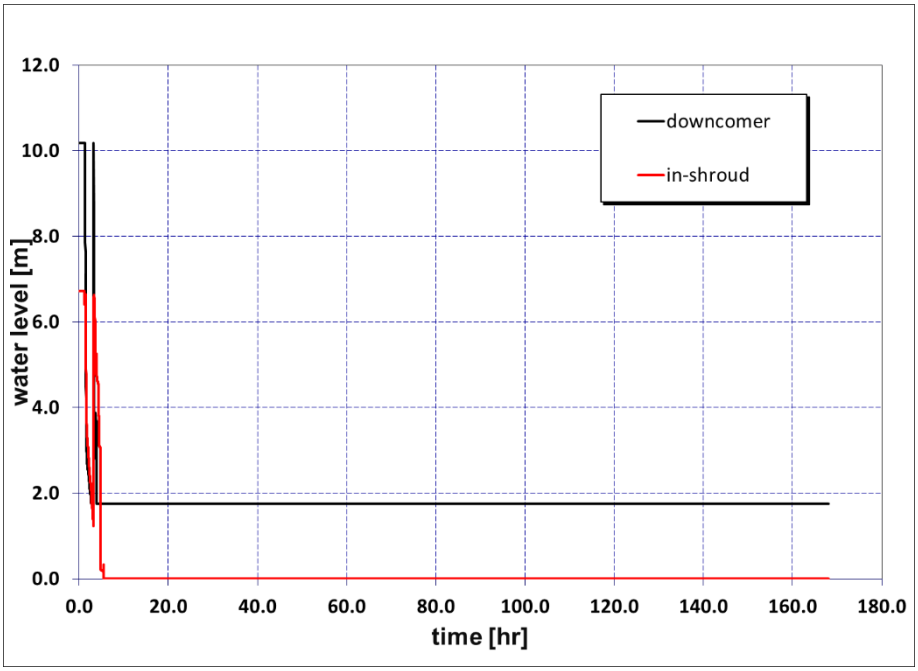

Figure D-2 Reactor vessel water levels – downcomer and in-shroud; Surry case 1A; HBU.

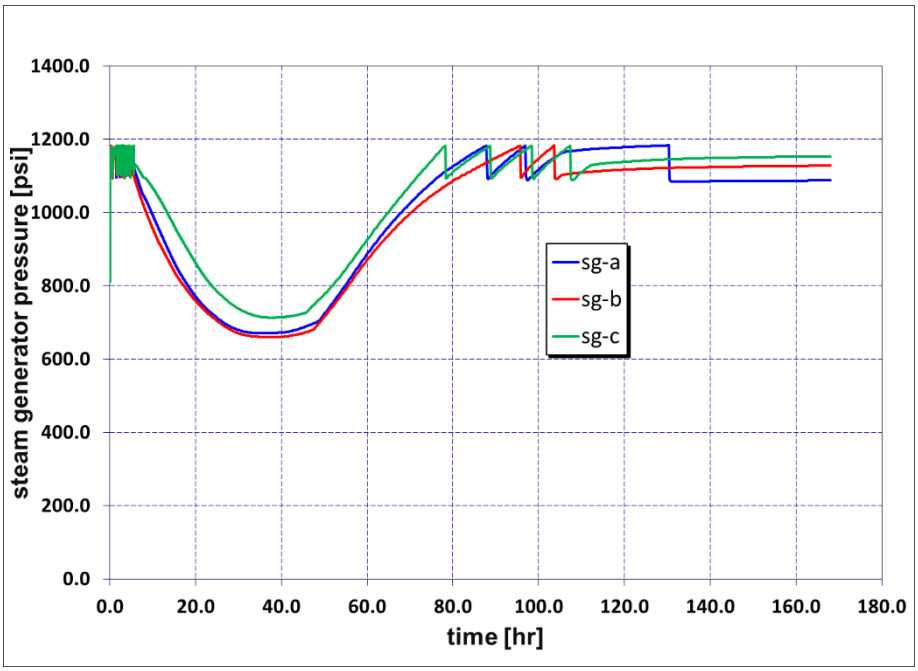

Figure D-3 Steam generator pressure; Surry case 1A; HBU

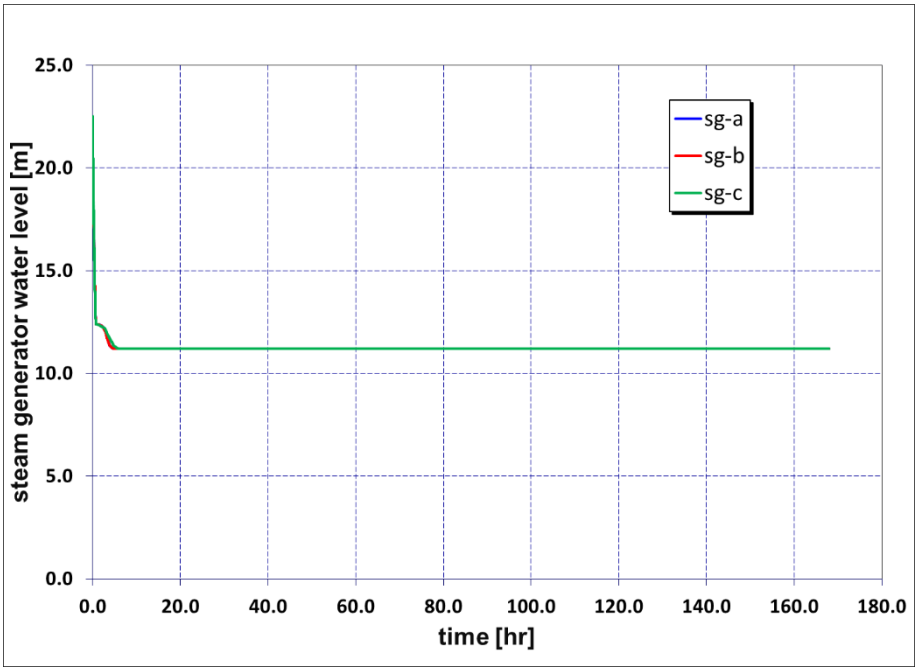

Figure D-4 Steam generator water level; Surry case 1A; HBU

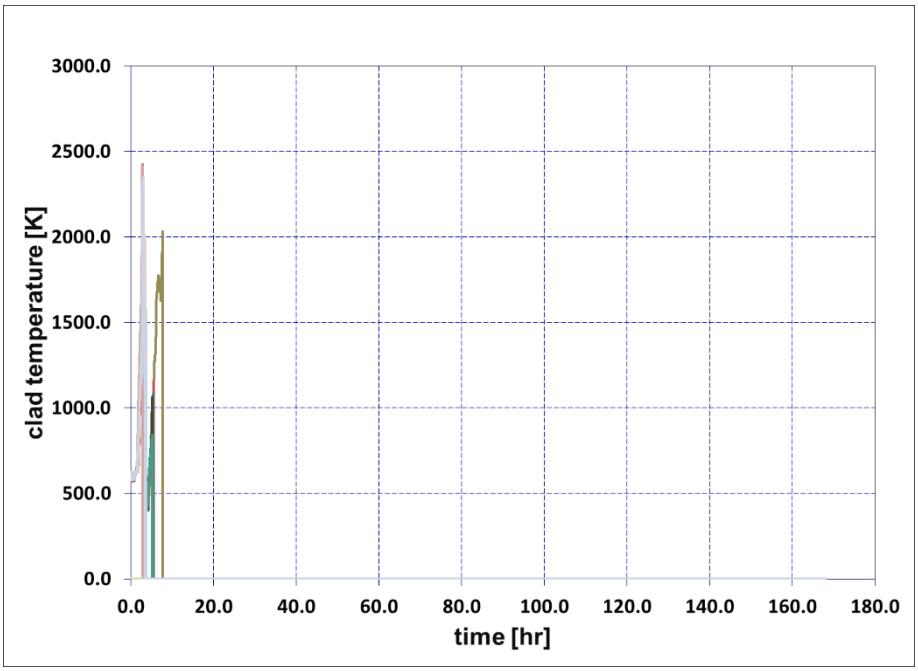

Figure D-5 Fuel cladding temperatures in the core; Surry case 1A; HBU

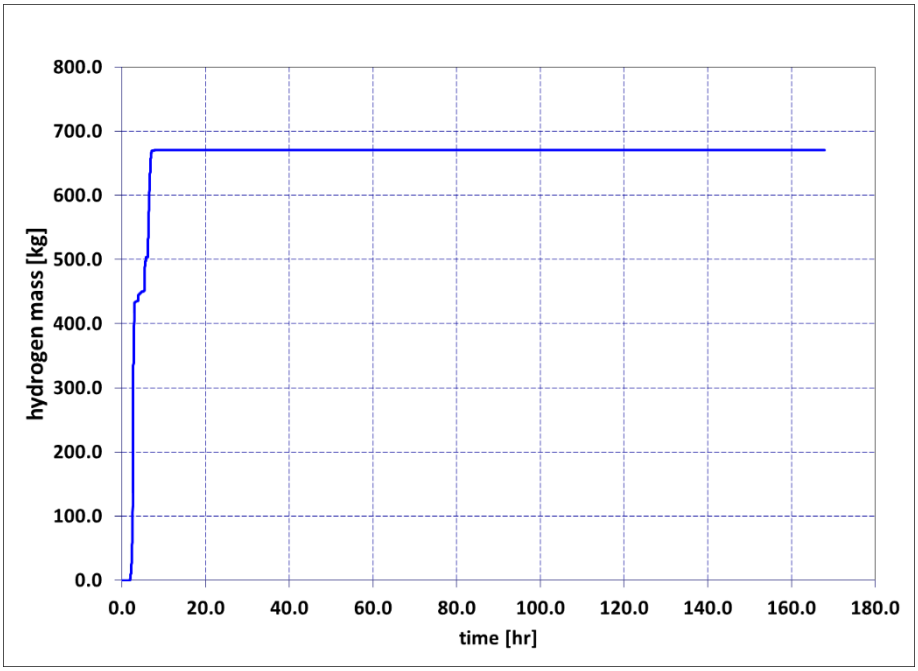

Figure D-6 • Total quantity of hydrogen generated by in-vessel oxidation; Surry case 1A; HBU

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