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Bounding Radionuclide Release Estimates for a Hypothetical Power Reactor Accident

Kenneth C. Wagner
Jennifer Leute

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ABSTRACT

The object of this study is to provide an estimate of bounding radionuclide releases from a nuclear power plant accident. The time frame of interest is the release phase from the initiating event through 30 days. The maximum credible initiating event includes an initial failure of the containment function with a primary system leak. All estimates include a complete loss-of-onsite power and no successful mitigative actions. The active safety injection systems are also assumed failed. The review considers the following commonly deployed reactor designs in the following order of interest: RBMK 1000, VVER-440, VVER-1000, 1000 MWe PWR, 1000 MWe BWR, BN-800, and the 600 MWe CANDU/PHWR. The review also considers spent fuel pool accident scenarios.

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ACRONYMS AND DEFINITIONS

Abbreviation	Definition
BWR	boiling water reactor
DF	decontamination factor (alternate method for describing radionuclide retention)
FLEX	Diverse and Flexible Coping Strategies
LOCA	loss-of-coolant accident
MACCS	MELCOR Accident Consequence Code System
MWe	megawatt electric
NEI	Nuclear Energy Institute
NRC	United States Nuclear Regulatory Commission
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
PWR	pressurized water reactor
RCP	reactor coolant pump
Sandia	Sandia National Laboratories
SBO	station blackout
SOARCA	State-of-the-Art Reactor Consequence Assessment
STSBO	short-term station blackout
SV	safety valve
UA	uncertainty analysis

1. OBJECTIVE

The object of this study is to provide an estimate of the bounding radionuclide releases from a nuclear power plant accident. The time frame of interest is the release phase from the initiating event through 30 days. The maximum credible initiating event includes an initial failure of the containment function with a primary system leak. All estimates include a complete loss-of-onsite power and no successful mitigative actions. The active safety injection systems are assumed failed. The review considers the following commonly deployed reactor designs in order of interest:

- RBMK-1000,
- VVER-440,
- VVER-1000,
- 1000 MWe PWR,
- 1000 MWe BWR,
- BN-800, and
- 600 MWe CANDU/PHWR.

The spent fuel pools at these reactors are another source of radionuclides, which will also be included.

2. HISTORICAL PERSPECTIVE OF BOUNDING SOURCE TERM

The definition of the bounding source term has been evolving over the history of commercial nuclear power. All nuclear plants are designed to safely recover from design-basis accidents. Section 2.1 briefly discusses United States Nuclear Regulatory Commission (NRC) licensing requirements for design basis accidents. It is important to mention design basis events to demonstrate the robustness of the designs to a wide range of initiating events. Since the beginning of the regulation of nuclear plants in the US, there were also prescriptive requirements for radiological release for a whole core melt-down. The history and evolution of the bounding whole core meltdown source term is summarized in Section 2.1. As nuclear research progressed, probabilistic risk assessment (PRA) presented a systematic and comprehensive approach to identifying core melt scenarios, their expected frequency, and their consequences, which is discussed in Section 2.2. Related and recent research at Sandia National Laboratories (SNL) for the NRC to improve the estimation of the source term from the most likely severe accidents is described in Section 2.3. Finally, it is also important to review historical accidents for insights into the magnitude of actual accident source terms. The most notable accidents are the Three Mile Island Unit 2 (TMI-2), Chernobyl Unit 4, and Fukushima Dai-ichi nuclear accidents, which are described in Section 2.4.

2.1. Design Basis Accidents

The history of commercial reactor regulatory applications has included laws and requirements to identify a range of accidents and their acceptable limiting consequences. Through defense-in-depth licensing requirements, the accidents range from expected operational events to severe events that are possible but not expected in the lifetime of the plant. The design basis safety assessment includes a variable success criteria depending on the anticipated frequency of event. In the United States (US), the range of initiating events include [1]:

- Condition I: Normal operation and operational transients
- Condition II: Faults of moderate frequency
- Condition III: Infrequent faults
- Condition IV: Limiting faults

The licensee must demonstrate that the plant can meet the stringent requirements for acceptable performance. The Condition IV events include instantaneous failure of the largest recirculation pipe and the most limiting additional failure. It is required that the reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events. Examples of Condition IV events include,

- Steam system piping failure
- Feedwater system pipe break
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break
- Spectrum of control rod ejection accidents
- Steam generator tube rupture
- LOCAs resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (large break)

The mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) are verified to achieve the following design basis safety criteria:

- Fuel damage, defined as perforation of the fuel cladding, does not occur during normal operation and anticipated operational transients.
- The minimum departure from nuclear boiling ratio (DNBR) in normal operation and anticipated transient conditions is always above the design threshold for fuel cooling without film boiling.
- Fuel melting does not occur at the overpower limit for Condition I or II events.
- The maximum fuel rod cladding temperature following a loss-of-coolant accident (including Condition IV events) is less than 2200°F.
- The maximum core average linear power is within regulatory limits for normal operation and anticipated transient conditions,
- The calculated total heat flux hot channel factor for normal operation and anticipated transient conditions are within regulatory limits.
- The calculated control rod worth has sufficient negative reactivity to provide the required shutdown margin even without the highest worth control rod not inserting.

Although these basis rules were historically implemented slightly differently, Russian, US, and European, and Canadian regulatory agencies have similar requirements for design basis safety requirements.

2.2. Bounding Regulatory Source Term

All licensed plants meet all applicable design basis requirements cited in Section 2.1. In addition, the regulatory requirements also include the plant response a bounding source term accident, which provides insights for this project. For example, while Condition IV design basis limiting faults are not expected in the lifetime of the plant, the number of concurrent failures that challenged the plant safety systems is limited to the initiating event and one additional failure. As an alternate approach to evaluate the plant's defense-in-depth, the Atomic Energy Commission (AEC) developed the bounding regulatory source term in Technical Information Document 14884 (TID-14884). Specifically, TID-14884 was used for plant siting [2]. TID-14884 includes a large radionuclide release from the fuel into the containment in this bounding nuclear accident. However, the final defense-in-depth barrier (i.e., the containment), was assumed to be intact but leak at the 0.1% mass/day at design conditions (i.e., the containment design pressure). TID-14884 specified a whole core meltdown radionuclide release from the nuclear fuel to the containment as follows,

- 100% of the noble gas,
- 50% of the halogens, which half is retained by absorption onto structures and settling, and
- 1% of the solid fission products

The TID-14884 source term represented approximately 15% of the fission product inventory of the nuclear fuel.

Further regulatory guidance on TID-14884 was provided via boiling water reactor (BWR) and pressurized water reactor (PWR) regulatory guides for acceptable methods to calculate the siting source term [3][4]. In particular, the 25% radioactive iodine inventory available for release should be specified as follows,

- 91% as elemental iodine (gaseous),

- 5% as particulates (aerosol chemical form), and
- 4% as organic iodides

2.3. Ongoing Activities to Characterize the Bounding Source Term

Following the establishment of TID-14884, the subsequent decades of bounding nuclear accident research led to three approaches to characterizing bounding accidents; (1) continued refinement of conservative bounding site licensing regulatory source terms, (2) best-estimate evaluations that considered frequency and consequences of events in PRAs, and (3) the more recent severe accident uncertainty assessments (e.g., like those performed at Sandia for the NRC). These research and regulatory studies also have relevance to forming a bounding source term estimate.

2.3.1. NUREG-1465 Regulatory Guidance and Follow-on Studies

The regulatory bounding siting methodology from TID-14484 and the associated BWR and PWR regulation guides were updated in NUREG-1465, which provided an acceptable method to perform bounding releases for siting. NUREG-1465 used time frames called release phases to better reflect the accident time progression. The key events that characterize the four accident phases are the first cladding failure, the significant buildup of noble gases in the containment, the reactor lower head failure, the end of ex-vessel releases, and the end of the in vessel revaporization release. The time intervals between the key events are identified as the gap release, in-vessel release, ex-vessel release, and late in-vessel release phases. A graphical view of the phases and transition events are shown in Figure 1. Each phase has a characterized start time, duration, and fission product release fraction. The specifications of the start and end of the four phases are shown in Table 1.

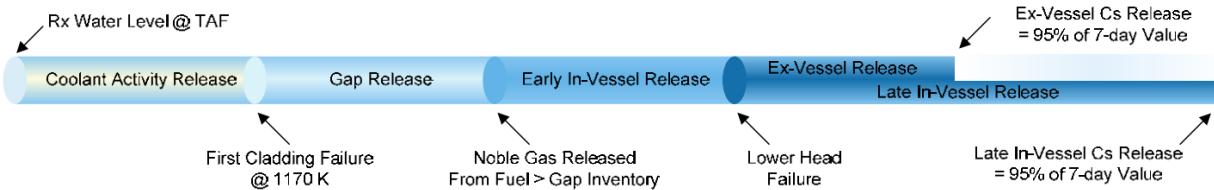


Figure 1 Release phase timing definitions from NUREG-1465 [5].

Table 1 Fission product release phase definition

Phases	Start Definition	End Definition
Gap Release	First Cladding Failure	5% Xe in containment
In-Vessel Release	5% Xe in Containment	Lower head failure
Ex-Vessel Release	Lower head failure	95% of I in containment from cavity
Late In-Vessel Release	Lower head failure	95% of I in containment from remaining sources

NUREG-1465 also significantly modified the source term based a review of experiments, code calculations, and PRAs for likely accident contributors. Table 2 and Table 3 show the corresponding source term specifications for BWRs and PWRs. The elements assigned to each radionuclide group

are defined in Table 4. The key source term changes with NUREG-1465 include (a) the time-dependent release from the fuel to the containment and (b) refinement of the halogen definition to be 95% in the cesium-iodide aerosol chemical form and 5% as Iodine and HI gas. The redefinition and percentage distributions of the chemical forms of iodine was a significant change. TID-14884 specified 95% of the iodine as gaseous forms that had no natural mechanism for deposition in the containment (i.e., the regulatory guides allowed for deposition of aerosols [3][4]).

Subsequently, Sandia has supported NRC with revising the guidance for NUREG-1465 using MELCOR. These new studies addressed some limitations in the NUREG-1465 study and allowed for higher burn-ups and mixed oxide cores [6][7][8][9]. These studies incrementally advanced the estimation of the source term with higher fidelity modeling of in-vessel natural circulation patterns, inclusion of molybdenum as a contributor to the early volatile release due to the formation of cesium molybdate, higher tellurium releases in a chemical form that does not readily react with residual fuel cladding and metal surfaces with the reactor coolant system, and larger late releases contributions from revaporization of previously deposited volatile radionuclides from primary system surfaces [8]. These studies are still ongoing at Sandia and progressing towards new regulatory guidance. Table 5 and Table 6 shows the updated recommendations for BWRs and PWRs, respectively, using the more advanced modeling and higher fuel burnups.

Table 2 NUREG-1465 BWR licensing source term [5].

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

* Values shown are fractions of core inventory.

** See Table 3.8 for a listing of the elements in each group

*** Gap release is 3 percent if long-term fuel cooling is maintained.

Table 3 NUREG-1465 PWR licensing source term [5].

	Gap Release***	Early 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
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

* Values shown are fractions of core inventory.

** See Table 3.8 for a listing of the elements in each group

*** Gap release is 3 percent if long-term fuel cooling is maintained.

Table 4 Listing of the radionuclide groups used in NUREG-1465 [5].

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	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 5 Comparison of BWR high burnup durations and release fractions (bold entries) with those recommended for BWRs in NUREG-1465 (parenthetical entries) [5] [8].

	Gap Release	In-vessel Release	Ex-vessel Release	Late In-vessel Release
Duration (hours)	0.16 (0.5)	8.0 (1.5)	2.9 (3.0)	12 (10)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr, Xe)	0.008 (0.05)	0.96 (0.95)	0.009 (0)	0.016 (0)
Halogens (Br, I)	0.002 (0.05)	0.47 (0.25)	0.013 (0.30)	0.39 (0.01)
Alkali Metals (Rb, Cs)	0.002 (0.05)	0.13 (0.20)	0.01 (0.35)	0.05 (0.01)
Alkaline Earths (Sr, Ba)	-	0.005 (0.02)	0.029 (0.10)	0.005 (0)
Tellurium Group (Te, Se, Sb)	0.002 (-)	0.39 (0.05)	0.002 (0.25)	0.33 (0.005)
Molybdenum (Mo, Tc, Nb)	-	0.02 (0.0025)	0.003 (0.0025)	0.0055 (0)
Noble Metals (Ru, Pd, Rh, etc.)	-	0.0027 (0.0025)	[0.0025]	1.0x10⁻⁴ (0)
Lanthanides (Y, La, Sm, Pr, etc.)	-	1.4x10⁻⁷ (2x10 ⁻⁴)	5x10⁻⁵ (0.005)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	1.3x10⁻⁷ (2x10 ⁻⁴)	0.0021 (0.005)	-

Table 6 Comparison of PWR high burnup durations and release fractions (bold entries) with those recommended for PWRs in NUREG-1465 (parenthetical entries) [5] [8].

	Gap Release	In-vessel Release	Ex-vessel Release	Late In-vessel Release
Duration (hours)	0.22 ± 0.04 (0.33 ± 0.12)	4.5 ± 2.4 (5.3 ± 1.2)	4.8 ± 1.3 (9 ± 10)	143 ± 8 (130 ± 20)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr, Xe)	0.017 ± 0.003 (0.022 ± 0.002)	0.94 ± 0.01 (0.85 ± 0.05)	0.011 ± 0.008 (0.08 ± 0.05)	0.003 ± 0.003 (0.002 ± 0.002)
Halogens (Br, I)	0.004 ± 0.002 (0.007 ± 0.002)	0.37 ± 0.13 (0.30 ± 0.13)	0.011 ± 0.008 (0.08 ± 0.03)	0.21 ± 0.16 (0.15 ± 0.11)
Alkali Metals (Rb, Cs)	0.003 ± 0.001 (0.005 ± 0.002)	0.23 ± 0.10 (0.23 ± 0.10)	0.02 ± 0.01 (0.03 ± 0.04)	0.06 ± 0.04 (0.03 ± 0.01)
Alkaline Earths (Sr, Ba)	0.0006 ± 0.0003 (0.0014 ± 0.0006)	0.004 ± 0.002 (0.004 ± 0.001)	0.003 ± 0.002 (0.002 ± 0.001)	-
Tellurium Group (Te, Se, Sb)	0.004 ± 0.002 (0.007 ± 0.003)	0.30 ± 0.12 (0.26 ± 0.11)	0.003 ± 0.002 (0.03 ± 0.01)	0.10 ± 0.10 (0.10 ± 0.07)
Molybdenum (Mo, Tc, Nb)	-	0.08 ± 0.03 (0.10 ± 0.02)	0.01 ± 0.01 (0.10 ± 0.09)	0.03 ± 0.03 (0.05 ± 0.06)
Noble Metals (Ru, Pd, Rh, etc.)	-	0.006 ± 0.006 (0.006 ± 0.004)	[0.0025]	-
Lanthanides (Y, La, Sm, Pr, etc.)	-	1.5 ± 1.2 x10⁻⁷ (1.1 ± 0.9 x10 ⁻⁷)	1.3 ± 0.3 x10⁻⁵ (2.6 ± 0.8 x10 ⁻⁵)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	1.5 ± 1.2 x10⁻⁷ (1.1 ± 0.9 x10 ⁻⁷)	2.4 ± 0.9 x10⁻⁴ (1.0 ± 0.8 x10 ⁻⁴)	-

2.3.2. PRA Accident Characterization Studies

In parallel to the regulatory source term research and regulatory guidance, PRAs evolved as an alternate method to characterize plant safety. PRAs are also now required for new designs. PRAs compliment design basis evaluations by considering the consequences from a systematic and comprehensive assessment of events, failures, and outcomes. For example, the Three Mile Island Unit 2 (TMI-2) accident was not considered in the licensing design basis evaluations due to multiple

equipment failures and incorrect operator actions. In contrast, PRAs systematically evaluate the potential for combinations of failures that could lead to fuel damage and evaluate their likelihood or frequency of occurrence. The consequences from the higher frequency sequences are evaluated to determine their consequences, such as the offsite dose or the risk of cancer and death. The multiplication of the frequency of the sequences and their consequences form risk measures. The NRC completed the landmark reactor safety study in 1978 [10], which was the first comprehensive nuclear reactor PRA. The reactor safety study was subsequently updated with the NUREG-1150 study, which examined the over plant risk at five of the prevalent reactor designs in the US [11]. The range of initiating events in the NUREG-1150 study includes both internal and external initiating events. Internal events comprise a wide range core damage challenges that originate from within the plant (e.g., component failures, fires, flooding). External events comprise natural phenomena possible at the plant site (e.g., seismic events, externally-caused floods and fires, high winds such as tornados and hurricanes, and non-malevolent aircraft crashes). The source terms from PRA severe accident sequences provide a basis for bounding source term events. Security events were not included in these landmark PRA studies.

2.3.3. SOARCA Severe Accident Studies

Finally, the NRC conducted the state-of-art reactor consequence assessment (SOARCA) project [12]. References [13] and [14] documented the accident progression calculations and estimates of the offsite radiological health consequences for high frequency severe reactor accidents identified for two pilot plants: the Peach Bottom Atomic Power Station, a BWR with a Mark I containment and the Surry Power Station PWR with a subatmospheric large dry containment, respectively. The SOARCA project's integrated modeling of accident progression and offsite consequences used both state-of-the-art computational analysis tools (the MELCOR code and the MELCOR Accident Consequence Code System [MACCS]) and best modeling practices drawn from the severe accident analysis community.

The SOARCA project was followed with three uncertainty assessments (UAs). Whereas the original SOARCA project provided best-estimate evaluations of high-frequency, severe accident sequences, the SOARCA UAs performed in-depth and integrated evaluation of the uncertainty in accident progression, radiological release, and offsite health consequence projections [15][16][17]. The three UAs continued with the Surry PWR and Peach Bottom BWR nuclear plants and added the Sequoyah nuclear power plant with an ice condenser containment. The results from the UAs provide an updated characterization of the uncertainty of a severe accident source term for the representative plants and scenarios, i.e., something not previous done. The SOARCA UA results are relevant to the bounding source term discussion and show a reduced source term with delayed failure of the containment without mitigation.

In Reference [18], the results from the Surry UA were reformulated into the NUREG-1465 accident phase format. Because the Surry UA propagates phenomenological uncertainties over one thousand simulations, the accident phase timings, durations, and fission product releases during the various phases are distributions rather than single values. The distributions provide additional insights for comparison to the NUREG-1465 results, which is based on single calculations for each sequence.

The radionuclide values from the Surry UA are reported as fractions of the initial inventory. However, the Surry UA considered 20 unique inventories spanning the full reactor time-in-the-cycle. The 20 radionuclide inventories represent the various states of achieving a secular equilibrium and masses as a function of the time-in-the-cycle. The impact of a range of fuel burnups on the non-dimensional release fractions to the containment provides an insights into the time-in-the-cycle

variations. Some of the key results from reformulation of the Surry UA results into NUREG-1465 accident phases is presented in the following sections.

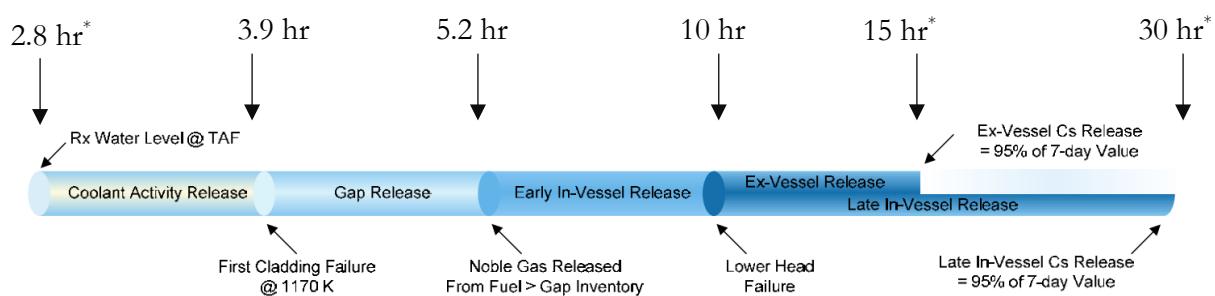
2.3.3.1. Comparison to NUREG-1465

Table 7 shows the mean, standard deviation, minimum, 5th, 25th, 50th, 75th, and 95th percentile, and maximum values of the Surry UA accident phase start times. While NUREG-1465 does not explicitly provide start times for each phase, the Surry UA results provide timing information. As seen in the table, the start timings between the 5th and 95th percentiles for all phases are relatively close, which means a small variability in the start times for the majority of Surry UA runs, which is somewhat expected since all results are from the same sequence (i.e., a station blackout accident without any onsite power). The mean timings are shown on Figure 2.

Table 7 Accident phase start times (hours)

Metrics	Gap	In-Vessel	Ex-Vessel	Late In-Vessel
NUREG-1465	n/r	n/r	n/r	n/r
Mean	3.9	5.2	10	10
Standard Deviation	2.6	3.1	6.3	6.3
Minimum	2.7	3.2	7.5	7.5
5%	3.3	3.9	8.6	8.6
25%	3.5	4.3	8.9	8.9
50%	3.7	5.0	9.2	9.2
75%	3.7	5.3	9.5	9.5
95%	4.0	5.7	10	10
Maximum	26	31	70	70

n/r = not reported.



* All timing based on the Surry PWR SOARCA UA timings based Table 7 except the time to the top of the active fuel (TAF) [17], the ex-vessel phase duration (Table 6), and the duration of the late in-vessel duration (Figure 4).

Figure 2 Release phase mean timings using Surry UA short-term station blackout results [18].

Table 8 shows a comparison of the noble gases releases by phase, which are represented in MELCOR as xenon (Xe) radionuclide class. Xenon is a non-condensable gas, which is used to identify the start and end of the gap release phase. Some xenon forms in the gap between the fuel and the cladding in the fuel rods. The start of the gap phase is when the first fuel cladding failures occur (i.e., at the beginning the core degradation process). The released xenon circulates through the primary system and enters the containment with the gas discharged to the PRT via the pressurizer safety valve (SV) cycling.¹ A 5% release of xenon to the containment is used to identify a non-trivial release of radionuclides from the fuel (i.e., the end of the gap phase). Since the released xenon remains mobile as a gas, it is the best radionuclide to identify the start of the release phase from the fuel. Xenon is also released from the fuel during the in-vessel core degradation. This is shown in the NUREG-1465 value and the Surry UA results data, which indicate ~100% of xenon inventory is released during the gap and in-vessel phases.² Unlike the phase durations, the xenon release fractions have a tight distribution meaning there is less uncertainty.

Table 8 Comparison of the NUREG-1465 and Surry UA fractional xenon releases to the containment for the four phases

Metrics	Gap	In-Vessel	Ex-Vessel	Late In-Vessel
NUREG-1465	0.05	0.95	0.0	0.0
Mean	0.052	0.93	0.012	0.0021
Stand. Dev.	0.0071	0.024	0.020	0.0025
Minimum	0.050	0.74	0.0003	0.0000
5%	0.050	0.90	0.0024	0.0000
25%	0.050	0.93	0.0056	0.0004
50%	0.051	0.93	0.0084	0.0013
75%	0.052	0.94	0.012	0.0028
95%	0.054	0.94	0.021	0.0063
Maximum	0.16	0.95	0.21	0.021

Table 9 shows a comparison of the alkali metals releases by phase, which are represented by the various forms of cesium. NUREG-1465 shows a distributed release over the in-vessel and ex-vessel phases, while the Surry UA release of cesium occurs almost entirely during the in-vessel phase. The Surry UA model tracks three chemical forms of cesium (i.e., CsOH, CsI, and Cs₂MoO₄). Although the volatility of the chemical forms vary, MELCOR predicts the in-vessel degradation in the relatively fast-progressing STSBO will release most of the volatile inventory, which includes xenon, cesium, and iodine. Consequently, a key insight is the significantly larger release during the in-vessel phase in the Surry UA than reported in NUREG-1465.

¹ The alternate pathways for release to the containment are not active at this phase of the accident (e.g., reactor coolant pump (RCP) seal leakage, the hot leg failure, or the lower head failure). At this time, the leakage through the RCP seals is liquid water. Once the xenon reaches the pressurizer relief tank (PRT), it is considered in the containment for the purposes of tracking the source term. The PRT rupture disk will open after a few SV cycles and release the xenon to the containment.

² The Surry UA assumed 5% of the xenon is in the gap. The remaining xenon is released during the in-vessel phase due to thermal-diffusion processes from the fuel matrix.

Table 9 Comparison of the NUREG-1465 and Surry UA fractional cesium releases to the containment for the four phases

Metrics	Gap	In-Vessel	Ex-Vessel	Late In-Vessel
NUREG-1465	0.05	0.25	0.35	0.1
Mean	0.015	0.74	0.011	0.0095
Stand. Dev.	0.0087	0.12	0.017	0.0095
Minimum	0.0000	0.16	0.0004	0.0000
5%	0.0065	0.53	0.0023	0.0032
25%	0.0088	0.74	0.0054	0.0055
50%	0.014	0.77	0.0081	0.0075
75%	0.019	0.80	0.011	0.010
95%	0.025	0.83	0.019	0.019
Maximum	0.10	0.84	0.17	0.088

Table 10 shows a comparison of the halogen releases by phase, which are represented by the various forms of iodine (i.e., I_2 gas and cesium iodide). Like the cesium release results, NUREG-1465 shows the iodine release is distributed over the in-vessel and ex-vessel phases, while the Surry UA release of iodine occurs almost entirely during the in-vessel phase.

Consequently, the majority of the xenon, cesium, and iodine releases occur during the in-vessel phase in the Surry UA. Although there is agreement on the xenon releases with NUREG-1465, NUREG-1465 shows a more balanced release of cesium and iodine between the in-vessel and ex-vessel phases.

Table 10 Comparison of the NUREG-1465 and Surry UA fractional iodine releases to the containment for the four phases

Metrics	Gap	In-Vessel	Ex-Vessel	Late In-Vessel
NUREG-1465	0.05	0.35	0.25	0.1
Mean	0.020	0.80	0.016	0.031
Stand. Dev.	0.0084	0.11	0.0090	0.047
Minimum	0.0004	0.24	0.0014	0.0016
5%	0.011	0.58	0.0051	0.0079
25%	0.014	0.80	0.011	0.015
50%	0.018	0.83	0.015	0.020
75%	0.024	0.86	0.019	0.029
95%	0.032	0.88	0.036	0.064
Maximum	0.082	0.89	0.057	0.36

Finally, SOARCA and the Surry UA calculations track all radionuclides shown in Table 4. This example only shows three radionuclides but could be easily expanded to all the values shown in NUREG-1465. Cesium and iodine were selected because they have important health consequences and were included in the key figures of merit in the SOARCA UA studies. The xenon results are interesting because they show the start and end of the gap phase and occur without any deposition in the primary system.

2.3.3.2. Horsetail Results

An alternate way to show the Surry UA results is through horsetail plots. The horsetail plot illustrates the variability in all the results in a manner not readily observable in the table values. The time-histories of the iodine releases are shown in Figure 3 and illustrate the variability and uncertainty in the Surry UA predictions. The iodine releases start quite early in the accident, which is shown in various statistical values in Table 10. While the horsetail figures show the same data as the tables, it allows for meaningful qualitative insights of the time-histories that is not possible with the end-values presented in the tables. The markedly different 0.5 day time-in-the-cycle results are clearly evident as the slowly developing results.

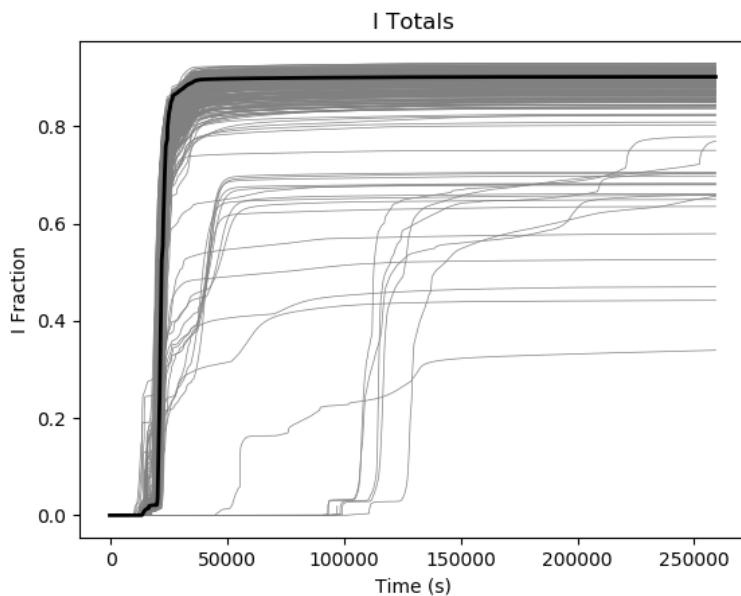


Figure 3 Example of a horsetail plot for total iodine fraction in containment as a function of time

2.3.3.3. Histogram Results

Another way to represent the Surry UA data is through histograms. The histograms visually show the distribution for a particular quantity of interest (such as timings, duration, or fission product release fractions) for each phase. A narrow distribution suggests a lower amount of uncertainty in the metric (e.g., phase start time, duration, or magnitude of the release) while wide distributions suggest higher variability and uncertainty. The histogram is an effective way of visualizing the percentiles shown in the tables. As an example, a histogram of late in-vessel duration is shown in Figure 4. The x-axis shows the duration of the phase while the y axis shows the number of

simulations per bin (bin width \sim 14 mins). If the majority of simulations are in a few bins, then the distribution is narrow; and if the simulations are spread out in many bins, then the distribution is wide. The distribution shown in Figure 3 is quite wide, spanning \sim 10 hr, which is confirmed in Figure 4. This corresponds to a large variability or uncertainty in the duration of the late in-vessel phase. The duration reported in NUREG-1465 for late in-vessel phase is much lower than predicted for the Surry UA. Some of the variability in the Surry UA results arise from the time-in-the-cycle sampling.

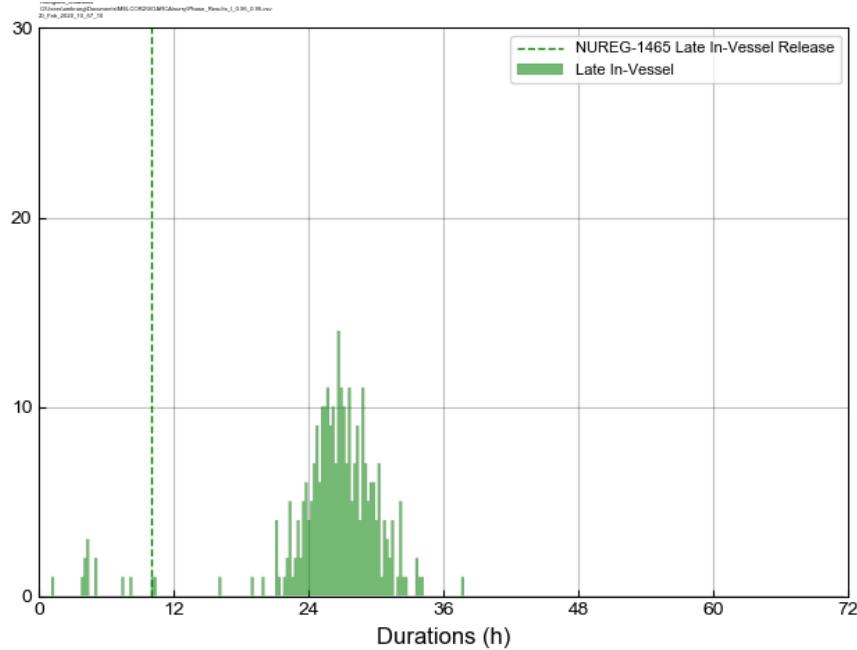


Figure 4 Example of a histogram describing the variation in late in-vessel duration

This data can help better understand the conservatisms built into regulatory guidance, such as release fraction data in NUREG-1465.

2.4. Historical Severe Nuclear Accidents

The magnitude of a bounding source term can also be informed from historical severe accidents. The following sections discuss three of the most important severe accidents; the TMI-2, Chernobyl Unit 4, and Fukushima Dai-ichi nuclear accidents.

2.4.1. Historical Accidents

The IAEA publishes a guide to characterize the severity of nuclear power accidents and incidents. The guide, known as the International Nuclear and Radiological Event Scale (INES) was developed to convey the severity of any event related to radiation risks [19]. It presents criteria for rating any event associated with radiation and radioactive material. Levels 1-3 are incidents and Levels 4-7 are accidents. The accident categories are described as follows for reactor accidents,

- 4 = Accident with Local Consequences – Fuel melting with $>0.1\%$ release of the core inventory.
“An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of tens to hundreds of terabecquerels of ^{131}I .”

- 5 = Accident with Wider Consequences – Severe damage of the entire core with release of radioactive material within the installation with high probability of public exposure.
“An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of hundreds to thousands of terabecquerels of ^{131}I .”
- 6 = Serious Accident – Significant release of radioactive material likely to require implementation of planned counter measures.
“An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of thousands to tens of thousands of terabecquerels of ^{131}I .”
- 7 = Major Accident – Major release of radioactive material with widespread health and environmental effects requiring implementation of planned countermeasures.
“An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of more than several tens of thousands of terabecquerels of ^{131}I .”

Categories 4-7 are designed to describe the amount of radioactive material released rather than the dose received by the public. The scale uses the concept of “radiological equivalence” in terms of terabecquerels of ^{131}I . The integral activities of other released radionuclides are converted to an equivalent ^{131}I activity through conversion factors.

The Three Mile Island (TMI) accident is an example of Category 5 accident. While there was severe damage to the core, the reactor vessel and the containment remained intact and the release was limited to primarily noble gases. In contrast, the Chernobyl and Fukushima Dai-ichi nuclear reactor accidents are examples of Category 7 accidents. Fukushima’s Category 7 rating considers the simultaneous accidents at Units 1, 2 and 3 as a single event on INES scale and uses estimated total release from all three reactors to the atmosphere as a justification. Otherwise, INES Level 5 ratings had been applied for Fukushima Units 1, 2 and 3 individually. The Japanese Nuclear and Industrial Safety Agency (NISA) estimated the releases from Units 1, 2, and 3 were 1.3×10^{17} Bq of iodine-131 and 6.1×10^{15} Bq of cesium-137, which was equivalent to approximately 10% of the amount of the radioactive materials released by the Chernobyl accident [20].

In contrast to US and the Fukushima Dai-ichi commercial reactors, the Chernobyl reactor did not have a steel and/or reinforced concrete, leak-tight containment to retain fission products. The graphite-moderated Chernobyl core design also had a positive void reactivity that increased power as the accident progressed. The resulting reactivity insertion induced a rapid disassembly and melting of the fuel, which was followed by an explosion that destroyed the reactor confinement building. As will be described in Section 2.4.2, there were important upgrades made to RBMK designs like Chernobyl to address some of the key issues leading to the accident.

The TMI-2 and individual Fukushima reactor accidents have the most likely characteristics expected in a postulated LWR severe reactor accident with a hardened, containment. If the containment remains intact with emergency actions, an INES category accident 5 is expected (e.g., TMI-2 consequences). A TMI-2-type severe accident includes eventual mitigation of the accident progression where the containment remains intact. The releases from containment would be very small and within the bounds of an INES Category 5 accident. Modern emergency preparedness in the US or a country using similar IAEA guidelines includes staging of national resources to aid and manage severe accident as well as additional onsite safety equipment with engineered external plant

connections. Section 2.4.2 discusses the post-Fukushima emergency planning and addition resources that were developed to address issues observed during the Fukushima accidents.

Despite world-wide efforts to improve nuclear safety following the TMI-2 and Fukushima accidents, a more severe accident without any successful mitigation is possible although very unlikely.

Examination of the consequences from Fukushima provide examples of accidents with limited mitigation. Sandia has performed forensic analysis of Fukushima accidents using MELCOR [21][22]. The Fukushima Unit 1 releases are well-predicted by the SOARCA unmitigated short-term station blackout (STSBO) calculations performed for SOARCA [14] and the subsequent Peach Bottom UA [15]. However, the Fukushima Units 2 and 3 were more complex accidents that included limited operability of reactor injection systems that changed the accidents into long-term station blackouts (LTSBOs) as characterized in SOARCA. Nevertheless, the SOARCA and UA results also provide insights into the maximum severe accident that includes no mitigation through containment failure, containment bypass, or increased containment leakage.

A more severe accident would be a containment bypass accident that is similar to the SOARCA interfacing-system loss-of-coolant accident (ISLOCA) [13]. The ISLOCA damage progression occurs with any radionuclide attenuation or delay in the containment. The start of the radionuclide release from an ISLOCA could be relatively fast (<24 hr) and includes 16% of the volatile iodine and 2% of the volatile cesium. In contrast, the results from the SOARCA PWR station blackouts with containment failure were <2% of the volatile species (i.e., cesium and iodine). For perspective, the TMI-2 release was <<1% of the gaseous iodine (i.e., from an intentional noble gas release over one year later as part of the long-term accident management).

2.4.2. Characterization of the Chernobyl Source Term

One proposed option for the bounding source term would be the release from the Chernobyl accident. This certainly has direct relevance to the RBMK-1000 design, which has significant differences from PWR and BWR designs. While the Chernobyl accident is considered incredible today after modifications to RBMKs, it is nevertheless an important result. The Nuclear Energy Agency (NEA of the Organization for Economic Cooperation and Development (OECD) organization published a consensus estimate of the source 10 years after the accident [23]. The Chernobyl accident is unique in many ways including,

- lack of a hardened containment,
- graphite reflectors that burned and contributed to a long-term heat source, and
- an explosive initiator that disrupted the core.

The accident progressed with a prompt release of fission products due fuel fragmentation and energetic lofting. Reference [23] estimates that 6.7 mt (3.7%) of the fuel mass was dispersed beyond the station boundary. The burning graphite and fuel decay heat contributed to oxidizing radionuclide releases that also occurred for a week after the initiating event.

The radionuclide release occurred in two high release phases. The first release associated with the explosive (i.e., believed to be a steam explosion following the interaction of molten fuel with the water following the rapid power escalation). The second release occurred about a week later due to a delayed oxidation phase. After about 9 to 10 days, the releases diminished significantly. The first phase is unexpected (i.e., a steam explosion that energetically disperses fuel and fission products). Due to the explosion and disruption of the core, the radionuclide release during the thermal oxidation phase, which is expected, was much later (~1 week after the accident initiation).

Reference [23] notes that the release of fuel as fragments unexpectedly carried low volatility elements (e.g., cerium, zirconium, and actinides), which are not typically released in such large quantities. The chemical form of iodine has been estimated with varying amounts of gaseous iodine. Researchers reported values of 60 to 80 percent in gaseous forms of iodine. The updated consensus estimate of the release is summarized in Table 11.

Finally, the authors did not find any information that suggested the Chernobyl Unit 4 SFP or any other unit's reactors or SFP had radionuclide releases. Figure 5 from Reference [25] shows two SFPs (north and south) at Unit 4 that appear undamaged.

Table 11 Chernobyl Source Term [23].

Radionuclide	Release Fraction
Noble gases (Kr, Xe)	100%
Iodine	50-60%
Tellurium	25-60%
Cesium	33-43%
Barium and Strontium	4-6%
Ruthenium and Molybdenum	>3.5%
Zirconium, Neptunium, Plutonium, Cerium, and Curium	3.5%

The average burnups of the fuel in the core was estimated to be 10,912 MWd/t. From the fuel burnups, the radionuclide activity was calculated by 7 researchers in Reference [23]. Using the estimated release fractions, the activity released into the air was 85 (\pm 26) PEBq (1 PEBq = 10^{15} Bq) of Cs 137 and 1.1 to 1.3 EBq (1 EBq = 10^{18} Bq) of I-131. These are two most dominate radionuclides for long-term and short-term health consequences, respectively.

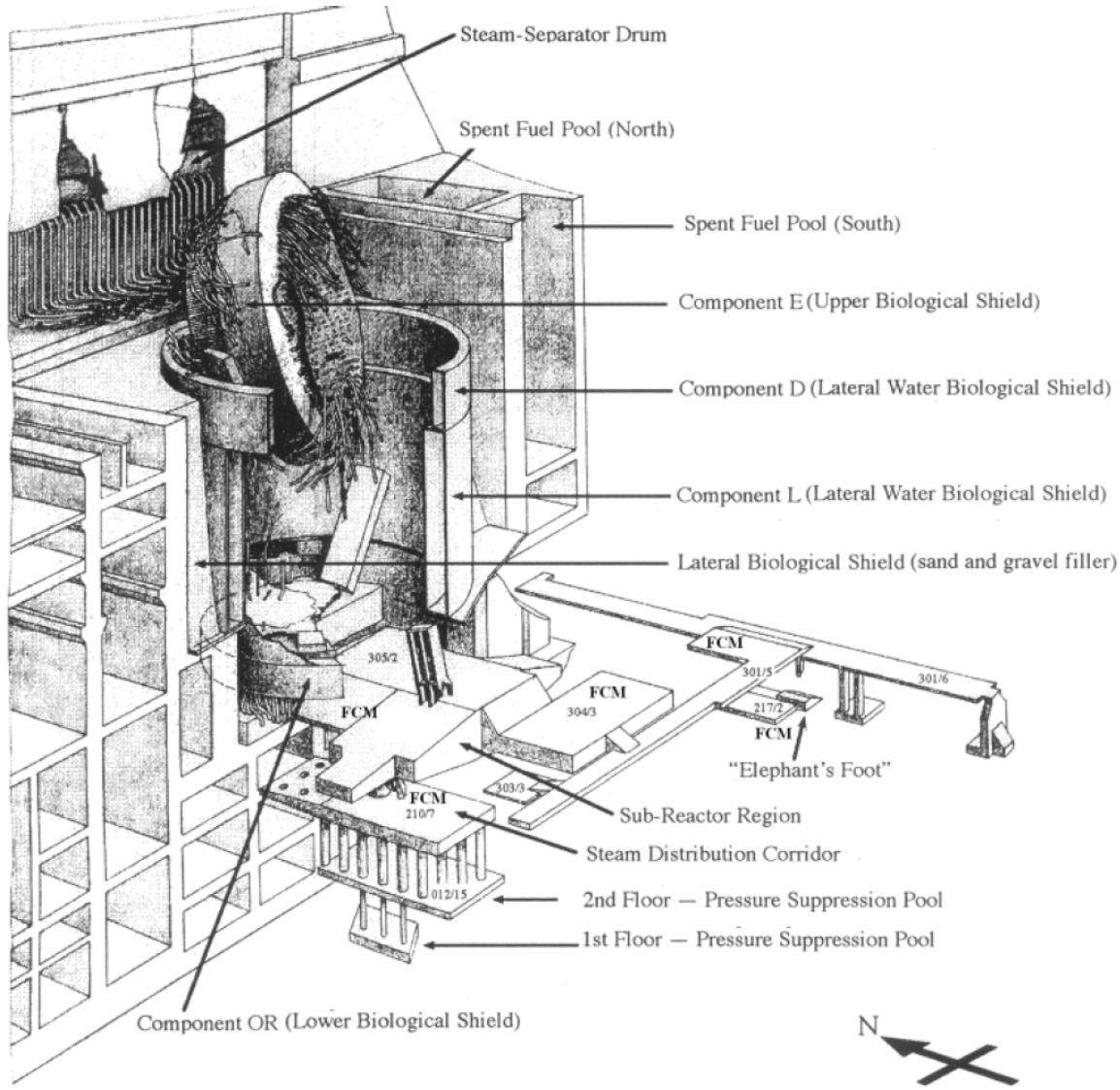


Figure 5 Chernobyl Unit 4 post-accident configuration showing SFP [24].

2.4.3. *World-wide Post-Accident Plant Updates*

Regulatory agencies across the world are continuously updating regulations and responding to evolving issues. Notable and significant changes occurred following the TMI-2, Chernobyl Unit 4, and Fukushima Dai-ichi nuclear accidents, which are briefly described in the following sections. These changes are important because they impact the severity of a bounding accident through design changes (e.g., RMBK core reactivity changes, BWR Mark I hardened vent), onsite emergency equipment, severe accident management procedures, and required operator training. A significant development world-wide following corrective actions from the Fukushima Dai-ichi nuclear is the ability to mitigate a long-term loss-of-power scenario. A summary of the key regulatory actions after the TMI-2, Chernobyl Unit 4, and Fukushima Dai-ichi nuclear accidents are summarized in the following sections.

2.4.3.1. TMI-2 Action Plan

After the TMI-2 accident, the NRC ordered a comprehensive number of changes for all current and future plants licensed in the US. These requirements are outlined in NUREG-0737 [25]. Notable required updates include staffing requirements and training, additional control room parameter displays, core level measurements, severe accident training procedures, additional valve position instrumentation, maintenance and testing requirements, training control room simulators, control room habitability requirements, and radiation monitoring. These requirements and upgrades were shared with western regulators, who incorporated similar requirements for their plants (i.e., many that were supplied by US vendors).

2.4.3.2. Post-Chernobyl Plant Upgrades

As discussed in Section 2.4.2, the Chernobyl reactor had unique design attributes that are different than other LWRs such as PWRs, BWRs, and VVERs. The design weaknesses identified in the post-Chernobyl regulatory reviews led to changes that improved the safety of the operating fleet of RBMKs [26]. The most notable changes included,

- Reduction of the void coefficient of reactivity,
- Improvement of the response efficiency of the emergency protection system,
- Prevention of the emergency safety systems from being bypassed while the reactor is operating,
- The installation of 80-90 additional fixed absorbers in the core to inhibit operation at low power, and
- retrofitting of the control rods with a design for augmenting the reactivity worth of the rod.

These design changes provided important additional protections of RBMKs to reactivity insertion accidents like Chernobyl Unit 4 accident.

2.4.3.3. Post-Fukushima Orders

After the accident at Fukushima Dai-ichi, the NRC established the Near-Term Task Force to review the accident and make recommendations to improve reactor safety for power plants in the US. The Near-Term Task Force issued SECY-11-0093 [27], which provided recommendations to the NRC on the lessons learned. This led to the NRC issuing three orders:

- Mitigation Strategies Order EA-12-049 [28],
- Spent Fuel Pool Instrumentation Order EA-12-051 [29], and
- Severe Accident Capable Hardened Containment Vent System Order EA-13-109 [30].

The Nuclear Energy Institute (NEI) provided guidance for US nuclear power plants to address post-Fukushima regulations in Reference [32] for Mitigation Strategies Order EA-12-049 and Reference [33] for the Spent Fuel Pool Instrumentation Order EA-12-051³. NEI guidance for addressing Mitigation Strategies Order EA-12-049 follows three steps: (1) establish a baseline coping capability, (2) determine applicable extreme external hazards, and (3) define site-specific Diverse and Flexible Coping Strategies (FLEX) capabilities.

Although the FLEX mitigation strategy is a US regulatory requirement, the concepts are embraced throughout the world including the Western European Nuclear Regulators Association [34], the

³ Reference [28] is for the US requirement. The IAEA has a similar reference in the No. SSR 2/1 (Rev 1) [31].

VVER Work Group [35], the International Atomic Energy Agency (IAEA) [36], China [37], Russia [38], etc.

3. SPENT FUEL POOL BOUNDING ACCIDENTS

The spent fuel is often stored in two locations at nuclear power plants. The recently offloaded fuel is stored in a spent fuel pool (SFP) adjacent to the reactor. During refueling activities, some of the fuel is permanently offloaded into the SFP. The SFP is in very close proximity to the containment to facilitate efficient movement during refueling operations. Depending on the specific fuel cycle at the plant, 33% to 50% of the fuel may be offloaded every 1 to 2 years. The variability is due to the power strategy and the fuel enrichment. Once the fuel has cooled in the SFP for approximately 5 years due to decay of the shorter lived radionuclides, it can be transferred to dry storage casks or shipped off for reprocessing. Most plants delay the spent fuel transfer to dry casks until the SFP racks are almost full.

It is judged that the older spent fuel stored in dry casks would not be susceptible to a significant release. The fuel is air-coolable and the casks are ruggedly designed for a wide range of postulated impacts and events. In contrast, the recently discharged fuel assemblies in the SFP are susceptible to overheating if the SFP water drains or boils away.

SFP accidents carry the additional risks of no surrounding concrete-reinforced containment and may have multiple cores worth of fuel. However, SFP accidents progress slowly due to the low decay heat power and the hazardous I-131 nuclide decays away due to its short half-life of 8.03 days. On The walls of the SFPs are constructed with thick concrete walls for strength and radiation shielding.

The NRC performed a detailed review of SFP accidents at a Mark I BWR plant [39]. The study compares high-density and low-density fuel loading strategies and assesses the benefits of post-9/11 mitigation measures. The post-9/11 mitigation measures include portable sprays for the SFPs, multiple methods for injection, and a well-organized dispersion of the recently offloaded fuel assemblies with the older and lower-powered fuel assemblies. The NRC SFP study illustrates how the post-9/11 mitigation measures are helpful to mitigating accidents in a variety of configurations. The study also performed calculations where the mitigation measures were not performed and the benefits of moving all the older fuel in the SFP to dry casks (i.e., not typically done)⁴.

The response of the fuel in the SFP varies significantly based on the time since the last offload. For example, the last offload often represents the majority of the total SFP decay heat power (e.g., up to 80%) although it may only be a small percentage of the total fuel (e.g., ~10%). The NRC SFP study characterized the results in terms of the plant's operational cycle phase (OCP). The OCP identifies the configuration of the SFP relative to refueling and power operation activities. Since the decay heat power of the last offloaded fuel dominates the spent fuel pool total decay power shortly after refueling, the OCP phase provides an indication of the fuel vulnerability and the rate of the accident progression. The connection of the SFP to other transfer water pools also varies across OCPs.

The NRC study calculated radionuclide releases for scenarios without mitigative measures during three OCPs. The OCPs included the refueling OCP as well as OCPs spanning to 60 days after the start of refueling. Seven radionuclide release scenarios with high-density fuel loading were examined. The scenarios varied the leak rate and OCPs. The cesium releases ranged from 0.6% to 42% (see Table 12). The biggest factor influencing the magnitude of the release was whether a hydrogen burn failed the reactor building. Hydrogen is generated when steam inside the reactor building room reacts with the hot zircaloy fuel cladding. The released hydrogen is highly susceptible to burning if

⁴ Plants typically just transfer enough spent fuel to maintain free rack cells for an emergency offload of the entire core. A typical SFP configuration uses high-density racks and the SFP is filled near capacity. This differs the costs of purchasing dry casks.

there is adequate oxygen in the room.⁵ If the reactor building remains intact, then the fuel cladding oxidation reaction consumes the oxygen in the refueling room. The subsequent oxygen-limited reaction slowly creates a thick oxide layer without significantly heating the fuel. The peak fuel temperature in these oxygen-limited sequences remained relatively low and the radionuclide releases were small. However, if the building ruptured and allowed a continuous supply of fresh air into the SFP, then the heat addition from the air oxidation reaction dominated the heating of the fuel. In these circumstances, the fuel heated to very high temperatures and released most of the volatile cesium to the environment (i.e., 17.1% and 42%).

There was a secondary factor in the status of the reactor building that further impacted the results. If the building remained intact, then the intact building offered 66% to 75% retention of the released fission products. Although the building was intact, the pressurization from hydrogen production and heating of the air led to a steady leakage of radionuclides from the building. In contrast, if the reactor building ruptured due to a hydrogen burn, then the retention in the reactor building was only 10-20%.

A scenario with failed reactor building and an overall 17% release to the environment is shown in Figure 6. This accident occurred in OCP2, which is between 8 to 25 days after the start of refueling operating (i.e., 13 days was selected for the analysis). It was a small leak with no mitigation. Each ring in Figure 6 represents a batch of fuel with similar discharge characteristics. Relative to the cesium release, there is no appreciable change in Cs-137 inventory across all of the fuel assemblies in the SFP. The environmental release of Cs-137 primarily impacts long-term land contamination, which is the greatest concern from an SFP accident. Ring 1 leads the temperature heatup and represents 88 highest-powered assemblies of the 284 most recently offloaded assemblies. There is a total of 3055 assemblies in the SFP. After the Ring 1 assemblies started an accelerated oxidation phase, the surrounding assemblies in Ring 2 (352 fuel assemblies) also overheated and released large amount of cesium. The remaining fuel assemblies in the other locations had smaller releases (i.e., <17%) of their cesium inventory. Consequently, the overall release is not 37% (i.e., the maximum in Ring 1) but there is the weighted average of the number of assemblies per ring times the ring release fraction. The total overall cesium release to the environment was 17%. However, it could be argued that the release at 72 hr is still increasing (i.e., the termination of the calculation).

The severity of the SFP accident is a strong function of the availability of oxygen when the fuel first heats above 1100 K. The variations in releases of the different cases show the sensitivity to the timing of the hydrogen burn that ruptures the reactor building walls to the timing of the heatup. Another key attribute is when an air natural circulation can occur through the racks. Once the water level drains or boils below the bottom of the rack base plate, air can naturally circulate through the spent fuel assemblies. This almost always has a cooling effect that ends the potential for more releases. However, there are instances where the circulation of fresh air into hot assemblies leads to a rapid oxidation response. This occurs if the fuel was hot and the existing oxide layer was not too thick when the air natural circulation through the racks starts. The sequence in Figure 6 had these attributes with the water level clearing the base plate at ~62 hr when the peak fuel temperatures had just approach 1200 K in the hottest location (i.e., Ring 1 fuel).

⁵ Although three reactor buildings ruptured following hydrogen burns in the Fukushima Dai ichi accident, it was due to hydrogen leakage into the reactor building from the reactor accident. All fuel in the SFPs remained covered by water throughout the accident.

To summarize the findings from the 6 cases without mitigation with a source term, the Cesium releases at 72 hr are 0.6%, 0.7%, 1.5% 1.6%, 17.1%, and 42% across OCP1 through OCP3. The estimate for timing for the uncover of the fuel for the start of fission product releases is shown in Table 13. If there is no leak, then the decay heat from the fuel must boil away the SFP water, which will take more than a week. A small leak is defined as a tear in the SFP stainless steel liner (i.e., initial leakage at \sim 200 gpm), such that the leakage rate is limited to the capacity of the leak detection passageways below the racks. A moderate leak is a through-wall crack in the concrete that is controlled by the size of the crack. The moderate leak in this study starts at 1500 gpm when the water level is 16-ft water level above the SFP floor. The leakage rate for both size leaks decreases as the SFP water level decreases.

Table 12 Summary of the NRC SFP study Cs-137 radionuclide releases [39].

High Density Case #	Scenario Characteristics					Release Characteristics			
	SFP Leakage?	50.54(hh)(2) Equipment?	Fuel Uncovery (hr)	Gap Release (hr)	Hydrogen Deflagration (hr)	Cs release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP1	None	Yes							
	None	No							
	Small	Yes							
	Small	No	39.7	54.2	No	0.6%	0.33	3.5%	0.27
	Moderate	Yes	7.4	15.1	No	0.5%	0.26	5.0%	0.39
	Moderate	No	5.9	8.7	No	1.5%	0.80	2.1%	0.16
OCP2	None	Yes							
	None	No							
	Small	Yes							
	Small	No	42.6	60.5	64.8	17.1%	7.90	17.1%	1.91
	Moderate	Yes							
	Moderate	No	5.9	11.6	No	1.6%	0.73	2.0%	0.22
OCP3	None	Yes							
	None	No							
	Small	Yes							
	Small	No	18.7	40.6	47.3	42.0%	24.20	51.2%	0.73
	Moderate	Yes							
	Moderate	No	2.5	16.9	No	0.7%	0.39	0.7%	0.01

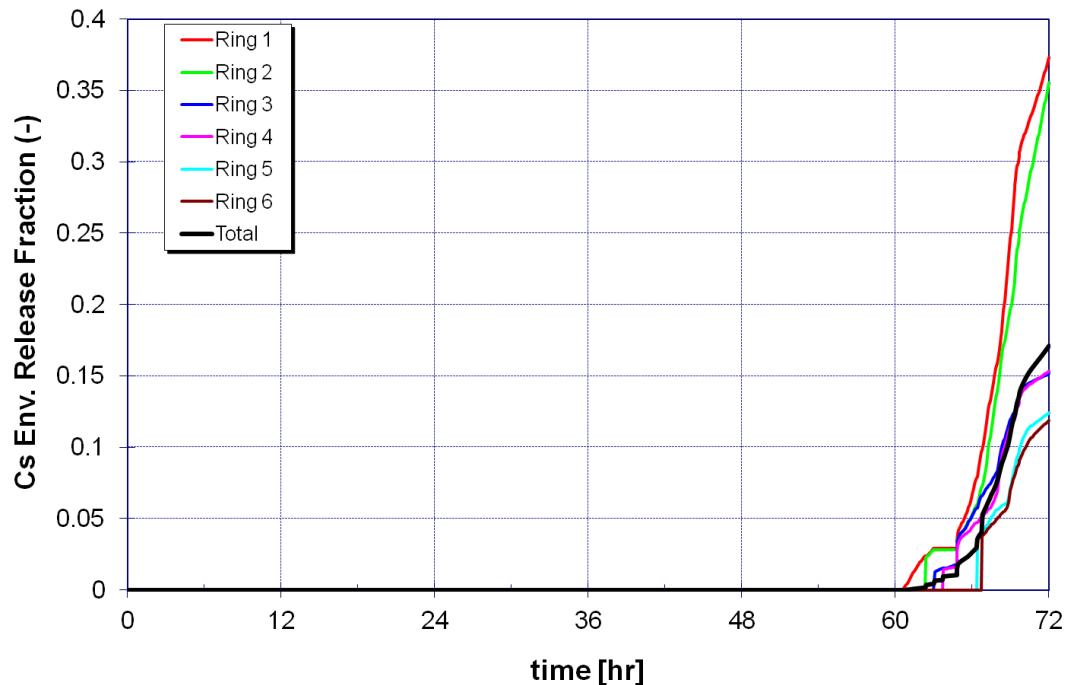


Figure 6 A severe SFP source term [39].

Table 13 Approximate Time of Fuel Uncovery [39].

Time	Days after the start of refueling	No Leak	Small Leak	Moderate Leak
OCPs 1 and 2	2 – 25 days	> 7 days	40 hours	6 hours
OCPs 3, 4, and 5	>25 days	> 7 days	19 hours	2.5 hours

4. REACTOR DESIGN CHARACTERISTICS

This review considers the following commonly deployed reactor designs in order of interest:

- RBMK-1000,
- VVER-440,
- VVER-1000,
- 1000 MWe PWR,
- 1000 MWe BWR,
- BN-800, and
- 600 MWe CANDU/PHWR.

A brief review of the key design attributes is provided in Table 14 that may have first order effects on the magnitude of a bounding source term. The thermal-power rating is a first-order factor in the magnitude of the radionuclide inventory. In lieu of plant or design-specific radionuclide inventory, the scaling using an end-of-cycle PWR high burn-up inventory is a reasonable approximation of radionuclide inventory. This is a reasonable approximation considering other uncertainties. For example, the plant-specific assembly loading, burn-ups, and time-in-the cycle are variable and can vary the inventory more variability than an overall thermal power scaling. Figure 8 illustrates how the two of the dominant health consequence isotopes vary across a PWR cycle from a detailed radionuclide inventory evaluations.

The second column in Table 14 provides an indication of the containment robustness. A reinforced-concrete containment provides significant protection against severe events. Containments typically have design pressures on the order of 40-50 psig and an ultimate failure pressure of more than twice that value [17]. The containment leakage rating is typically 0.1% mass/day, which is a regulated technical specification. Even when the stainless steel liner of a reinforced-concrete containment fails, the leakage rate gradually increases [17]. Reinforced concrete containments are designed to provide protection against wind and missiles from hurricanes and tornadoes, seismic events, and heavy snows. In contrast, a confinement is a relatively low-leakage industrial building that has a design pressure rating of <1 psig. For example, the design pressure of a BWR reactor building (i.e., typical of a confinement structure) has a design pressure of 0.25 psi [40]. The Russian RBMK-1000s, VVER-440s, and BN-800 use confinements. A spent fuel pool would also be located in a confinement structure.

The fourth column in Table 14 illustrates several important points on the bounding source term from a particular nuclear power plant site. First, most of the sites have multiple reactors. A severe external initiating event at one reactor is likely to impact multiple reactors. Furthermore, although Novovoronezh only has one VVER-1000, it also has two VVER-440s at the site (i.e., a multi-unit site). The second important observation is that each site with multiple reactors will have multiple SFPs. Since these plants are older, their spent fuel pools are filled with at least five years of fuel discharges (i.e., >1.5 full cores) and likely filled to capacity (i.e., common in the US⁶). Consequently, multiple SFPs could be involved at each location.

Even BN-800, the most unusual reactor in this list, has another sodium fast reactor on site (i.e., a BN-600), which is the smaller predecessor to the BN-800 reactor. Both reactors use a confinement around the reactor with concrete reinforcement around the sides of the reactor (see Figure 7). The BN-600 has been in operation since 1980 [41], which suggest significant spent fuel.

⁶ NUREG-2161 identifies 2859 assemblies in the Peach Bottom SFP, which corresponds to ~3.7 cores.

Table 14 Key Design Attributes.

Reactor	Rated Thermal Power [MW]	Containment or Confinement	Multi-unit	Moderator or working fluid
RBMK	3200	Confinement Building	Kursk x 4 Leningrad x 2 Smolensk x 3	H ₂ O Graphite-moderated
VVER-440	1375	Confinement Building	Bohunice x 2 Dukovany x 4 Kola x 4 Loviisa x 2 Metsamor x 1 Mochovce x 2 Rivne x 2 Novovoronezh x 2 Paks x 4	H ₂ O
VVER-1000	3000	Reinforced Containment	Balakovo x 4 Bushehr x 1 Kalinin x 4 Khmelnitskyi x 2 Kozloduy x 2 Novovoronezh x 1 Rivne x 2 Rostov x 2 South Ukraine x 3 Temelin x 2 Zaporizhzhia x 6	H ₂ O
PWR	3000	Reinforced Containment	Unspecified, CPR-1000s are at: Fangchenggang x 2 Ningde x 4 Ling Ao x 2 Hongyanhe x 4 Yangjiang x 2	H ₂ O
BWR	2923	Steel containment in reactor building	Unspecified.	H ₂ O
BN-800	2100	Confinement building	Beloyarsk x 1	Sodium
CANDU	1700	Reinforced Containment	Embalse x 1 Wolsong x 3 Cernavodă x 2 Tarapur x 2	D ₂ O
Notes:				
1. RBMK-1000 reference [26].				

Table 14 Key Design Attributes.

2. VVER-440 is primarily based on Models 213 and 230. Novovoronezh also has a VVER-1000 onsite. Rivne has 2xVVER-1000s onsite. Loviisa units have a reinforced concrete containment [43].
3. VVER-1000 is based on Models 320. The advanced AES 91 & 92 models at Tianwan and Koodankulam are not included. Novovoronezh also has a 2 x VVER-440s onsite. Rivne has 2xVVER-440s onsite [43].
4. 1000 MWe PWR is assumed to be similar to the 3-loop CPR1000 [44].
5. 1000 MWe BWR is assumed to be similar to the Brunswick General Electric BWR/4 Mark I [45].
6. BN-800. Beloyarsk nuclear power station also has BN-800 nuclear power plant onsite [46].
7. CANDU 600 MWe thermal power scaled from enhanced CANDU 6 EC8 rated at 730 MWe. There are 19 CANDU reactors in Canada and 16 smaller CANDUs or CANDU-derived reactors in India not included above. Tarapur also has 2 x BWRs at the site [47].

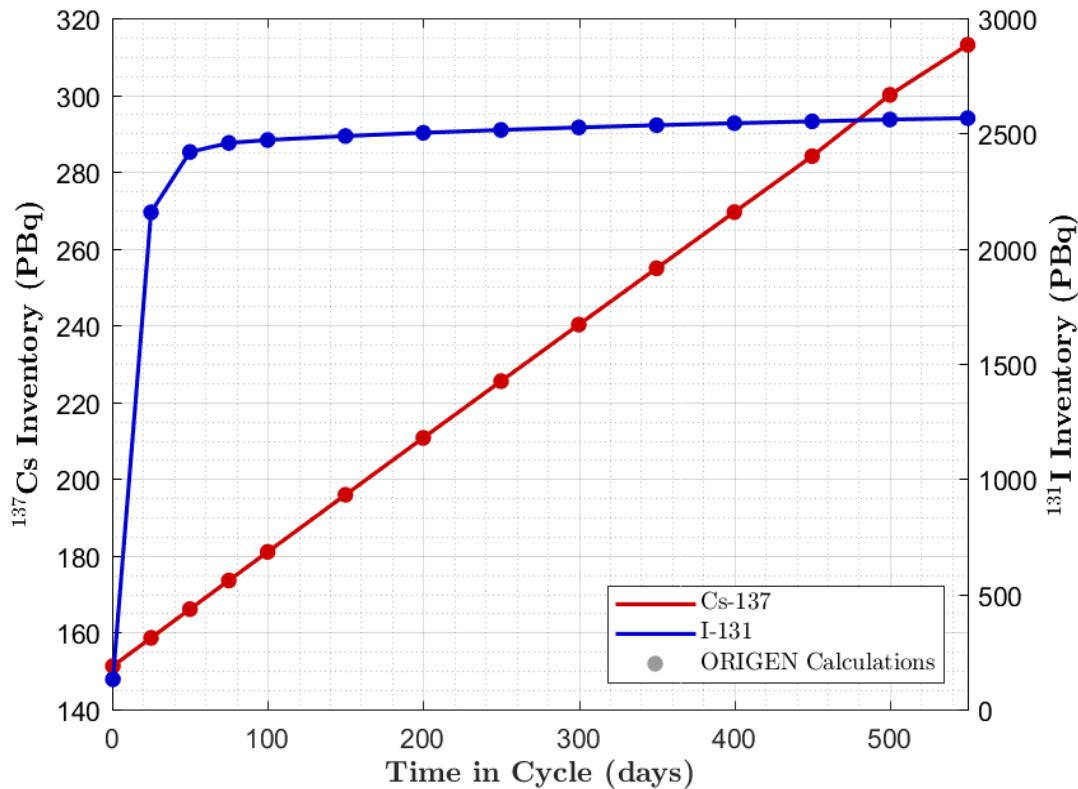


Figure 8 Activity levels for I-131 and Cs-137 with respect to time in cycle [17].

5. ESTIMATION OF THE BOUNDING SOURCE TERM

The previous sections outlined some of the issues, challenges, and important first-order effects impacting the estimation of a bounding source term. Just in the US, the research and regulations for the determination of a bounding source term for BWRs and PWRs has been ongoing since 1962 with the development of the TID-14884 siting requirements. There has been active research to further refine the bounding source term, which is partially outlined in Section 2. Section 2 also describes historical accidents as well as the responsive regulatory requirements to address specific and related generic causes of the accident. Section 3 identified SFPs as an important and usually the largest onsite source of radioactive inventory. SFPs have much different accident progressions due to lower decay heat but benefit from the decay of short-lived nuclides (e.g., I-131). However, there can be up to multiple cores worth of spent fuel assemblies stored in only a confinement structure.

Finally, Section 4 surveys some of the first order effects that could impact the magnitude of the source term. The reactors are almost always located at power stations with multiple reactors, which would have a corresponding number of SFPs. The thermal power rating is a reasonable first-order method for scaling the magnitude of the radioactive inventory, which is also indicated in Section 4. This judgment considers the uncertainty in the number of effected reactors at a multi-unit site and their time in cycle may be difficult to assess. The approximation is also relevant considering the number of assemblies and offload times of the fuel of SFP is unknown and difficult or impossible to know without cooperation from the site. The SFP inventory and source term could potentially be much larger in magnitude than the reactor accident. Finally, Section 4 indicates whether the reactor is located in a reinforced concrete containment or a confinement, the working fluid, and whether there is a graphite moderator (i.e., only the RBMK). Whether the containment remains intact has a first-order importance on the magnitude of the source term as discussed in Section 2. The sodium and graphite in the BN-800 and RBMK reactors, respectively, are important as co-located fire sources that can enhance the thermal release of radionuclides from the fuel. However, References [12] through [17] show how zircaloy oxidation in LWR accidents is an equally important driving heat source for radionuclide release.

Without a detailed analysis of the specific consequences from an initiating event, there is an insurmountable amount of uncertainty to reasonably characterize the accident consequences. As guidelines for this project, the damage from the initiating event will be specified as an unverified assumption. For example, it is reasonable to question whether a reinforced containment could be significantly damaged without damaging the reactor itself. Similarly, whether the SFP(s) progress to an accident and whether multiple reactors have some combination of these consequences must be addressed parametrically by summing combinations of source terms. Finally, as indicated in Section 2, our understanding of the complex phenomena that occurs in severe accidents already has significant uncertainties, whether in the code simulations or the forensic evaluations of historical accidents. Usually the biggest uncertainty would be the timing of the containment failure, which is one of the scenario possibilities to be considered. Any delay or hold-up of radionuclides in the containment has great benefits for natural aerosol retention due to gravitation settling. The destruction of the containment retention capability eliminates a very important radionuclide barrier that will have a significant impact on the size of the source term.

For the light-water reactor designs, MELCOR includes a CORSOR-Booth radionuclide release model that scales other element class releases to the cesium release. The CORSOR-Booth model for radionuclide release was validated to experimental results from the VERCOS experiments and benchmarked to Phebus data [48]. The scaling factors shown in Table 15 can be used to estimate the whole source term when there is limited information is available. The radionuclide class scaling

factors shown in Table 15 can be applied to the various elements as shown in Table 16. MELCOR groups similar elements together based on similar chemical characteristics. For the purposes of this project, a Cs-137 release can be extended to all the listed alkali metals in Table 16 and the CsI compound using the scaling factor in Table 15. The following subsections provide estimates of the radionuclide releases.

Table 15 CORSOR-Booth radionuclide release scaling factors

Class	Scaling Factor
Xe	1.0
Cs	1.0
Ba	4.0e-4
I ₂	0.64
Te	0.64
Ru	2.5e-3
Mo	6.25e-2
Ce	4.0e-8
La	4.0e-8
UO ₂	3.2e-4
Cd	0.25
Ag	0.16
CsI	0.64
Cs ₂ MoO ₄	1.0

Table 16 MELCOR radionuclide classes

Class	Class Name	Chemical Group	Representative	Member Elements
1	XE	Noble Gas	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N
2	CS	Alkali Metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu
3	BA	Alkaline Earths	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm
4	I ₂	Halogens	I ₂	F, Cl, Br, I, At
5	TE	Chalcogens	Te	O, S, Se, Te, Po
6	RU	Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni
7	MO	Early Transition Elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W
8	CE	Tetravalent	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C
9	LA	Trivalent	La	Al, Sc, Y, La, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf
10	UO ₂	Uranium	UO ₂	U
11	CD	More Volatile Main Group	Cd	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi
12	AG	Less Volatile Main Group	Ag	Ga, Ge, In, Sn, Ag
16	CSI	Cesium iodide	CsI	CsI
17	CSM	Cesium molybdate	Cs ₂ MoO ₄	Cs ₂ MoO ₄

5.1. RBMK-1000 source term

The following recommendations are made for an RBMK-1000 source term. Due to limited accident progression and source term knowledge for an RBMK-1000s and the unique characteristics of their confinement structure, the radionuclide release is assumed to be an adaptation of the Chernobyl Unit 4 accident source term. The key change would be the timing of the accident source term. Chernobyl Unit 4 progressed with a high-energy dispersion of the fuel following the 10X to 100X peak power increase during the reactivity event. Consequently, it took nearly a week for a thermal oxidation release to occur. Following the changes in the RBMK-1000s to reduce the likelihood of reactivity events (see Section 2.4.3.2), it is assumed that the accident will progress in a more coherent manner with an early oxidation transient, no dispersion of the fuel, and ignition of the graphite blocks.

The accidents of interest include,

- (a) loss-of-power with no confinement damage and no successful mitigation,
- (b) loss-of-power with confinement damage and no successful mitigation,
- (c) loss-of-power with confinement damage, a leak in the primary system, and no successful mitigation,
- (d) a loss-of-power with an SFP accident without confinement failure and no mitigation,
- (e) a loss-of-power with an SFP accident with confinement failure and no mitigation, and
- (f) a loss-of-power with confinement failure, an SFP leak, and no mitigation.

As noted in Section 4, the RBMK-1000 has a significantly lower threshold for failure versus a reinforced containment.

For the reactor accidents (b) and (c), the source term will be assumed to be equal to the upper estimates from the Chernobyl Unit 4 accident. However, the release is assumed to occur in one thermal oxidation phase due to the lack of a dispersing explosive event. The release is estimated to start at 2 hr with no reactor leak and 30 min with a reactor leak, which is estimated to be about half as long as the PWR⁷ short-term station blackout value from Table 7 (i.e., $\sim\frac{1}{2} \times 3.9$ hr for the start of the gap phase). The duration is estimated from Table 6 to be 4.7 hr. It could be argued that the BWR values are also applicable, but the timing differences are relatively small and the RBMK-1000 design is also significantly different than the BWR. For the leaky reactor sequence, the reactor is estimated to drain-down and heatup in 30 min. The RBMK-1000 has a complex piping layout and some of the smaller pipes are assumed to fail for a quick but not instantaneous blowdown like a PWR large recirculation piping break. The author is not aware of any passive safety system accumulators in the RBMK-1000 design that would slow the accident progression. Table 17 summarizes the RBMK-1000 bounding source term specifications.

A 66% retention is applied to the releases if the confinement remains intact based on the intact SFP reactor building results from the NRC SFP study (see Section 3). However, there is a large uncertainty whether any significant amount of low volatile radionuclides would be released with an intact confinement and whether a robust, long-term graphite fire could be sustained in an intact building. For example, see the NRC SFP study, which had a very limited oxidation reaction with an intact reactor building.

⁷ The RBMK-1000 is a boiling water reactor and does not have steam generators. However, it does have two very large steam drums that contain a large amount of water. The inventory in the steam drums and the recirculation loop must be boiled away before the accident commences.

The SFP accident also has the uncertainty of two SFPs per RBMK-1000. A two SFP configuration is also consistent with the new VVER-1200 designs. Russia also has reprocessing facilities.

Consequently, the fuel in the SFP is likely removed for reprocessing after cooling. It is assumed that 3 cores are stored in the two SFP pools. This corresponds to a ~9-year SFP loading, assuming 0.5 cores discharged per 1.5 years. Each SFP is assumed to have 1.5 cores. The start of the accident is assumed to occur per the leakage variations cited in Table 13. It is assumed that most of the I-131 has decayed away. The 42% cesium release fractions are based on the bounding value from NRC SFP study Table 12 [39], which includes a failed confinement. The other radionuclides are estimated from Table 15. If the confinement building remains intact, a 2% release is estimated based on the corresponding NRC SFP study (see Section 3).

The radionuclide inventory of an RBMK-1000 SFP is complicated due to multiple offloads with multiple decay times. It may only be practical to model the Cs-137 releases in SFP accidents, which would show minimal decay across multiple offloads. Otherwise assume the following decay times and inventory fractions, (a) 16.7% at 32 days decay, (b) 16.7% at 1.5 years decay, (c) 16.7% at 3 years decay, (d) 16.7% at 4.5 years decay, 16.7% at 6 years decay, (e) 16.7% at 7.5 years decay, and (f) 16.7% at 9.0 years decay. The RBMK-1000 is further complicated with online refueling, so the previous discrete values only approximate the actual inventory discharge history. Some additional key uncertainties include how the continuous online refueling impacts the RBMK-1000 SFP decay heat power, the vulnerability of this configuration to heatup, and the sustained presence of a small iodine source term.

5.2. PWR LWR source terms with a containment

The following recommendations are made for the LWR PWR designs with a containment, which include the VVER-1000 and the PWR. These designs are lumped together for simplification and their similarities. Both designs will have excellent retention if the containment remains intact. [13]

The accidents of interest include,

- (a) loss-of-power with no initial containment damage and no successful mitigation,
- (b) loss-of-power with containment damage and no successful mitigation,
- (c) loss-of-power with containment damage, a leak in the primary system, and no successful mitigation,
- (d) a loss-of-power with an SFP accident without confinement failure and no mitigation,
- (e) a loss-of-power with an SFP accident with confinement failure and no mitigation, and
- (f) a loss-of-power with confinement failure, an SFP leak, and no mitigation.

It is expected that a reinforced PWR containment has a significantly higher threshold for failure versus the RBMK-1000, VVER-440, or the SFP confinement structures. However, if an initiating event is severe enough to fail the containment, it is likely to impact multiple units, the SFPs and perhaps the primary system.

The results from an initially intact containment are estimated using insights from the Surry PWR SOARCA study [14]. Although this study was replaced by the subsequent SOARCA Uncertainty Assessment [17], it offers a few advantages for this application. First, the original SOARCA study calculations went to 7 days versus 72 hr for the UA study. Second, the magnitude of the source term is larger. In the process of updating the original Surry SOARCA study for the uncertainty analysis, it was determined that the original SOARCA study used limestone concrete in the reactor cavity under the reactor vessel versus basaltic concrete (e.g., see full discussion in Appendix A of Reference [49]). This led to a faster pressurization of the containment during the ensuing core-

concrete interactions, an earlier containment failure, and a larger source term. Finally, for the purposes of determining a bounding source term, the original SOARCA results provide conservatism and allowance for a possibly less robust construction than a US plant or weakening due to the initiating event.

Figure 9 shows a comparison of the iodine release in the SOARCA study as compared to the SOARCA UA study. The steel liner within the reinforced concrete walls provided leak-tightness. However, the pressurization of the containment due to core-concrete interactions after the vessel failed eventually caused the failure of the concrete steel liner within the reinforced concrete walls. After liner failure, the release of the radionuclides from the containment to the environment increased significantly. Liner failure also occurred in the SOARCA UA study but not until approximately 50 hr versus 25 hr in the SOARCA study. With an additional 25 hr of time for natural gravitational settling in the SOARCA UA study, the available airborne aerosols for release at the time of the containment failure was significantly smaller. In addition, the leakage rate was also smaller due to the smaller gas generation rate with the basaltic concrete in the SOARCA UA study.

Figure 10 shows the 7-day environmental source term from the SOARCA study, which will be the basis for the environmental release fractions for the bounding, initially intact PWR containment. Table 19 shows an approximation of the releases versus times for the twelve radionuclide classes. The initial release starts at approximately 3.8 hr (i.e., defined as the noble gas fraction $> 10^{-6}$ of the initial inventory) and extends through 7 days.

For the reactor accidents (b) and (c), the source term will be estimated using the revised high-burnup fuel NUREG-1465 study presented in Section 2.3.1. The radionuclide releases are specified using Table 6 results with an estimated retention of 0.66 based on the NRC SFP study with a failed reactor building. The start of the intact reactor scenario is estimated using the PWR short-term station blackout value from Table 7 (i.e., ~ 4 hr for the start of the gap phase). The duration is estimated from Table 6 to be 9.5 hr by lumping the gap, in-vessel, and ex-vessel together. The 143 hr long-term release of revolatilized radionuclides from the vessel is also estimated from Table 6.

For the leaky reactor sequence, the reactor is estimated to drain-down and heatup in 1 hr. PWRs have accumulators that passively refill the core after a pipe break, so the release would be slightly delayed while the water boils away. Table 17 summarizes the PWR bounding source term specifications.

It is assumed that 3 cores are stored in a PWR SFP pools. This appears applicable for PWRs and corresponds to a ~ 9 -year SFP loading, assuming 0.5 cores discharged per 1.5 years. Consequently, the SFP is assumed to have 3 cores. The start of the accident is assumed to occur at 1-week per Table 13. It is assumed that most of the I-131 has decayed away. 2% and 42% cesium release fractions are based on the bounding values from NRC SFP study for intact and failed building, respectively (see Table 12 [39]). The other radionuclides are estimated from Table 15. It may only be practical to model Cs-137 releases in SFP accidents, which would show minimal decay across multiple offloads. Otherwise assume the following decay times and inventory fractions, (a) 16.7% at 32 days decay, (b) 16.7% at 1.5 years decay, (c) 16.7% at 3 years decay, (d) 16.7% at 4.5 years decay, 16.7% at 6 years decay, (e) 16.7% at 7.5 years decay, and (f) 16.7% at 9.0 years decay. The SFP source term description is also described in Table 19, which refers to Table 18 for specific values.

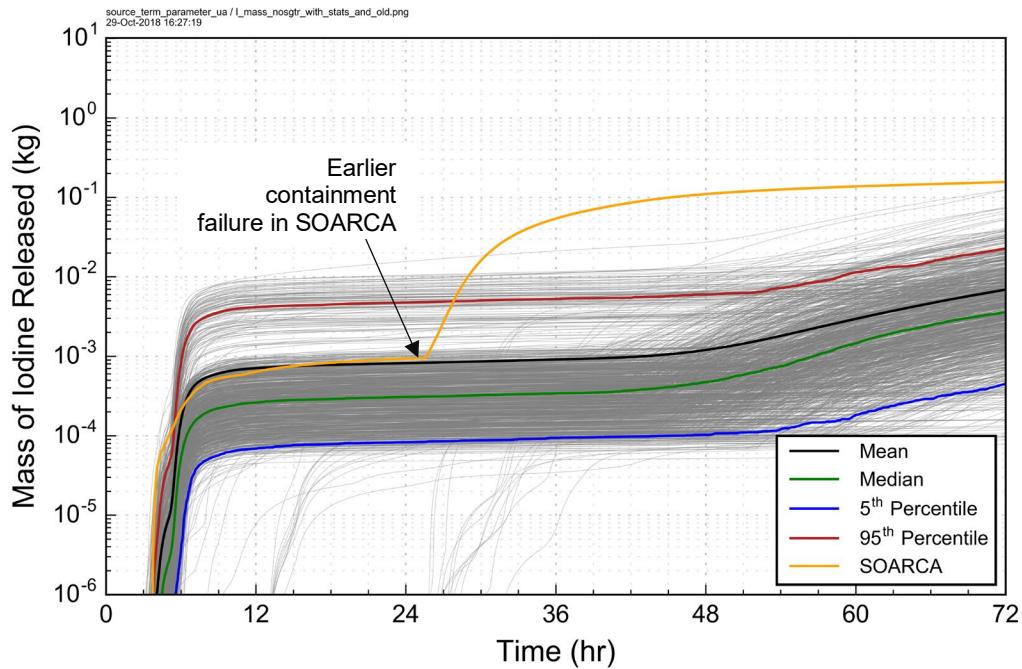


Figure 9 Comparison of the iodine releases to the environment from the SOARCA UA study as compared to the original SOARCA result [17].

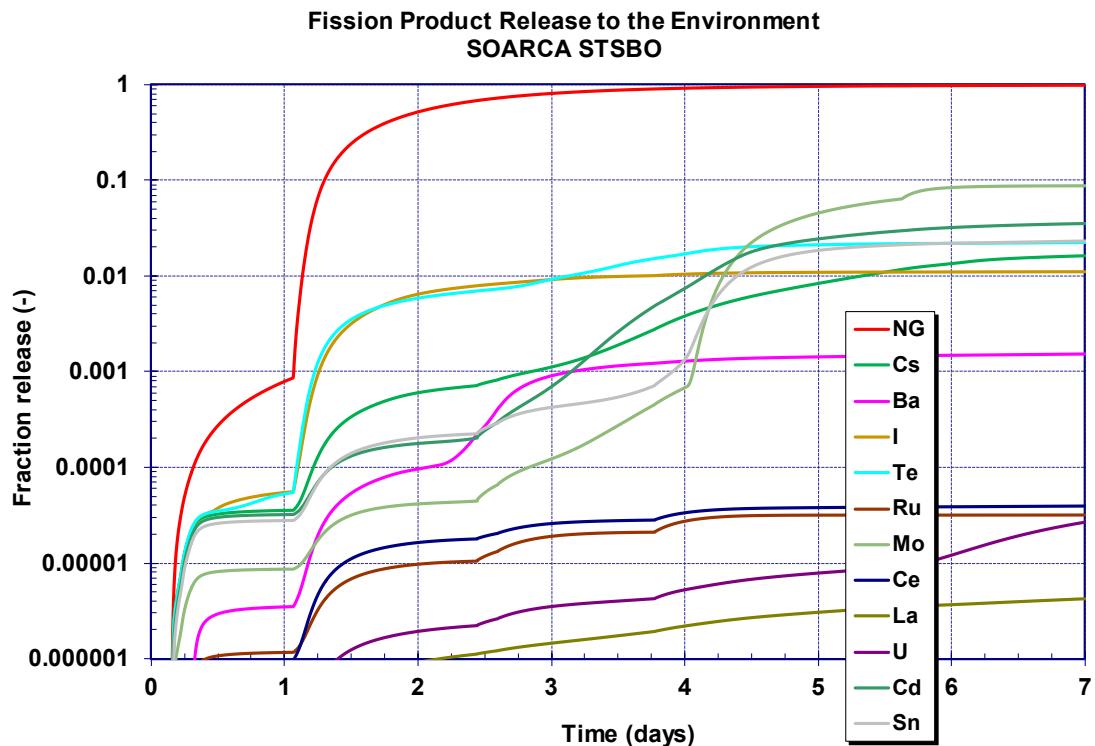


Figure 10 Environmental source term from the SOARCA study [14].

5.3. BWR LWR source terms with a containment

The following recommendations are made for the BWR designs. The accidents of interest include,

- (a) loss-of-power with no initial containment damage and no successful mitigation,
- (b) loss-of-power with containment damage and no successful mitigation,
- (c) loss-of-power with containment damage, a leak in the primary system, and no successful mitigation,
- (d) a loss-of-power with an SFP accident without confinement failure and no mitigation,
- (e) a loss-of-power with an SFP accident with confinement failure and no mitigation, and
- (f) a loss-of-power with confinement failure, an SFP leak, and no mitigation.

The BWR steel containment has a significantly higher threshold for failure versus a confinement building (e.g., RBMK-1000 or VVER-440). However, the proximity of the reactor vessel to the containment wall is much closer in a BWR/4 Mark I containment than the other reactor designs. Consequently, it is likely that an initiating event that damages the containment would also damage the reactor vessel.

For the reactor accidents (b) and (c), the source term will be estimated using the revised high-burn-up fuel NUREG-1465 study presented in Section 2.3.1. The release is assumed to be specified by Table 5 results with an estimated retention of 0.66 and 0.20 for an intact and failed building, respectively, based on the NRC SFP study. The start of the intact reactor releases are estimated based on a BWR short-term station blackout value from Reference [13] (i.e., ~1 hr for the start of the gap phase). The duration is estimated from Table 5 to be 11 hr, lumping gap, in-vessel, and ex-vessel together. The 12 hr long-term release of revolatilized radionuclides from the vessel is also estimated from Table 5. For the leaky reactor sequence, the reactor is estimated to drain-down and heatup in 30 min. Table 20 summarizes the BWR bounding source term specifications.

The results from an initially intact containment are characterized by the SOARCA Peach Bottom short-term station blackout results [13], see Figure 8. This is recommended because the Fukushima Dai-ichi nuclear results discussed in Section 2.4.1 included some emergency core cooling system operation and other mitigative actions. Reference [13] shows relatively high releases (e.g., Iodine, Barium, and Tellurium releases at 10% of the initial inventory) due the drywell containment liner failure at 8.2 hr.

It is assumed that 3 cores are stored in a BWR SFP pools. This corresponds to a ~9-year SFP loading, assuming 0.5 cores discharged per 1.5 years. Consequently, the SFP is assumed to have 3 cores. The start of the accident is assumed to occur at 1-week per Table 13. It is assumed that most of the I-131 has decayed away. 2% and 42% cesium release fractions are based on the bounding value from NRC SFP study for an intact and failed building, respectively (i.e., see Table 12 [39]). The other radionuclides are estimated from Table 15. It may only be practical to model Cs-137 releases in SFP accidents, which would show minimal decay across multiple offloads. Otherwise assume the following decay times and inventory fractions, (a) 16.7% at 32 days decay, (b) 16.7% at 1.5 years decay, (c) 16.7% at 3 years decay, (d) 16.7% at 4.5 years decay, 16.7% at 6 years decay, (e) 16.7% at 7.5 years decay, and (f) 16.7% at 9.0 years decay. The SFP source term description is also described in Table 20, which refers to Table 18 for specific values.

5.4. VVER-440 LWR source terms with a confinement

The following recommendations are made for the VVER-440 design. The accidents of interest include,

- (a) loss-of-power with no initial containment damage and no successful mitigation,
- (b) loss-of-power with containment damage and no successful mitigation,
- (c) loss-of-power with containment damage, a leak in the primary system, and no successful mitigation,
- (d) a loss-of-power with an SFP accident without confinement failure and no mitigation,
- (e) a loss-of-power with an SFP accident with confinement failure and no mitigation, and
- (f) a loss-of-power with confinement failure, an SFP leak, and no mitigation.

The VVER-440 confinement has a significantly lower threshold for failure versus a reinforced concrete PWR containment.

For the reactor accidents (b) and (c), the source term will be estimated from the high-burn-up fuel NUREG-1465 study presented in Section 2.3.1. The release is specified by Table 6 results with an estimated onsite retention of 0.66 and 0.80 for an intact and failed building, respectively, based on the NRC SFP study results. The start of the intact reactor scenario is estimated based on a PWR short-term station blackout value from Table 7 (i.e., ~4 hr for the start of the gap phase). The duration is estimated from Table 6 to be 9.5 hr by lumping the gap, in-vessel, and ex-vessel together. The 143 hr long-term release of revolatilized radionuclides from the vessel is also estimated from Table 6.

For the leaky reactor sequence, the reactor is estimated to drain-down and heatup in 1 hr. PWRs have accumulators that passively refill the core after a pipe break, so the release would be slightly delayed while the water boils away. Table 21 summarizes the VVER-440 bounding source term specifications.

It is assumed that 3 cores are stored in a VVER-440 SFP pool. This corresponds to a ~9-year SFP loading, assuming 0.5 cores discharged per 1.5 years. Consequently, the SFP is assumed to have 3 cores. The start of the accident is assumed to occur at 1-week per Table 13. It is assumed that most of the I-131 has decayed away. A 42% cesium release fractions are based on the bounding value from NRC SFP study Table 12 [39]. The other radionuclides are estimated from Table 15. It may only be practical to model Cs-137 releases in SFP accidents, which would show minimal decay across multiple offloads. Otherwise assume the following decay times and inventory fractions, (a) 16.7% at 32 days decay, (b) 16.7% at 1.5 years decay, (c) 16.7% at 3 years decay, (d) 16.7% at 4.5 years decay, 16.7% at 6 years decay, (e) 16.7% at 7.5 years decay, and (f) 16.7% at 9.0 years decay. The SFP source term description is also described in Table 21, which refers to Table 18 for specific values.

5.5. BN-800 source terms with a confinement

No severe accident source term has yet been identified for the BN-800 reactor. Either the VVER-440 source term or the RBMK-1000 are suggested as surrogates with appropriate inventory scaling to the smaller BN-800 thermal rating of 2100 MW. The BN-800 has a confinement building similar to a RBMK-1000. For the SFP source term, assume same parameters as the RBMK-1000 with appropriate inventory scaling for the smaller thermal power.

5.6. CANDU source terms with a containment

No severe accident source term has yet been researched for a CANDU design reactor. The PWR LWR source term is suggested as a surrogate with appropriate inventory scaling to the smaller thermal rating of 1700 MW. The CANDU has a containment building similar to a PWR reinforced concrete containment. For the SFP source term, assume same parameters as the PWR SFP with appropriate inventory scaling for the smaller thermal power.

5.7. Summary of key source term assumptions

The following assumptions were used in the development of the source terms.

- The scope of the consequential failures at the site is not estimated. The source terms were evaluated based on specified endstates (e.g., the containment is not functioning). The individual source terms can be summed to estimate the total site source term based on number of reactors impacted and their containment/confinement operational status (i.e., failed or intact) and the number of SFPs impacted and their enclosure building operation status (i.e., failed or intact). The following judgments are offered.
 - Nearly every site has multiple reactors.
 - A confinement is expected to be significantly less robust than a reinforced concrete containment.
 - The SFP enclosure building is also expected to be significantly less robust than a reinforced concrete containment.
 - The spacing between the steel BWR containment and the vessel and recirculation system is very small. If the containment is damaged, then it is likely that the vessel or recirculation system will have damage. The BWR SFP is located above the containment. The reactor building and the SFP would be damaged if the containment is damaged.
- It was assumed there was no mitigation of the accidents. The timing from the initiating event and the start of the radionuclide release is estimated using the results from the SOARCA study.
- The start of the PWR and BWR radionuclide releases were based on the SOARCA study for a short-term station blackout with no onsite power [13][14]. If the primary system was damaged and leaking, the estimated timing to the start of the radionuclide releases was accelerated. The start of the PWR releases was slightly later due to the water injection from passive accumulators.
- The radioactive activity inventory can be estimated by scaling the thermal power of the reactor design to the available high burn-up inventories in the MACCS database. This is judged acceptable due to the uncertainties in the magnitude of the source term (see summary in Section 5).
- The release fractions from the updated NUREG-1465 high burn-up study [8] are characteristic of the releases from the BWRs and PWRs to the containment. The NRC bounding source terms assume a functioning containment with design leakage. The assumptions cited below describe application to a failed containment.

- The Russian reactors were licensed according to their country's regulations, which had significantly less rigor prior to the Chernobyl accident. They have often been criticized for less effective safety systems, the lack of a reinforced containment in most early designs, poor construction and quality assurance, and ineffective regulation.
- The radionuclide release from the SFP is characterized using results from the NRC SFP study. The magnitude of the cesium releases were characterized based on whether the enclosure building remained intact or failed. The release of the other radionuclides are scaled relative to the cesium release based on the CORSOR-Booth release model in MELCOR.
- When a containment is initially functioning properly, then the releases are very delayed and small (e.g., see Section 5.2). The results from the SOARCA study were used to estimate the releases.
 - The PWR releases with an initially intact were not fully characterized due to their low magnitude relative to the other endstates. The calculated release fractions for all chemical forms of cesium and iodine to be 0.0001 and 0.001 at 72 hr, respectively [14].
 - The BWR releases with an initially intact containment were estimated from the SOARCA study. The containment failed early than the PWR containment as the accident progressed and the full magnitude of the source term was better characterized [13], albeit small relative to the other endstates.
- When a confinement is functioning properly, then only 33% of the radionuclides were released based on the estimated retention of 66% to 75% from the NRC SFP study [39].
- When a containment or confinement is not functioning properly, then 90% of the radionuclides were released based on the estimated retention of 10% to 20% from the NRC SFP study [39].
- The radionuclide inventory in the SFP is assumed to consist of 3 effective core loads. The possibility of a shared SFP for multiple reactors was not considered (i.e., possible but included here). Chernobyl had 2 SFPs per reactor. It was assumed the SFP was evenly split between each SFP. Consequently, their accidents and source terms could be constructed differently.

Table 17 RBMK-1000 bounding reactor accident source term

Start of release after the initiating event in the reactor accident

No reactor damage and leak: 2 hr
Reactor damage and leak: 0.5 hr

Accident duration

Thermal oxidation phase duration: 4.7 hr

Accident source term per reactor with a failed confinement

Radionuclide	Xe	Iodine	Cesium	Ba/Sr	Ru/Sr	Zr, Np, Pu, Ce, Cm
Release	100%	60%	43%	6%	6%	3.5%
Release duration [hr]	4.5	4.5	4.5	4.5	4.5	168

Accident source term per reactor with an intact confinement

Assume building holdup of 66% based on SFP intact building DF from the NRC SFP study (see Section 3) and no changes to durations. Assume all Xe is leaked.

Radionuclide	Xe	Iodine	Cesium	Ba/Sr	Ru/Sr	Zr, Np, Pu, Ce, Cm
Release	100%	20%	14%	2%	2%	1%
Release duration [hr]	4.5	4.5	4.5	4.5	4.5	168

Reactor radionuclide Inventory

The RBMK-1000 thermal power rating is 3200 MW.

Use available MACCS PWR or BWR high-burn-up inventory and scale by rated thermal power (e.g., the Surry PWR thermal rating is 2546 MW and the Peach Bottom BWR rated thermal power is 3514 MW)

Table 18 RBMK-1000 bounding SFP accident source term**Start of release after the initiating event in the reactor accident**

See Table 12 Summary of the NRC SFP study Cs-137 radionuclide releases [39]. Table 12, >7 days to 2.5 hr.

Accident duration

Thermal oxidation phase duration: 12 hr (estimated from Figure 6).

SFP accident source term per reactor with failed confinement

Class	Cs-137 Release	Scaling Factor	Release fraction
Xe		1	3.8E-01
Cs	0.42	1	3.8E-01
Ba		4.00E-04	1.5E-04
I		0.64	2.4E-01
Te		0.64	2.4E-01
Ru		2.50E-03	9.5E-04
Mo		6.25E-02	2.4E-02
Ce		4.00E-08	1.5E-08
La		4.00E-08	1.5E-08
UO ₂		3.20E-04	1.2E-04
Cd		0.25	9.5E-02
Ag		0.16	6.0E-02

Assumes 10% retention per the NRC SFP study with a failed reactor building [39].

Start of release after the initiating event in the SFP accident

No SFP damage and leak: 168 hr

SFP leak: 19 hr

SFP accident source term per reactor with intact building

Class	Cs-137 Release	Scaling Factor	Release fraction
Xe		1	6.6E-03
Cs	0.02	1	6.6E-03
Ba		4.00E-04	2.6E-06
I		0.64	4.2E-03
Te		0.64	4.2E-03
Ru		2.50E-03	1.7E-05
Mo		6.25E-02	4.1E-04
Ce		4.00E-08	2.6E-10
La		4.00E-08	2.6E-10
UO ₂		3.20E-04	2.1E-06
Cd		0.25	1.7E-03
Ag		0.16	1.1E-03

Assumes 66% retention per the NRC SFP study with an intact reactor building [39].

SFP radionuclide Inventory

RBMK-1000 thermal power rating is 3200 MW. Assume each SFP contains 1.5 cores.

Use available MACCS PWR or BWR high-burn-up inventory and scale by rated thermal power (e.g., the Surry PWR thermal rating is 2546 MW and the Peach Bottom BWR rated power is 3514 MW). If possible, use an inventory at >32 days (~4 I-131 half-lives) and/or neglect the iodine source term.

Table 19 PWR bounding reactor and SFP accident source term**Accident source term per reactor with a failed containment**

Class	Gap -> Ex-vessel	With DF	Late in-vessel	With DF
Xe	0.968	9.7E-01	0.0003	3.0E-04
Cs	0.253	2.3E-01	0.06	5.4E-02
Ba	0.0076	6.8E-03		
I ₂	0.385	3.5E-01	0.21	1.9E-01
Te	0.307	2.8E-01	0.1	9.0E-02
Ru	0.006	5.4E-03		
Mo	0.09	8.1E-02	0.03	2.7E-02
Ce	2.40E-04	2.2E-04		
La	1.32E-05	1.2E-05		

Assumed failed containment retention of 10% based on the NRC SFP study with a failed reactor building [39].

Start of release after the initiating event in the reactor accident

No reactor damage and leak: 4 hr

Reactor damage and leak: 1 hr

Duration of early release: 9.5 hr

Start of late in-vessel phase: At 4.7 hr after the start of the releases

Accident source term per reactor with an initially intact containment

Time	Cumulative Release					
	3.8 h	9.6 h	24.4 h	36 h	48 h	7 days
Xe	1.E-06	1.78E-04	8.66E-04	2.42E-01	5.19E-01	9.87E-01
Cs	<1.E-06	3.03E-05	3.56E-05	3.38E-04	6.03E-04	1.63E-02
Ba	<1.E-06	2.39E-06	3.50E-06	5.37E-05	9.61E-05	1.53E-03
I	<1.E-06	3.22E-05	5.57E-05	3.17E-03	6.46E-03	1.11E-02
Te	<1.E-06	3.25E-05	5.48E-05	3.52E-03	5.87E-03	2.24E-02
Ru	<1.E-06	9.83E-07	1.17E-06	6.83E-06	9.71E-06	3.18E-05
Mo	<1.E-06	7.64E-06	8.64E-06	3.17E-05	4.15E-05	8.77E-02
Ce	<1.E-06	6.47E-07	9.67E-07	1.11E-05	1.64E-05	3.94E-05
La	<1.E-06	1.68E-08	2.58E-08	5.25E-07	9.14E-07	4.23E-06
U	<1.E-06	1.28E-07	1.56E-07	1.24E-06	1.93E-06	2.68E-05
Cd	<1.E-06	2.79E-05	3.21E-05	1.31E-04	1.78E-04	3.54E-02
Sn	<1.E-06	2.41E-05	2.79E-05	1.40E-04	2.04E-04	2.32E-02

The results from an initially intact containment are characterized by the SOARCA Surry short-term station blackout results, see Figure 10 [14]. All values in the table are cumulative values.

Reactor radionuclide Inventory

The VVER-1000 and PWR thermal power ratings are 3000 MW.

Use available MACCS PWR high-burn-up inventory and scale by rated thermal power (e.g., the Surry PWR thermal rating is 2546 MW)

Accident source term per SFP with an intact or failed confinement

The intact and failed confinement results in Table 18 should be used for the PWR SFP accident source term. A 3 core-equivalent inventory should be assumed. See discussion in Section 5.1 for inventory aging recommendations.

Table 20 BWR bounding reactor and SFP accident source term

Accident source term per reactor with a failed containment

Class	Gap -> Ex-vessel	With DF	Late in-vessel	With DF
Xe	0.968	9.7E-01	0.003	3.0E-03
Cs	0.253	2.3E-01	0.06	5.4E-02
Ba	0.0076	6.8E-03		
I	0.385	3.5E-01	0.21	1.9E-01
Te	0.307	2.8E-01	0.1	9.0E-02
Ru	0.006	5.4E-03		
Mo	0.09	8.1E-02	0.03	2.7E-02
Ce	0.00024015	2.2E-04		
La	0.00001315	1.2E-05		

Assumed failed containment retention of 10% based on the NRC SFP study with a failed reactor building [39].

Start of release after the initiating event in the reactor accident

No reactor damage and leak: 1 hr

Reactor damage and leak: 0.5 hr

Duration of early release: 11 hr

Start of late in-vessel phase: At 8 hr after the start of the releases

Accident source term per reactor with an initially intact confinement

Class	Cumulative Release				
	8.2 hr	8.3 hr	25.5 hr	30 hr	48 hr
Xe	Containment Failure - Releases are <10 ⁻⁶ prior to 8.2 hr	0.8			0.95
Cs		0.003	0.008	0.015	0.018
Ba		0.08			0.09
I ₂		0.006	0.015	0.10	0.105
Te		0.007	0.011	0.04	0.1
Ru		0.000003			0.0000035
Mo		0.0007			0.0025
Ce		0.007			0.007
La		0.0001			0.00012
Cd		0.028			0.014
Sn		0.0013			0.003

The results from an initially intact containment are characterized by the SOARCA Peach Bottom short-term station blackout results [13], see Figure 11. All values in the table are cumulative values.

Reactor radionuclide Inventory

The BWR thermal power ratings is 2923 MW.

Use available MACCS PWR high-burn-up inventory and scale by rated thermal power (e.g., the Peach Bottom BWR thermal rating is 3514 MW)

Accident source term per SFP with an intact or failed confinement

The intact and failed confinement results in Table 18 should be used for the BWR SFP accident source term. A 3 core-equivalent inventory should be assumed. See discussion in Section 5.1 for inventory aging recommendations.

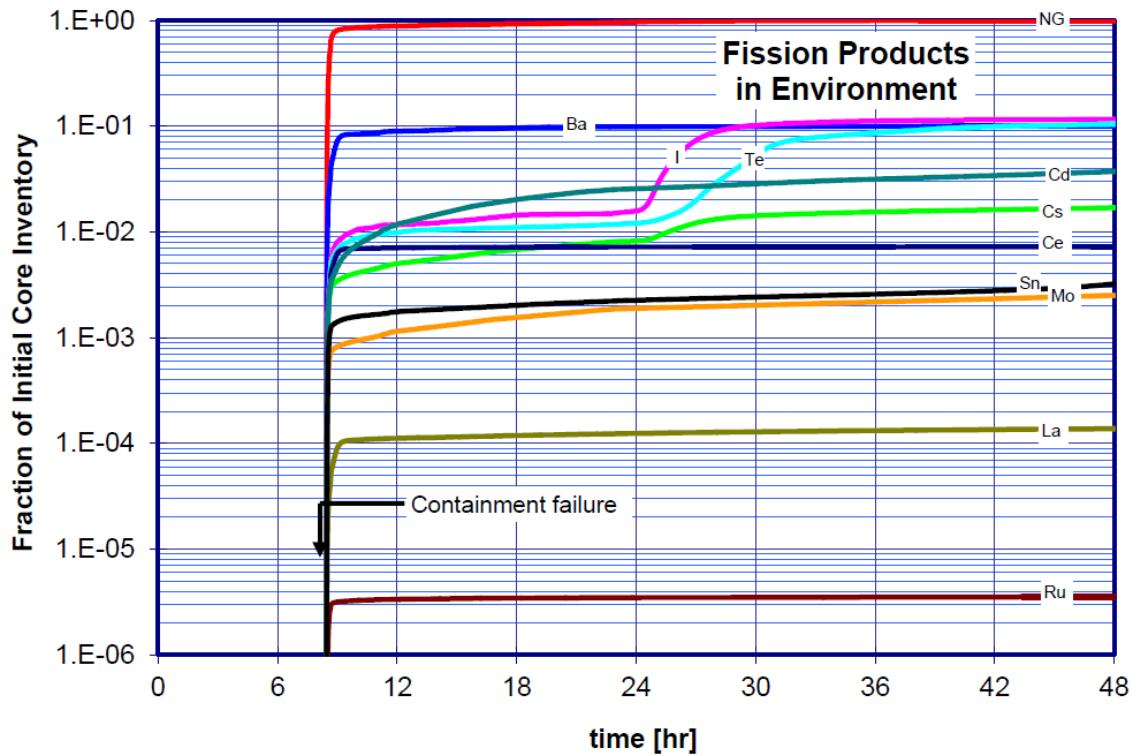


Figure 11 Fission product release in Peach Bottom short-term station blackout without mitigation [13].

Table 21 VVER-440 bounding reactor and SFP accident source term**Accident source term per reactor with a failed confinement**

Class	Gap -> Ex-vessel	With DF	Late in-vessel	With DF
Xe	0.968	9.7E-01	0.0003	3.0E-04
Cs	0.253	2.3E-01	0.06	5.4E-02
Ba	0.0076	6.8E-03		
I	0.385	3.5E-01	0.21	1.9E-01
Te	0.307	2.8E-01	0.1	9.0E-02
Ru	0.006	5.4E-03		
Mo	0.09	8.1E-02	0.03	2.7E-02
Ce	2.40E-04	2.2E-04		
La	1.32E-05	1.2E-05		

Assumed failed containment retention of 10% based on the NRC SFP study with a failed reactor building [39].

Start of release after the initiating event in the reactor accident

No reactor damage and leak: 4 hr

Reactor damage and leak: 1 hr

Duration of early release: 9.5 hr

Start of late in-vessel phase: At 4.7 hr after the start of the releases

Accident source term per reactor with an initially intact confinement

Class	Gap -> Ex-vessel	With DF	Late in-vessel	With DF
Xe	0.968	9.7E-01	0.0003	3.0E-04
Cs	0.253	8.3E-02	0.06	2.0E-02
Ba	0.0076	2.5E-03		
I	0.385	1.3E-01	0.21	6.9E-02
Te	0.307	1.0E-01	0.1	3.3E-02
Ru	0.006	2.0E-03		
Mo	0.09	3.0E-02	0.03	9.9E-03
Ce	2.40E-04	7.9E-05		
La	1.32E-05	4.3E-06		

Assumes 66% retention per the NRC SFP study with an intact reactor building [39].

Reactor radionuclide Inventory

The VVER-440 PWR thermal power rating is 1375 MW.

Use available MACCS PWR high-burn-up inventory and scaled by rated thermal power (e.g., the Surry PWR thermal rating is 2546 MW)

Accident source term per SFP with an intact or failed confinement

The intact and failed confinement results in Table 18 should be used for the VVER-440 SFP accident source term. A 3 core-equivalent inventory should be assumed. See discussion in Section 5.1 for inventory aging recommendations.

6. ACTIVITY INVENTORY AND RELEASE CONSIDERATIONS

In conjunction with specifying the release fractions and timing for these postulated accident scenarios, several additional factors are influential to fully characterize the radioactivity released to the environment, and ultimately the consequences of such an event should it occur. These additional factors include:

- Activity inventory
- Particle size distributions
- Energy of release

Each of these factors are discussed in terms of their influence on atmospheric dispersion modeling, values used in previous analyses, and postulated bounding approximations for this study. In determining bounding approximations, it was estimated that more dispersion is considered more severe for this application, which would result in lower values of activity spread over a greater geographic area, rather than higher values of activity concentrated in a much smaller area near the site.

6.1. Activity Inventory

For the PWR and BWR source terms, the activity inventory used for these calculations is based on the high burnup Surry PWR radionuclide inventory developed for the SOARCA study [17]. Values for activity (converted from mass at the time of reactor shutdown) are displayed in Table 22 below. From this information, scaling factors and release fractions from Section 5 of this report are applied to determine total activity released for each accident scenario. An additional analysis was later completed for the RBMK-1000 and VVER-440 which applies more specific inventories modeled by Los Alamos National Laboratory using the code Monteburns, which links MCNP and ORIGEN/CINDER [52].

Table 22 Radionuclide Inventory for Activity Calculations

Surry Initial Inventory from NUREG 7110		
(Scale 1 = 2546 MW)		
Group	Isotope	Activity (Bq)
Xe	Kr-85	2.94E+16
	Kr-85m	8.07E+17
	Kr-87	1.60E+18
	Kr-88	2.14E+18
	Xe-133	6.07E+18
	Xe-135	1.80E+18
	Xe-135m	1.29E+18
Cs	Cs-134	3.32E+17
	Cs-136	1.57E+17
	Cs-137	3.05E+17
	Rb-86	5.36E+15

Table 22 Radionuclide Inventory for Activity Calculations

Surry Initial Inventory from NUREG 7110		
(Scale 1 = 2546 MW)		
Group	Isotope	Activity (Bq)
	Rb-88	2.16E+18
Ba	Ba-139	5.54E+18
	Ba-140	5.37E+18
	Sr-89	2.98E+18
	Sr-90	2.27E+17
	Sr-91	3.75E+18
	Sr-92	4.00E+18
	Ba-137m	2.92E+17
I	I-131	2.78E+18
	I-132	4.08E+18
	I-133	5.76E+18
	I-134	6.48E+18
	I-135	5.49E+18
Te	Te-127	2.60E+17
	Te-127m	4.22E+16
	Te-129	7.79E+17
	Te-129m	1.49E+17
	Te-131m	5.71E+17
	Te-132	4.29E+18
	Te-131	2.55E+18
Ru	Rh-105	2.90E+18
	Ru-103	4.61E+18
	Ru-105	3.14E+18
	Ru-106	1.40E+18
	Rh-103m	4.61E+18
	Rh-106	1.56E+18
Mo	Nb-95	5.18E+18
	Co-58	4.79E+13
	Co-60	2.65E+14
	Mo-99	5.68E+18
	Tc-99m	5.03E+18
	Nb-97	5.24E+18
	Nb-97m	4.95E+18
Ce	Ce-141	4.87E+18
	Ce-143	4.55E+18
	Ce-144	3.42E+18

Table 22 Radionuclide Inventory for Activity Calculations

Surry Initial Inventory from NUREG 7110		
(Scale 1 = 2546 MW)		
Group	Isotope	Activity (Bq)
	Np-239	5.67E+19
	Pu-238	8.31E+15
	Pu-239	9.56E+14
	Pu-240	1.17E+15
	Pu-241	3.39E+17
	Zr-95	4.96E+18
	Zr-97	5.00E+18
La	Am-241	3.43E+14
	Cm-242	1.14E+17
	Cm-244	1.13E+16
	La-140	5.67E+18
	La-141	5.10E+18
	La-142	4.92E+18
	Nd-147	2.04E+18
	Pr-143	4.65E+18
	Y-90	2.39E+17
	Y-91	3.93E+18
	Y-92	4.11E+18
	Y-93	4.62E+18
	Y-91m	2.20E+18
	Pr-144	3.63E+18
	Pr-144m	5.06E+16

The spreadsheet accompanying this report provides a detailed breakdown of activity released for each bounding scenario, with the highest being 63.5 EBq total released activity for the RBMK-1000 failed confinement scenario. This value applies the CORSOR Booth model described in Section 5 of this report and applies these estimated release fractions to the RBMK-1000 specific inventory developed by Los Alamos National Laboratory.

6.2. Particle Size Distributions

Particle size distributions are influential in the atmospheric transport of radionuclide releases in that they determine the deposition velocity, or how quickly particulates settle onto a surface and is no longer being transported in the atmosphere. In general, smaller diameter particles have lower deposition velocities, and therefore remain airborne for longer periods of time. However, there is a point (at approximately 0.5 μm) where diffusion becomes a dominant factor and the deposition velocities begin the increase with decreasing size. This relationship is depicted in Figure 9.

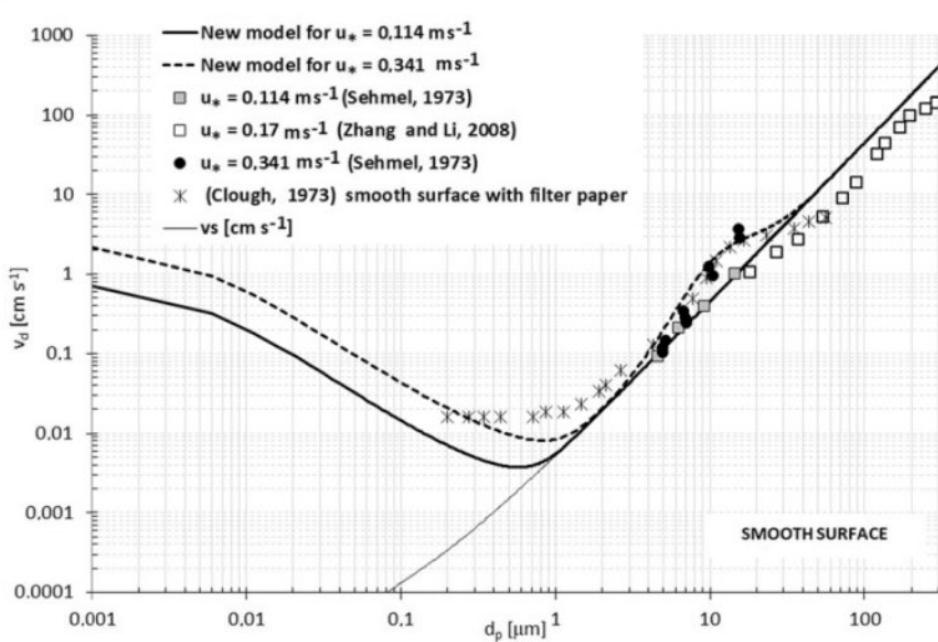


Figure 12 Relationship between dry deposition velocity and particle size [50]

TID 14844 made several acknowledgments to the importance of particle size distributions and resulting deposition velocities in stating that the source term produces “vapors as well as liquid and solid aerosols of a wide range of sizes” some of which “may conceivably burn on contact with air, thus increasing the volatiles and fractions of fine particles” but did not account for deposition in their calculations. Furthermore, TID 14844 did not take into account any potential energy or heat in the release and modeled the source term as a ground level non-buoyant plume.

NUREG-1150 did consider particle sizes and deposition velocities but made the assumption that all particles (regardless of their size) deposited at the same velocity. It wasn’t until SOARCA that particle size distributions and corresponding deposition velocities were assigned to chemical groups. NUREG/CR-7110 [13][14] utilized 10 particle size groups based on mass median aerosol diameter sizes, which are reflected in Table 23 below:

Table 23 SOARCA particle size groups

Particle Size Group	Median Aerosol Diameter Size (μm) [14]
1	0.15
2	0.29
3	0.53
4	0.99
5	1.8
6	3.4
7	6.4
8	12
9	22
10	41

Although the exact particle size distributions were specific to the accident scenario being evaluated, there are some observations that can be gathered from the Surry SOARCA analysis that may be used for determining a bounding scenario. Table 24 through Table 28 show the fractions of the release for each chemical group, per each particle size groups.

Table 24 SOARCA Surry LTSBO particle size group distributions [14]

Surry LTSBO										
Class	Bin 1	Bin 2	Bin 3	Bin 4	Bin 5	Bin 6	Bin 7	Bin 8	Bin 9	Bin 10
Xe	1.00E-01									
Cs	1.73E-03	1.26E-02	6.19E-02	2.00E-01	3.36E-01	2.52E-01	1.09E-01	2.37E-02	2.32E-03	1.05E-03
Ba	6.94E-03	3.56E-02	1.35E-01	3.69E-01	3.45E-01	8.98E-02	1.49E-02	2.50E-03	2.54E-04	3.58E-04
I	6.47E-03	3.22E-02	1.21E-01	3.28E-01	3.57E-01	1.32E-01	1.85E-02	1.95E-03	3.19E-04	1.44E-03
Te	7.53E-03	3.45E-02	1.31E-01	3.49E-01	3.40E-01	1.14E-01	1.86E-02	2.44E-03	2.44E-04	1.34E-03
Ru	8.80E-03	3.73E-02	1.35E-01	3.24E-01	3.13E-01	1.27E-01	2.71E-02	1.02E-02	3.38E-03	1.40E-02
Mo	2.39E-04	3.89E-03	2.78E-02	1.01E-01	2.67E-01	3.30E-01	1.98E-01	6.34E-02	8.46E-03	3.76E-04
Ce	7.52E-03	3.22E-02	1.15E-01	2.85E-01	3.34E-01	1.73E-01	3.59E-02	7.97E-03	1.65E-03	8.02E-03
La	4.89E-03	2.37E-02	9.23E-02	2.54E-01	3.44E-01	2.04E-01	6.15E-02	1.20E-02	1.34E-03	2.86E-03

Table 25 SOARCA Surry Unmitigated STSBO particle size group distributions [14]

Surry Unmitigated STSBO										
Class	Bin 1	Bin 2	Bin 3	Bin 4	Bin 5	Bin 6	Bin 7	Bin 8	Bin 9	Bin 10
Xe	1.00E-01									
Cs	1.22E-03	1.01E-02	5.26E-02	1.73E-01	3.33E-01	2.86E-01	1.21E-01	2.05E-02	1.15E-03	1.19E-03
Ba	7.02E-03	3.61E-02	1.42E-01	3.43E-01	3.18E-01	1.19E-01	2.65E-02	3.58E-03	2.58E-04	4.40E-03
I	6.19E-03	2.94E-02	1.06E-01	2.66E-01	3.44E-01	2.03E-01	3.96E-02	2.54E-03	1.97E-04	2.36E-03
Te	4.03E-03	2.31E-02	9.45E-02	2.70E-01	3.62E-01	1.94E-01	4.57E-02	4.65E-03	2.33E-04	1.50E-03
Ru	5.15E-03	2.69E-02	1.07E-01	2.75E-01	3.43E-01	1.80E-01	4.05E-02	6.86E-03	1.67E-03	1.36E-02
Mo	2.51E-04	4.22E-03	3.10E-02	1.14E-01	2.91E-01	3.42E-01	1.78E-01	3.66E-02	2.37E-03	9.36E-05
Ce	5.06E-03	2.57E-02	9.95E-02	2.57E-01	3.41E-01	2.04E-01	4.92E-02	6.85E-03	1.30E-03	9.52E-03
La	3.14E-03	1.80E-02	7.61E-02	2.18E-01	3.44E-01	2.45E-01	8.08E-02	1.18E-02	7.95E-04	2.68E-03

Table 26 SOARCA Surry ISLOCA particle size group distributions [14]

Surry ISLOCA										
Class	Bin 1	Bin 2	Bin 3	Bin 4	Bin 5	Bin 6	Bin 7	Bin 8	Bin 9	Bin 10
Xe	1.00E-01									
Cs	9.37E-03	2.56E-02	6.96E-02	2.73E-01	4.01E-01	1.78E-01	4.04E-02	2.62E-03	4.33E-05	5.48E-04
Ba	2.23E-02	4.72E-02	1.38E-01	2.92E-01	3.24E-01	1.41E-01	3.05E-02	2.46E-03	7.74E-05	2.58E-03
I	9.15E-03	2.41E-02	7.56E-02	2.81E-01	3.96E-01	1.72E-01	3.85E-02	2.53E-03	4.28E-05	6.21E-04
Te	1.28E-02	3.01E-02	8.65E-02	2.83E-01	3.82E-01	1.65E-01	3.72E-02	2.50E-03	4.58E-05	5.42E-04
Ru	1.13E-02	3.05E-02	9.69E-02	2.91E-01	3.75E-01	1.59E-01	3.34E-02	2.13E-03	3.54E-05	4.80E-04
Mo	1.02E-02	2.46E-02	6.13E-02	2.34E-01	3.80E-01	2.16E-01	6.41E-02	8.89E-03	7.41E-04	3.93E-04
Ce	7.90E-03	3.31E-02	2.20E-01	4.00E-01	1.92E-01	8.59E-02	4.09E-02	1.27E-02	1.26E-03	6.03E-03
La	2.07E-02	4.49E-02	2.46E-01	3.92E-01	1.74E-01	7.55E-02	3.43E-02	9.34E-03	8.08E-04	2.15E-03

Table 27 SOARCA Surry Mitigated TISGTR particle size group distributions [14]

Surry Mitigated TISGTR										
Class	Bin 1	Bin 2	Bin 3	Bin 4	Bin 5	Bin 6	Bin 7	Bin 8	Bin 9	Bin 10
Xe	1.00E-01									
Cs	2.39E-02	3.93E-02	6.33E-02	2.06E-01	3.47E-01	2.16E-01	7.68E-02	2.30E-02	4.35E-03	6.63E-04
Ba	1.29E-02	1.87E-02	5.47E-02	2.44E-01	4.15E-01	1.96E-01	4.50E-02	1.17E-02	2.38E-03	3.65E-04
I	3.45E-02	4.99E-02	6.35E-02	2.09E-01	3.47E-01	2.03E-01	6.82E-02	2.00E-02	3.72E-03	5.92E-04
Te	1.50E-02	2.65E-02	5.69E-02	2.17E-01	3.69E-01	2.16E-01	7.29E-02	2.20E-02	4.16E-03	6.11E-04
Ru	1.60E-03	5.33E-03	3.94E-02	2.20E-01	4.24E-01	2.24E-01	6.31E-02	1.87E-02	3.45E-03	4.43E-04
Mo	1.61E-02	5.22E-02	6.90E-02	2.98E-01	4.12E-01	1.16E-01	2.79E-02	6.85E-03	1.28E-03	1.92E-04
Ce	1.27E-03	8.45E-03	7.31E-02	2.80E-01	4.20E-01	1.79E-01	2.96E-02	7.03E-03	2.00E-03	2.70E-04
La	8.71E-03	2.82E-02	8.14E-02	2.99E-01	4.07E-01	1.45E-01	2.36E-02	5.32E-03	1.43E-03	1.89E-04

Table 28 SOARCA Surry Unmitigated TISGTR particle size group distributions [14]

Surry Unmitigated TISGTR										
Class	Bin 1	Bin 2	Bin 3	Bin 4	Bin 5	Bin 6	Bin 7	Bin 8	Bin 9	Bin 10
Xe	1.00E-01									
Cs	4.84E-03	1.55E-02	5.85E-02	1.91E-01	4.04E-01	2.69E-01	5.05E-02	5.54E-03	7.83E-04	1.17E-04
Ba	6.37E-03	2.58E-02	8.09E-02	2.35E-01	3.81E-01	2.25E-01	4.35E-02	2.94E-03	2.34E-04	4.00E-05
I	1.14E-02	2.75E-02	6.76E-02	2.02E-01	3.54E-01	2.49E-01	7.56E-02	1.12E-02	1.36E-03	2.20E-04
Te	8.56E-03	2.61E-02	7.12E-02	2.00E-01	3.56E-01	2.59E-01	7.19E-02	7.63E-03	6.94E-04	1.06E-04
Ru	1.80E-03	6.34E-03	4.20E-02	2.21E-01	4.19E-01	2.23E-01	6.39E-02	1.89E-02	3.48E-03	4.41E-04
Mo	3.95E-03	1.86E-02	6.60E-02	1.76E-01	3.29E-01	2.92E-01	1.04E-01	1.05E-02	3.31E-04	8.27E-05
Ce	4.18E-03	2.17E-02	9.15E-02	2.48E-01	3.59E-01	2.03E-01	5.65E-02	1.35E-02	2.38E-03	2.42E-04
La	2.27E-03	1.47E-02	6.50E-02	2.00E-01	3.99E-01	2.64E-01	5.12E-02	3.73E-03	2.51E-04	2.79E-05

With the exception of Xe which does not deposit due to its nature as a noble gas, each of the other chemical groups contains a mixture of particle sizes with a distribution centered between particle size groups 4 and 6, with 5 being the most common corresponding to a median aerosol diameter of 1.8 μm . An alternate approach to defining particle size distributions can be found in the NUREG-2161 spent fuel pool study where particles sizes are divided into the same particle size groups/bins as Table 24 through Table 28, but allocated based on percentages that are applied to all chemical groups. Table 29 provides this distribution method for particle sizes.

Table 29: NUREG-2161 Particle Size Groups [37]

Particle Size Group	Particle Size Distribution	Dry Deposition Velocity (m/s)
1	3.2%	0.0011
2	15%	0.001
3	29%	0.0014
4	21%	0.0023
6	10%	0.0045
6	3.0%	0.0092
7	1.50%	0.0177
8	0.60%	0.0291
9	0.20%	0.0367
10	16%	0.0367

Also in NUREG-2161, for health effects specifically, “a median aerodynamic diameter of 1 micron was assumed with a log-normal form for the distribution and with a geometric standard deviation of

about 2.5" [39]. In determining a bounding approximation for particle size distributions to represent a scenario with maximum dispersion, a particle size distribution closer to the minimum represented in Figure 9 would represent a conservative case for maximum dispersion. Therefore, the simple lognormal distribution utilized in NUREG-2161 would represent a realistic and conservative bounding scenario for particle size distributions.

6.3. Energy of Release

Characterizing the energy of a release is influential in that it determines the degree of buoyancy for the plume, and subsequently the plume rise. Similar to particle size distributions, initial studies took a very simplistic view of release energy, often modeling the release as a non-buoyant plume. Historical accident examples have shown that may not be the case, as energetic releases can cause radioactive particles to lift higher in the atmosphere, which impacts their subsequent transport and time before being deposited onto the ground.

These source terms are expected to be diffuse plumes of hot aerosol-filled gases leaving the failed confinement, containment, or SFP enclosure. These plumes have buoyancy but not explosive lofting. The plume source will have some diffuse qualities as it rises from the likely leakage locations (i.e., a broken pipe, relief tank, failed lower reactor head) and work their way through the damaged containment compartments to the environment. Turbulent mixing and entrainment should cool the plume as it rises from the building.

In order to characterize the buoyancy of the plume upon release, values of plume heat (W) from the SOARCA analysis from both Surry and Beach Bottom were examined for the starting plume segment, the final plume segment, and plume segments representative of the middle of release. Values are depicted in Table 30 below and vary by several orders of magnitude depending on the particular accident scenario. To represent a bounding approximation, the Peach Bottom Long Term Station Blackout scenario is highlighted as a recommended bounding scenario. The recommendation from the analysis is that applying a constant average value of 1.4×10^6 Watts is a simple and straightforward method to represent the heat of release but could be refined for discrete plume segment values if desired.

Table 30: Bounding Plume Energies

Accident Scenario (from NUREG/CR-7009)	Start (W)	Mid Release (W)	End (W)
Surry LTSBO	5.11E+04	5.75E+05	6.27E+05
Surry STSBO	5.78E+04	6.71E+05	7.05E+05
Surry ISLOCA	3.65E+05	4.70E+04	2.69E+04
Surry TISGTR	4.32E+01	4.41E+04	1.01E+05
Surry Unmitigated TISGTR	4.32E+01	2.11E+05	1.95E+05
Peach Bottom LTSBO	9.18E+06	1.40E+06	1.24E+06
Peach Bottom STSBO w/ RCIC Blackstart	8.80E+06	1.30E+06	1.24E+06
Peach Bottom STSBO w/o RCIC Blackstart	1.94E+00	4.68E+00	8.90E+01

6.4. Summary of Source Term Parameters

Bounding source term parameters for each reactor type and scenario are summarized in Table 31, with the failed confinement scenario for the RBMK-1000 representing the highest value for total activity released. Additional source terms are located in the accompanying spreadsheet, which include additional variations on intact confinements/containments and spent fuel pools. These additional source terms were not considered bounding and therefore were not included in the summary table, but could be examined for additional studies if desired.

Table 31: Summary of Bounding Source Term Parameters

Release Scenario	Total Activity Released (Bq)*	Timing Considerations**	Energy (W)	Particle Sizes
RBMK-1000 Failed Confinement	6.35E+19	Start: .5-2 hrs Duration: 4.5-168 hrs	1.40E+06	(See Section 6.2)
RBMK-1000 Intact Confinement	3.66E+19	Start: .5-2 hrs Duration: 4.5-168 hrs	1.40E+06	(See Section 6.2)
VVER-440 Failed Containment: Ex-vessel w/DF	1.10E+19	Start: 1-4 hrs Duration: 9.5 hrs	1.40E+06	(See Section 6.2)
VVER-440 Failed Containment: Late in-vessel w/DF	2.72E+18	Start: + 4.7 hrs	1.40E+06	(See Section 6.2)
VVER-440 Intact Containment: Ex-vessel w/DF	7.25E+18	Start: 1-4 hrs Duration: 9.5 hrs	1.40E+06	(See Section 6.2)
VVER-440 Intact Containment: Late in-vessel w/DF	9.90E+17	Start: + 4.7 hrs	1.40E+06	(See Section 6.2)
PWR Failed Containment: Ex-vessel w/DF	2.74E+19	Start: 1-4 hrs Duration: 9.5 hrs	1.40E+06	(See Section 6.2)
PWR Failed Containment: Late in-vessel w/DF	6.32E+18	Start: + 4.7 hrs	1.40E+06	(See Section 6.2)
BWR Failed Containment: Ex-vessel w/DF	3.78E+19	Start: .5-1 hr Duration: 11 hrs	1.40E+06	(See Section 6.2)
BWR Failed Containment: Late in-vessel w/DF	8.77E+18	Start: + 8 hrs	1.40E+06	(See Section 6.2)

*Radionuclide breakdown can be found in accompanying spreadsheet

**Timing consideration details can be found in Section 5

7. RECOMMENDATIONS FOR FUTHER REFINEMENT

The following recommendations are suggested for further refinement of the bounding source term analysis.

- Chemical form of iodine (i.e., how much is gaseous)
- Long-term release after the thermal oxidation phase, especially in the RBMK-1000.
- Better characterization of the aerosol size distribution with a failed containment. An initial estimate is in Section 6.2. This could be done by simulating a MELCOR severe accident with a failed containment.
- Assessment of uncertainty in a failed confinement DF (i.e., onsite retention). An initial estimate is in Section 5. This could be done by simulating a MELCOR severe accident with a failed containment.
- Assessment of dispersed thermal plume characteristics for LLNL. This could be done by simulating a MELCOR severe reactor accident with a failed containment.
- Estimate of iodine content in a RBMK-1000 SFP & number of “cores”. A spreadsheet approach with refueling assumptions.
- Recommendations for the BN-800 and Candu-600 – i.e., the accidents in these smaller reactors were bounded by the other reactor types. Note, there is only 1xBN-800 and just a few Candu's outside of Canada and India.

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