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High Burnup Fuel Accident Source Terms

ACRS Briefing Nov 16, 2023

Presented by Lucas I. Albright and David L. Luxat

U.S. DEPARTMENT OF ENERGY NNSA

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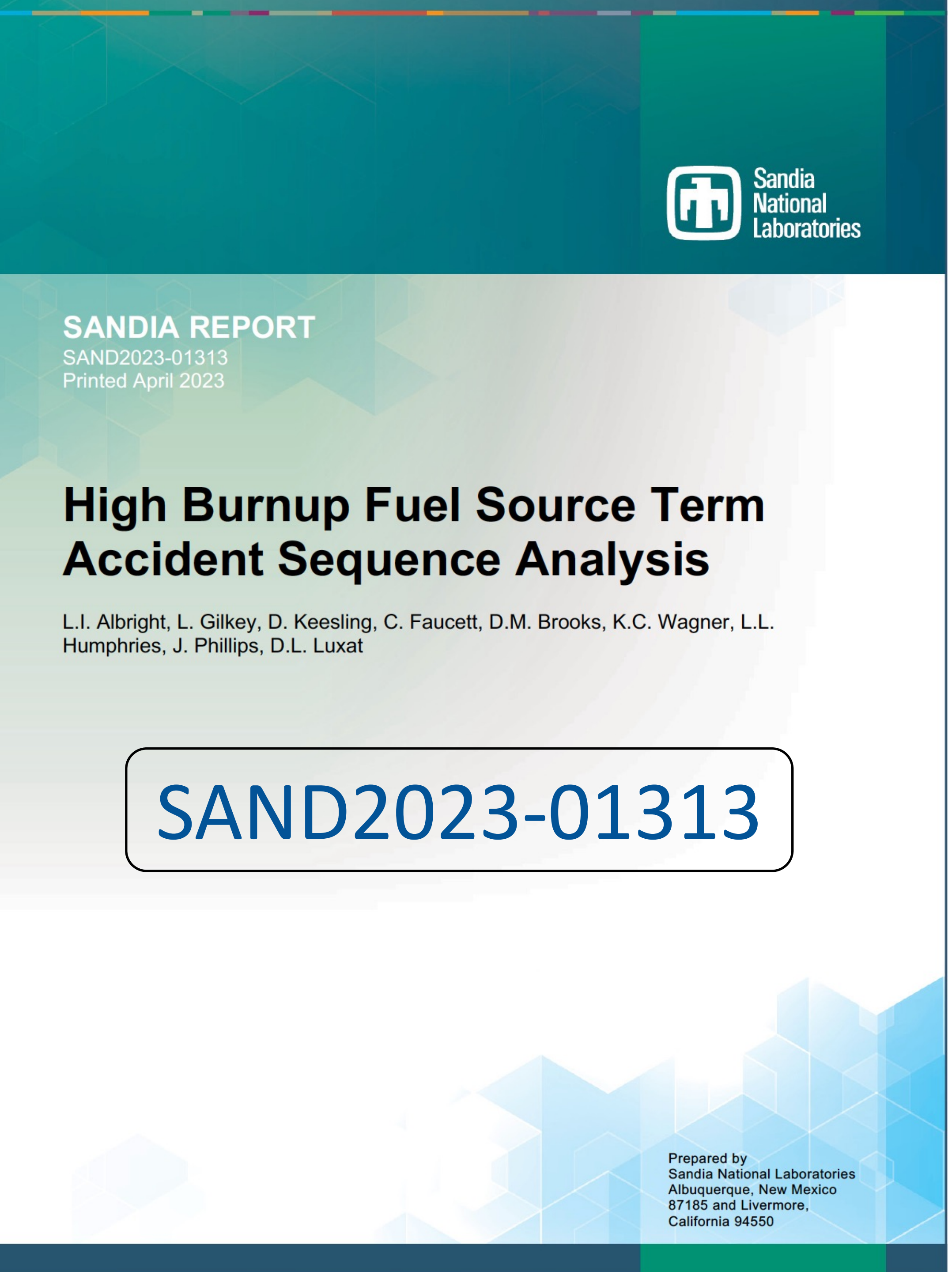
Motivation and Background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SANDIA REPORT
SAND2023-01313
Printed April 2023

High Burnup Fuel Source Term Accident Sequence Analysis

L.I. Albright, L. Gilkey, D. Keesling, C. Faucett, D.M. Brooks, K.C. Wagner, L.L. Humphries, J. Phillips, D.L. Luxat

SAND2023-01313

Prepared by
Sandia National Laboratories
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High Burnup Fuel Source Term Analysis Motivation

- Develop alternative source term applicable to LWR cores with HBU/HALEU fuel
  - Different burnup levels and enrichments considered
- Extends NUREG-1465 and SAND2011-0128 alternative source terms

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Historically Relevant Studies

- **TID-14844**: “Calculation of Distance Factors for Power and Test Reactors,” – USAEC 1962
- **NUREG-1465** – “Accident Source Terms for Light-Water Nuclear Power Plants,” – USNRC 1995 (code: STCP)
- **SAND2011-0128** – “Accident Source Terms for Light- Water Nuclear Power Plants Using High-Burnup or MOX Fuel” (code: MELCOR 1.8.5)

Source Term Timeline

Major Developments

- TID-14844
- MELCOR
- NUREG-1560
- RG 1.183
- Phébus FP
- Fukushima Daiichi
- SOARCA V1 Peach Bottom
- SOARCA V2 Surry
- BSAF
- SOARCA UAs
- HBU/HALEU/ATF SA PIRT
- ATF Source Term

'95 — '11 — '23

Accident Source Terms

NUREG-1465 → **SAND2011-0128** → **SAND2023-01313**

NUREG–1465

Accident Source Terms for Light-Water Nuclear Power Plants

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

L. Soffer, S. B. Burson, C. M. Ferrell, R. Y. Lee, J. N. Ridgely

SANDIA REPORT
SAND2011-0128
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January 2011

Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel

D.A. Powers, M.T. Leonard, R.O. Gauntt, R.Y. Lee, and M. Salay

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Participation in OECD/NEA International Programs

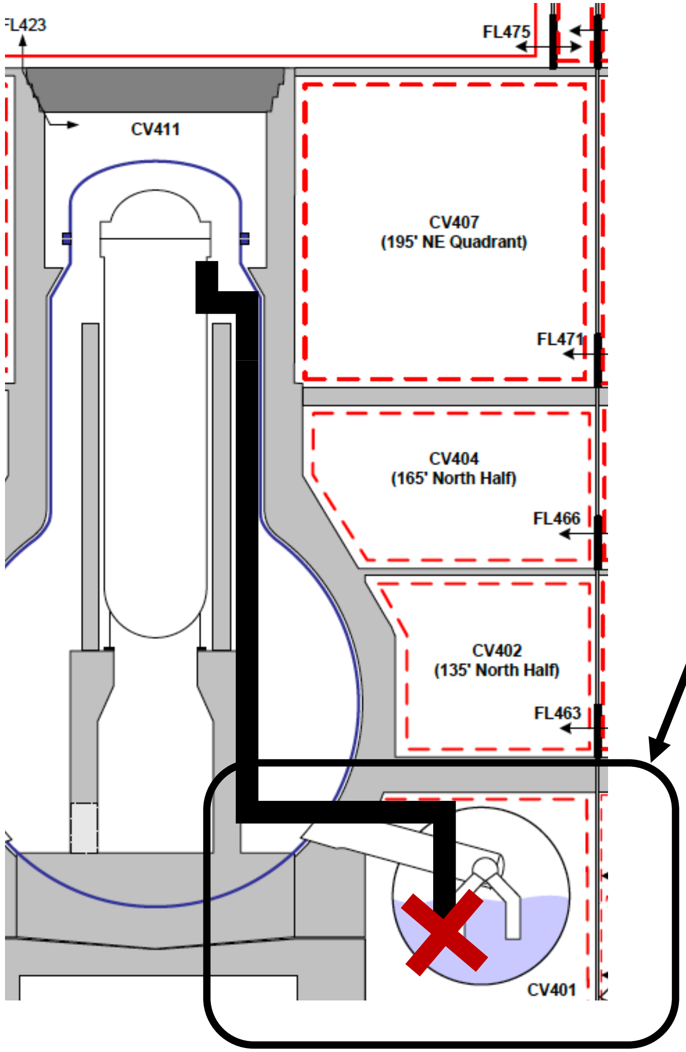
Severe Accident Modeling Advancements

- **Heterogeneous, integrated reactor core modeling** tends to promote to progressive and extended core degradation.
  - 2D discretization of the reactor core
  - No more distinct "gap release phase"
  - Prolonged core damage progression
  - Longer times to lower head failure

- **Prevalence of accident-induced low-pressure scenarios – SOARCA**
  - Thermally induced SRV seizure for majority of BWR sequences
  - Hot leg creep rupture for majority of PWR sequences

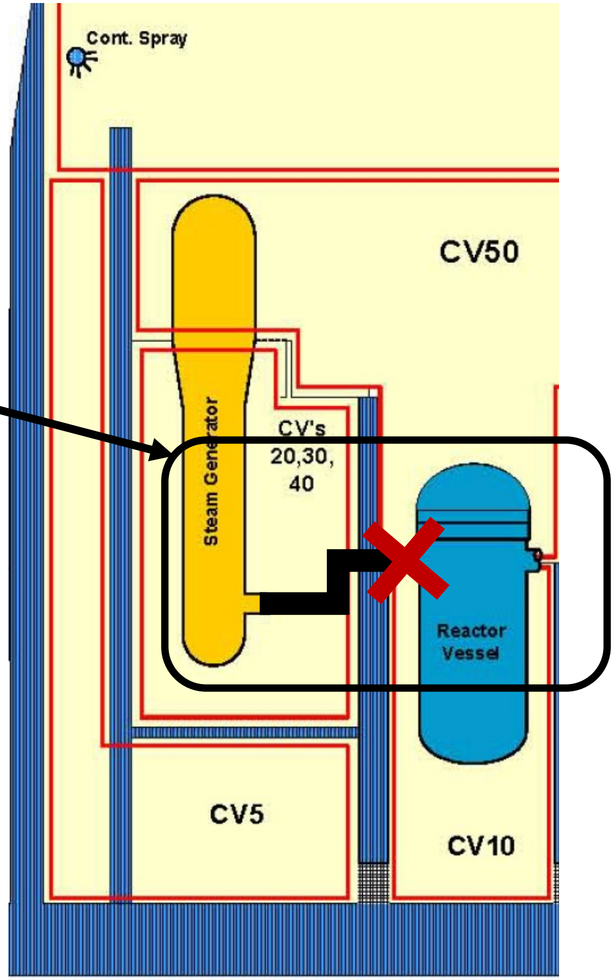
Impact of Early Depressurization

BWR: Thermally induced SRV seizure



Early loss of the primary pressure boundary induces depressurization of the reactor coolant system and opens a release pathway for radionuclides to transport directly to containment during early in-vessel core degradation

PWR: Hot Leg Creep Rupture



*Diagrams are for illustration purposes only

Selected Severe Accident Datasets

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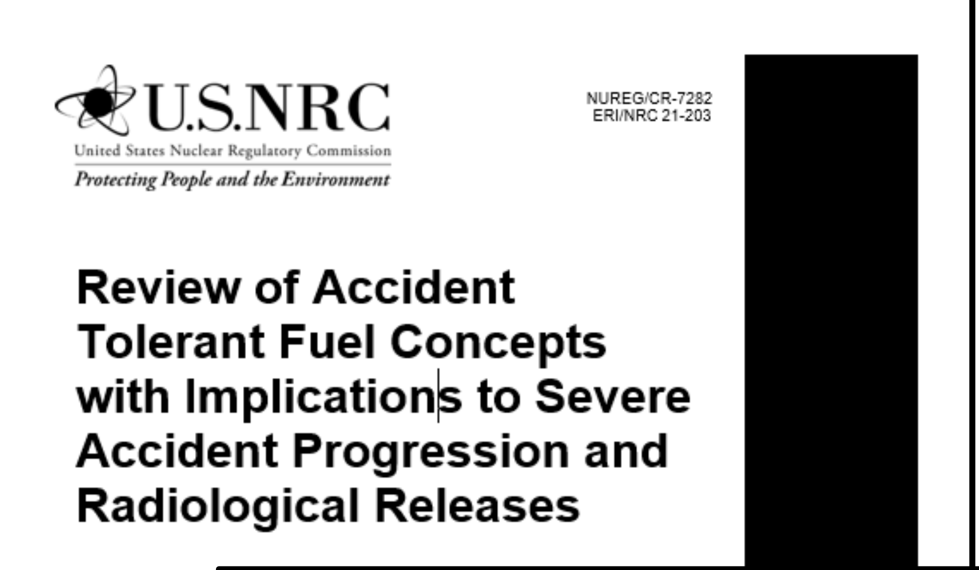
More recent severe accident datasets have improved characterization of core damage progression and subsequent radionuclide releases since NUREG-1465

- Severe accident experiments used to validate severe accident codes
  - Phébus FP
    - Early fuel failure
    - Hypothesized $CsMoO_4$ as the dominant chemical form of Cs
  - VERCORS
    - Early fuel failure
    - High burnup fission product release rates
- Severe accidents are a primary data source for severe accident code validation
  - Fukushima Daiichi
    - Existing data confirms that $CsMoO_4$ is the dominant chemical form of Cs

Severe Accident Knowledge Advancements

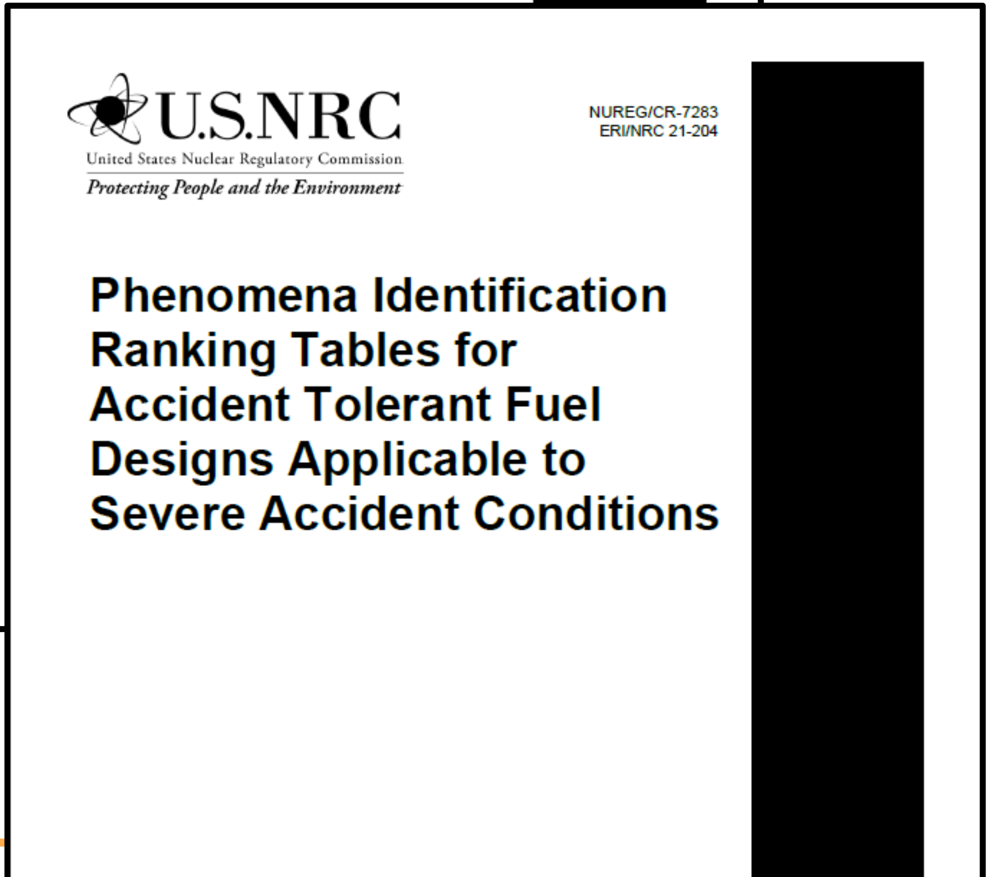
- **Chemical form of iodine:**
  - NUREG-1465 assumed 95% of iodine in the form of CsI
  - Current practice assumes all Iodine to be CsI
  - Still assume 5% of the total iodine inventory is present in the gap inventory
- **Chemical form of cesium:**
  - NUREG-1465 assumed Cs predominantly in the form of volatile CsOH
  - Current best-practice assumes 5% of cesium present in the gap inventory as both CsI and CsOH
  - All remaining cesium assumed to react with Mo to form Cs_2MoO_4
- **Mo release:**
  - Mo releases are now higher than other metallic fission products such as Ru and Pd.
- **Te release:**
  - Current best practice is more extensive Te release than reported in NUREG-1465
  - Due to change in chemical form with more efficient transport of Te to containment

HBU/HALEU/ATF PIRT


U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/CR-7282
ERI/NRC 21-203

Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases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NUREG/CR-7283
ERI/NRC 21-204

Phenomena Identification Ranking Tables for Accident Tolerant Fuel Designs Applicable to Severe Accident Conditions

- **HBU/HALEU fuel severe accident behavior**
  - No significant differences between HBU and HBU/HALEU fuels
  - Thermophysical property differences expected
  - Fuel fragmentation and sintering can impact core degradation
  - Fission product chemistry may change
  - Possibility of cladding embrittlement
  - Greater potential for recriticality during reflood using unborated water for HALEU

Key Findings

Study Highlights

Key Finding 1: Increased burnup and enrichment does not strongly impact in-containment source term

- Most significant variation in source term arises due to differences between accident scenarios

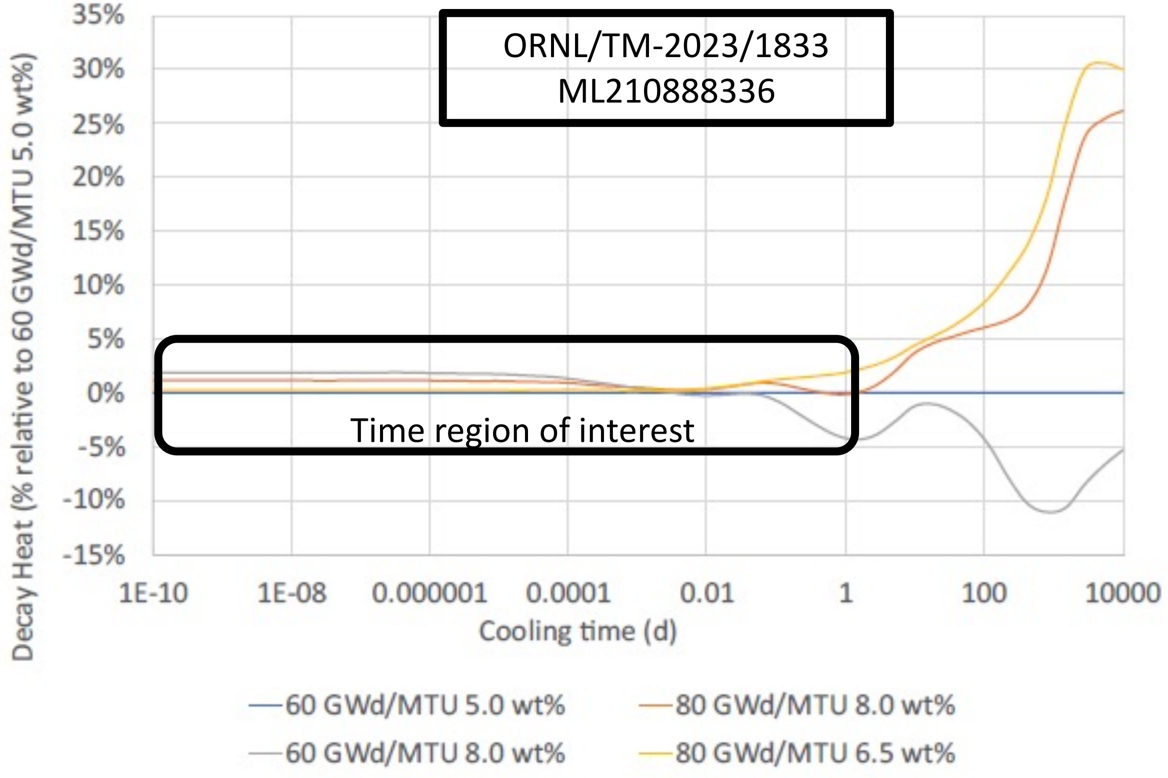
Key Finding 2: Larger early releases to containment result from early primary pressure boundary failure

- Set of accident scenarios dominated by low pressure accident sequences
- NUREG-1465 prescribed a larger number of high pressure scenarios

Key Finding 3: Releases to containment significantly reduced if primary pressure boundary remains intact

- Low pressure scenarios lead to more significant releases to containment
- Evolution of severe accident modeling state-of-art since NUREG-1465 (e.g., SOARCA)

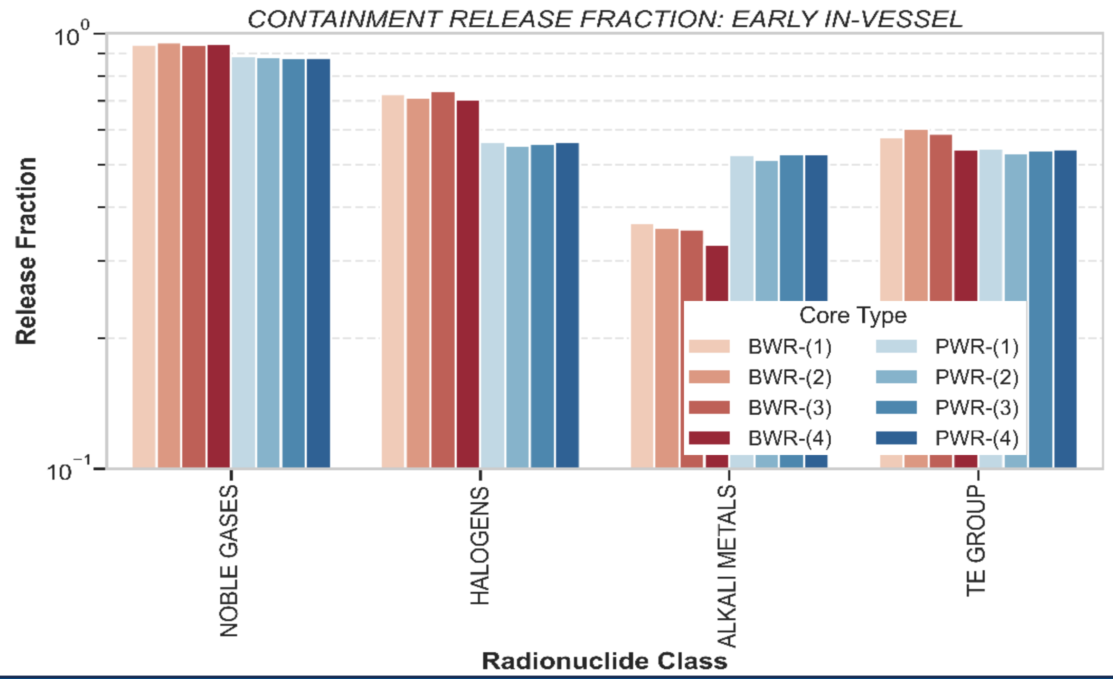
High Burnup and Extended Enrichment Impact on Source Term



Core Types:

(1) 60 GWd/MTU LEU (2) 80 GWd/MTU LEU

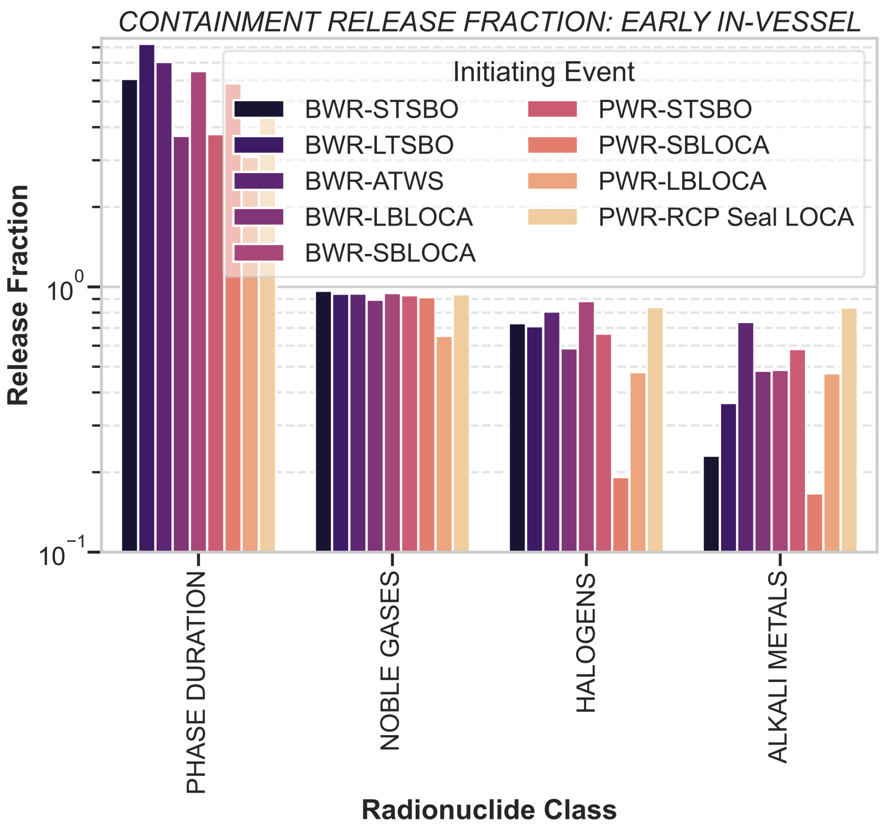
(3) 60 GWd/MTU HALEU (4) 80 GWd/MTU HALEU



Burnup and enrichment do not significantly change decay heat after reactor shutdown

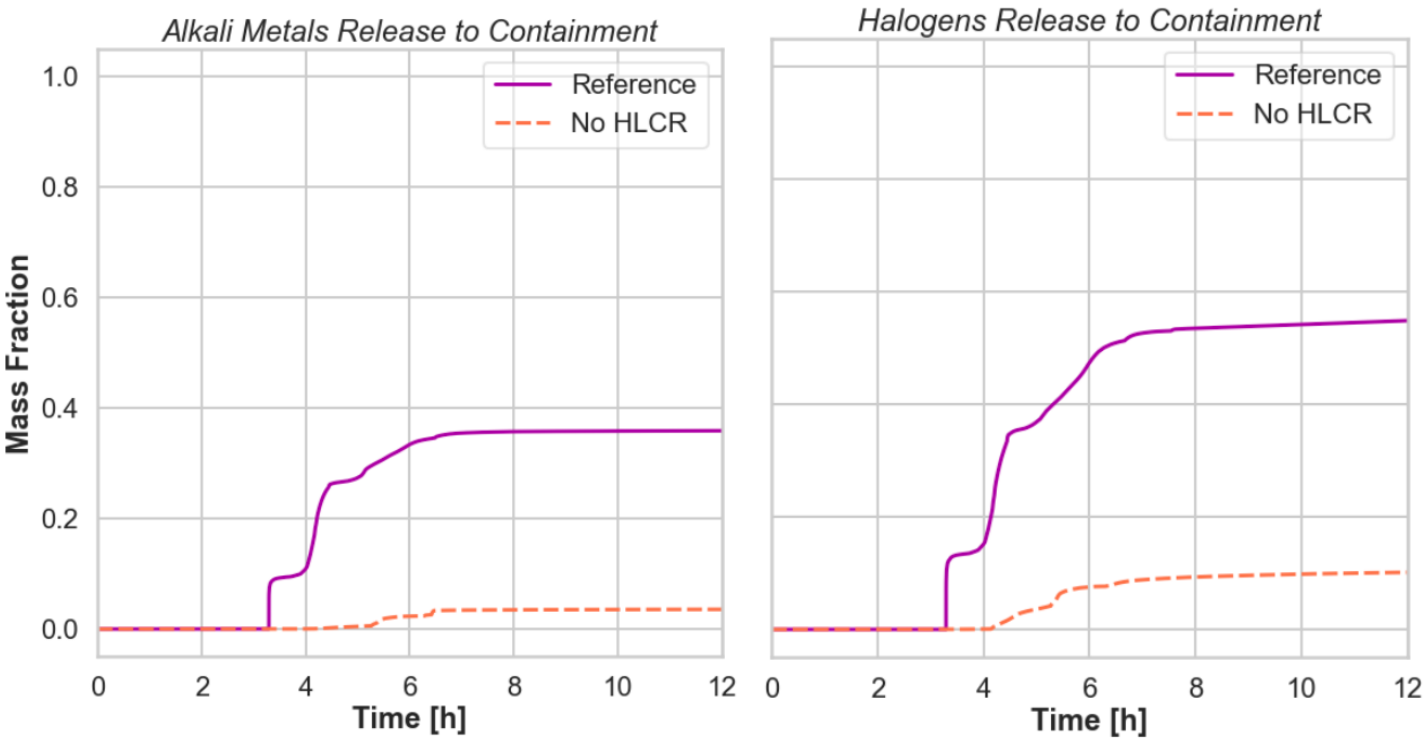
Increased burnup and enrichment does not strongly impact in-containment source term

Impact of Accident Scenarios on In-containment Source Term



Accident progression and in-containment source terms different across accident sequences

Reference	Hot leg creep rupture enabled
No HLCR	Hot leg creep rupture disabled



Primary pressure boundary failure during critical accident phases is a significant factor in accident progression and in-containment source term

In-Containment Source Term Differences

		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
BWR	Report	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	Phase Duration	0.70	0.50	6.7	1.5	44.6	10.0	3.1	3.0
	Noble Gases	0.016	0.050	0.95	0.95	0.005	0.0	0.011	0.0
	Halogens	0.005	0.050	0.71	0.25	0.16	[illegible]	[illegible]	.30
	Alkali Metals	0.005	0.050	0.32	0.20	0.021	[illegible]	[illegible]	.35
	Te Group	0.003	0.0	0.56	0.050	0.19	[illegible]	[illegible]	.25
PWR		Gap Release		Early In-vessel		Late			
	Phase Duration	1.3	0.50	4.0	1.3	24.0	[illegible]	[illegible]	2.0
	Noble Gases	0.026	0.050	0.93	0.95	0.010	[illegible]	[illegible]	.0
	Halogens	0.007	0.050	0.58	0.35	0.031	0.10	0.020	0.25
	Alkali Metals	0.003	0.050	0.50	0.25	0.013	0.10	0.015	0.35
	Te Group	0.006	0.0	0.55	0.050	0.019	0.005	0.005	0.25

The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Longer in-vessel phase durations due to progressive core degradation

In-Containment Source Term Differences

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	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
		2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
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The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Longer in-vessel phase durations due to progressive core degradation
- Progressive releases to containment due to enhanced reactor coolant system modeling

In-Containment Source Term Differences

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		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
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The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Longer in-vessel phase durations due to progressive core degradation
- Progressive releases to containment due to enhanced reactor coolant system modeling
- Larger release magnitudes prior to lower head failure due to early loss of the primary pressure boundary (by safety relief valve seizure and hot leg creep rupture)

In-Containment Source Term Release Rates

		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
BWR	Report	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	Phase Duration	0.70	0.50	6.7	1.5	44.6	10.0	3.1	3.0
	Noble Gases	0.023	0.10	0.14	0.63	0.0001	0.0	0.003	0.0
	Halogens	0.007	0.10	0.11	0.17	0.004	[illegible]	[illegible]	[illegible]
	Alkali Metals	0.007	0.10	0.047	0.13	0.0006	[illegible]	[illegible]	[illegible]
	Te Group	0.005	0.0	0.091	0.033	0.005	[illegible]	[illegible]	[illegible]
PWR		Gap Release		Early In-vessel		Late [illegible]		[illegible]	
	Phase Duration	1.3	0.50	4.0	1.3	24.0	[illegible]	[illegible]	[illegible]
	Noble Gases	0.019	0.10	0.21	0.73	0.0008	[illegible]	[illegible]	[illegible]
	Halogens	0.003	0.10	0.16	0.27	0.001	0.010	0.009	0.12
	Alkali Metals	0.001	0.10	0.15	0.19	0.0005	0.010	0.008	0.17
	Te Group	0.003	0.0	0.15	0.038	0.0008	0.0005	0.002	0.12

The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Assumes uniform release rate across the entire phase duration

In-Containment Source Term Release Rates

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BWR

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PWR

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- Assumes uniform release rate across the entire phase duration
- Release rates (release fraction/hour) are generally smaller

In-Containment Source Term Release Rates

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BWR		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
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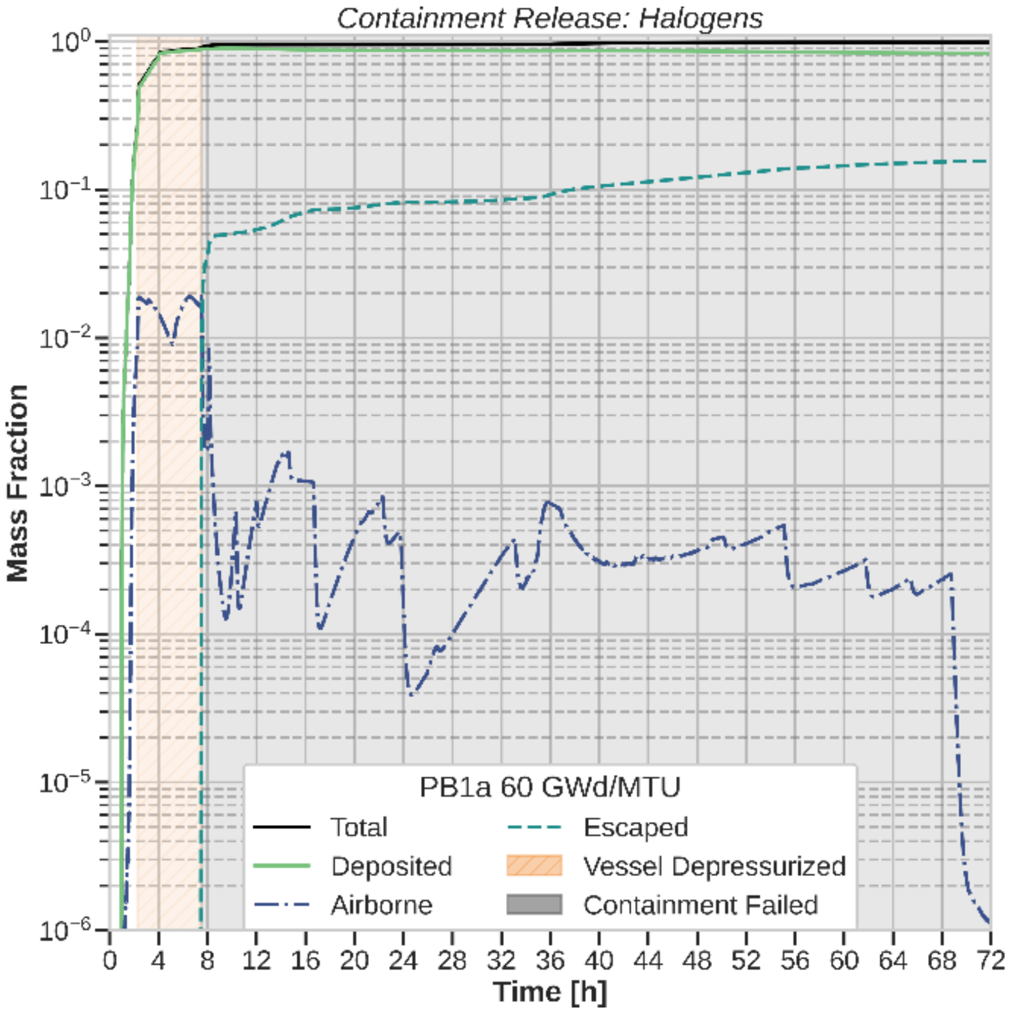
- Assumes uniform release rate across the entire phase duration
- Release rates (release fraction/hour) are generally smaller
- Larger Te group release magnitude prior to lower head failure

Deep Dive

In-containment Source Term

- In-containment source term characterizes **total** radioactive inventory in containment
  - In-containment source term combines deposited, airborne, and escaped radionuclide inventories
- MELCOR simulations can track deposited and airborne masses separately
  - This additional information not used in determining in-containment source term
- Radionuclide removal mechanisms accounted for in downstream calculations with RADTRAD



10 CFR 50.2 – Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release

Alternative Source Term

"Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183

SAND2023-01313 Peer Review Assessment (ERI/NRC 23-201)

Fulfills Criteria	Alternative Source Term (AST) must be based on major accidents involving a substantial meltdown of the core
Fulfills Criteria	AST must be represented in terms of the quantities, times, rates, chemical speciation for fission product release into containment
Fulfills Criteria	AST must not based on a single accident scenario but characterizes a spectrum of credible severe accident events
Fulfills Criteria	AST must have a defensible technical basis
Fulfills Criteria	AST must be peer reviewed

Process for Source Term Development

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BWR and PWR core damage accident scenario identification

Develop radionuclide inventory and decay heat using the SCALE code package

Perform accident progression and source term analyses using MELCOR

Develop statistically representative source term across all accident scenarios and BWR/PWR plants

Evolution from SAND2011-0128

- Overall SAND2023-01313 methodology is consistent with SAND2011-0128
  - Focus on assessing impact of HBU/HALEU fuel on alternative source term
- Key areas of consistency between the studies are
  - Nuclear power plants modeled
  - Accident scenarios simulated
  - Radionuclide chemical classes represented
  - Radiological release phases first identified in NUREG-1465 are defined using SAND2011-0128 criteria
  - Representative release phase source terms and timings are statistical median values

Extending SAND2011-0128 Source Terms

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- Plants analyzed – *from SAND2011-0128*
  - BWR: Mark I containment (Peach Bottom) and Mark III containment (Grand Gulf)
  - PWR: Ice Condenser containment (Sequoyah) and large-dry containment (Surry)
- Accident scenarios analyzed – *from SAND2011-0128*
  - BWR: SBLOCA, LBLOCA, STSBO, LTSBO, ATWS
  - PWR: SBLOCA, LBLOCA, STSBO

Phase	Onset Criteria – *from SAND2011-0128*	End Criteria – *from SAND2011-0128*
Gap Release	RPV water level below top of active fuel	Release of 5% of initial, total Xe inventory from fuel
Early In-Vessel	Release of 5% of initial, total Xe inventory from fuel	Lower Head Failure
Ex-Vessel	Lower Head Failure	95% of total ex-vessel Cs releases
Late In-Vessel	Lower Head Failure	95% of total late in-vessel Cs releases

Peer Review Findings

- Ex-vessel and late in-vessel phase criteria have limited technical justification
- NRC determined (SECY-94-302, December 19, 1994) design basis source terms will not include ex-vessel and late in-vessel phases

SAND2023-01313 Accident Selection

- NUREG-1560 – "Individual Plant Examination Program"
- Based on SAND2011-0128 accident selection
  - Consistent with NUREG 1560 IPE results
- Representative accident sequences similar to those selected for NUREG-1465
  - Provides coverage of all major sequences
- Incorporates SBO, LOCA and ATWS scenarios and range of mitigating system operation

Peer Review Findings

- More recent PRA studies may potentially show different core damage contributors
- For the intended applications the scenarios used in the current [SAND2023-01313] appropriate with regards to the progression of severe accidents, radionuclide release and transport.

Peach Bottom Accident Scenarios

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Initiating Events

7 SBOs

- 4 immediate loss of DC power
- 3 with prolonged DC power

2 LOCAs

Coolant Injection

RCIC operation

- 3 scenarios credited RCIC

No coolant injection

- 6 scenarios had no credit for any coolant injection system

RPV Status

RPV at low pressure prior to lower head failure

- 8 low pressure scenarios

RPV at high pressure until lower head failure

- 1 high pressure scenario

Containment Status

Early failures

- Drywell liner melt-through
- Torus overpressure
- Drywell head flange leakage

Late Failure

- High-temperature penetration failure

Containment failures occurred at or after lower head failure

Grand Gulf Accident Scenarios

Initiating Events

5 SBOs

1 ATWS

1 LOCA

- Recirculation line break

Coolant Injection

RCIC operation

- 3 scenarios credited RCIC

No coolant injection

- 4 scenarios had no credit for any coolant injection system

RPV Status

RPV at low pressure prior to lower head failure

- 6 low pressure scenarios

RPV at high pressure until lower head failure

- 1 high pressure scenario

Containment Status

Early failures

- Hydrogen deflagration
- High containment pressure (ATWS)

Late Failure

- High containment pressure

Containment failures generally occurred at or after lower head failure

Surry Accident Scenarios

Initiating Events

2 SBOs

3 LOCAs

Coolant Injection

1 scenario crediting coolant injection

4 scenarios with no coolant injection

RPV Status

RPV at low pressure prior to lower head failure

- 3 low pressure scenarios

RPV at high pressure in SBOs

- **All** SBOs exhibit hot-leg creep rupture prior to lower head failure

Containment Status

Early failures

- Hydrogen deflagration

Late Failure

- High containment pressure

Containment failures occurred at or after lower head failure

Sequoyah Accident Scenarios

Initiating Events

2 SBOs

5 LOCAs

Coolant Injection

5 scenarios crediting coolant injection

2 scenarios with no coolant injection

RPV Status

RPV at low pressure prior to lower head failure

- 2 low pressure scenarios

RPV at high pressure in SBOs

- **All** SBOs and RCP seal LOCAs exhibit hot-leg creep rupture prior to lower head failure

Containment Status

Early failures

- Hydrogen deflagration

Late Failure

- High containment pressure

Containment failures occurred at or after lower head failure

BWR Radionuclide Inventories

Class (kg)	60 GWd/MTU - 5 wt% Enrichment	80 GWd/MTU - 5 wt% Enrichment	60 GWd/MTU - 10 wt% Enrichment	80 GWd/MTU - 10 wt% Enrichment
BWR Mark I – Peach Bottom				
Noble Gases	1323.99	1848.13 (+40%)	1280.34 (-3%)	1790.12 (+35%)
Halogens	52.83	73.70 (+40%)	49.41 (-6%)	69.53 (+32%)
Alkali Metals	748.78	980.11 (+31%)	817.97 (+9%)	1082.33 (+45%)
Te Group	142.94	195.01 (+36%)	139.99 (-2%)	190.51 (+33%)
Ba/Sr Group	551.99	763.09 (+38%)	586.41 (+6%)	814.05 (+47%)
Ru Group	1058.01	1598.56 (+51%)	919.02 (-13%)	1374.61 (+30%)
Mo Group	973.05	1305.64 (+34%)	1007.92 (+4%)	1364.59 (+40%)
Lanthanides	2943.70	3702.34 (+26%)	2922.84 (-1%)	3686.46 (+25%)
Ce Group	2469.33	2916.84 (18%)	2559.90 (+4%)	3107.02 (+26%)
*percent differences shown relative to reference core (60 GWd/MTU - 5 wt% enrichment) ** all fuel bundles assumed to reach reported burnup				

Peer Review Findings

- **Radionuclide class mass** differences are not equal to **radionuclide class activity** differences for the considered enrichments and burnups
- Unlikely that siting calculations would be significantly impact by burnup

PWR Radionuclide Inventories

Class (kg)	60 GWd/MTU - 5 wt% Enrichment	80 GWd/MTU - 5 wt% Enrichment	60 GWd/MTU - 8 wt% Enrichment	80 GWd/MTU - 8 wt% Enrichment
PWR with Large-Dry Containment – Surry				
Noble Gases	740.20	987.15 (+33%)	717.66 (-3%)	959.00 (+30%)
Halogens	29.31	39.35 (+34%)	27.44 (-6%)	37.06 (+26%)
Alkali Metals	421.27	537.41 (+28%)	455.26 (+8%)	584.21 (+39%)
Te Group	74.62	99.01 (+33%)	73.02 (-2%)	96.81 (+30%)
Ba/Sr Group	305.28	401.76 (+32%)	323.92 (+6%)	428.01 (+40%)
Ru Group	559.35	807.23 (+44%)	487.92 (-13%)	701.18 (+25%)
Mo Group	530.59	689.06 (+30%)	546.71 (+3%)	714.95 (+35%)
Lanthanides	1035.01	1396.16 (+35%)	1048.46 (+1%)	1409.24 (+36%)
Ce Group	1535.14	1780.67 (+16%)	1599.41 (+4%)	1903.19 (+24%)

*percent differences shown relative to reference core (60 GWd/MTU - 5 wt% enrichment)
** all fuel bundles assumed to reach reported burnup

Peer Review Findings

- **Radionuclide class mass** differences are not equal to **radionuclide class activity** differences for the considered enrichments and burnups
- Unlikely that siting calculations would be significantly impact by burnup

Iodine and Cesium Chemical Form

- NUREG-1465
  - 5% Iodine inventory is gaseous (I_2 and other organic iodides)
  - 95% Iodine inventory is CsI
  - Remaining Cs inventory assumed volatile (CsOH)
- SAND2023-01313 – consistent with SOARCA
  - 100% Iodine inventory reacts with Cesium to form CsI
  - 5% of the total Iodine and Cesium inventory present in gap
  - Of Cesium not forming CsI
    - 5% assumed to form CsOH
    - 95% assumed to form Cs_2MoO_4

Peer Review Findings

- Uncertainty in Iodine speciation persists despite experimental studies (FPT3, DF-4, and BECARRE)
- Fukushima Daiichi post-accident analyses confirm assumption that Cs_2MoO_4 is dominant chemical form of Cs
- Recommended consideration of/validation against French CEA HBU VERDON tests

Other Analysis Assumptions

- In-containment source term does not consider impact of
  - Variation in the gap inventory at the start of the accident
  - Fraction of aerosolized iodine in containment
  - Radionuclide removal and retention in containment
- Source term analyses based on current state-of-the-art
  - Latest major code version – MELCOR 2.2
  - Majority of modeling best-practices established under SOARCA
- Some modeling best-practices have evolved since SOARCA
  - Time-at-temperature fuel rod failure model uses default time-at-temperature fuel rod lifetime curve
  - UO_2 and ZrO_2 liquefaction temperatures reduced to 2479 K to account for material interactions
  - Failure temperature of oxidized fuel rods have been reduced to 2479 K

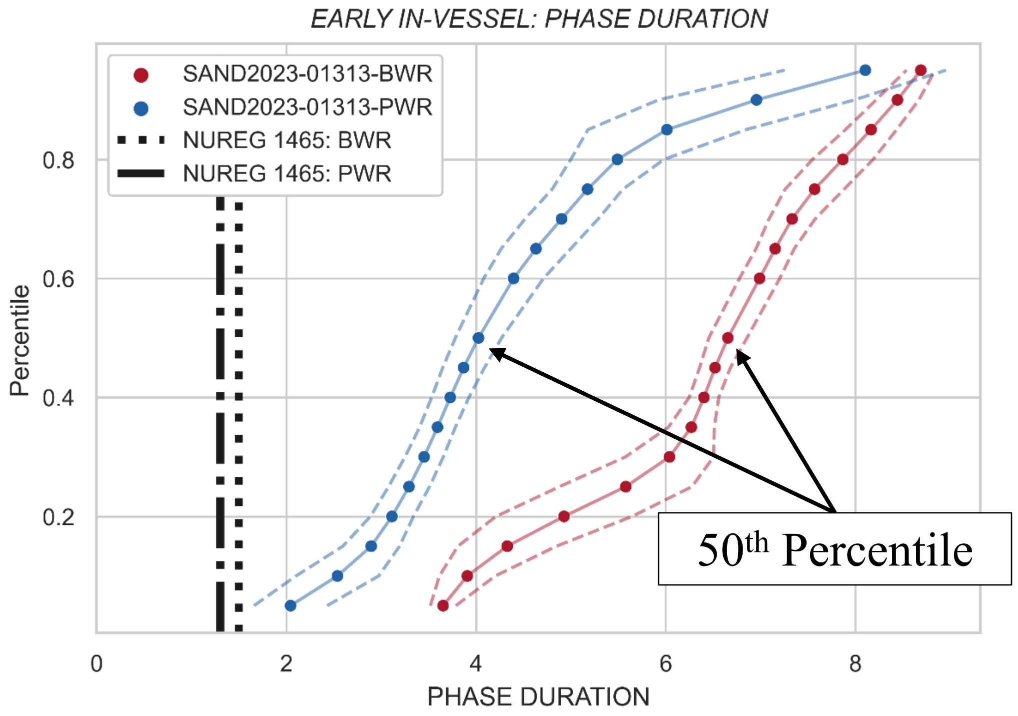
Other Analysis Assumptions

- Relative contribution of accident sequences to total BWR/PWR CDF not changed by cores with extended enrichment HBU
- Dominant uncertainty from range of possible accidents that could be realized (i.e., aleatory uncertainty)
  - Phenomenological (or epistemic) uncertainty not incorporated into BWR/PWR in-containment source terms
  - Impact of phenomenological uncertainties considered through sensitivity calculations
  - Key phenomena identified in a PIRT study are investigated through sensitivity studies
- Containment removal mechanisms not credited
  - Some removal mechanisms, such as containment sprays, are incorporated in downstream RADTRAD calculations
  - Suppression pool scrubbing not credited
- Release fractions (source terms) below 1×10^{-6} considered negligibly small and truncated

Non-Parametric Statistical Analysis

MELCOR

- Non-parametric bootstrap methodology used to determine statistically representative source term across accident scenarios
  - Can be applied to data that follow any distribution
  - Utilizes repeated re-sampling (bootstrapping) of data
  - Estimates empirical cumulative distribution function (ECDF) of a given quantity of interest (QoI)
- Representative source term is the median (50th percentile) estimate from the ECDF
  - Equally weights all simulations

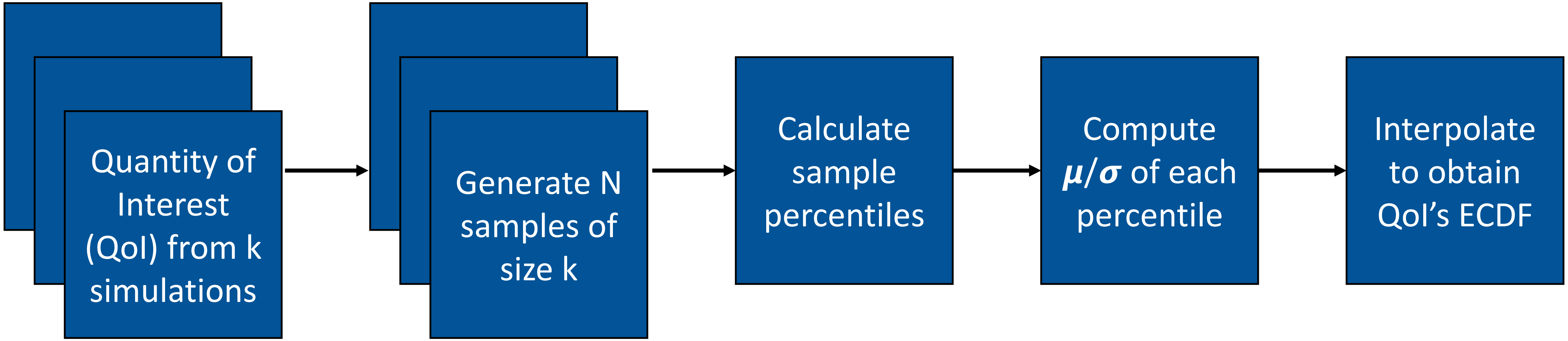


*Dashed colored lines illustrate confidence intervals spanning ± standard deviation ($\boldsymbol{\sigma}$) at each percentile

Peer Review Finding

- Representative source term based on median value appropriate to avoid introducing bias from potential outliers

Bootstrap Procedure



- Incorporates variability due to different plants and accident scenarios in representative source term
  - Bounds on empirical cumulative distribution function (ECDF) characterize sampling uncertainty

Results and Discussion

Restating Key Aspects of the Analysis

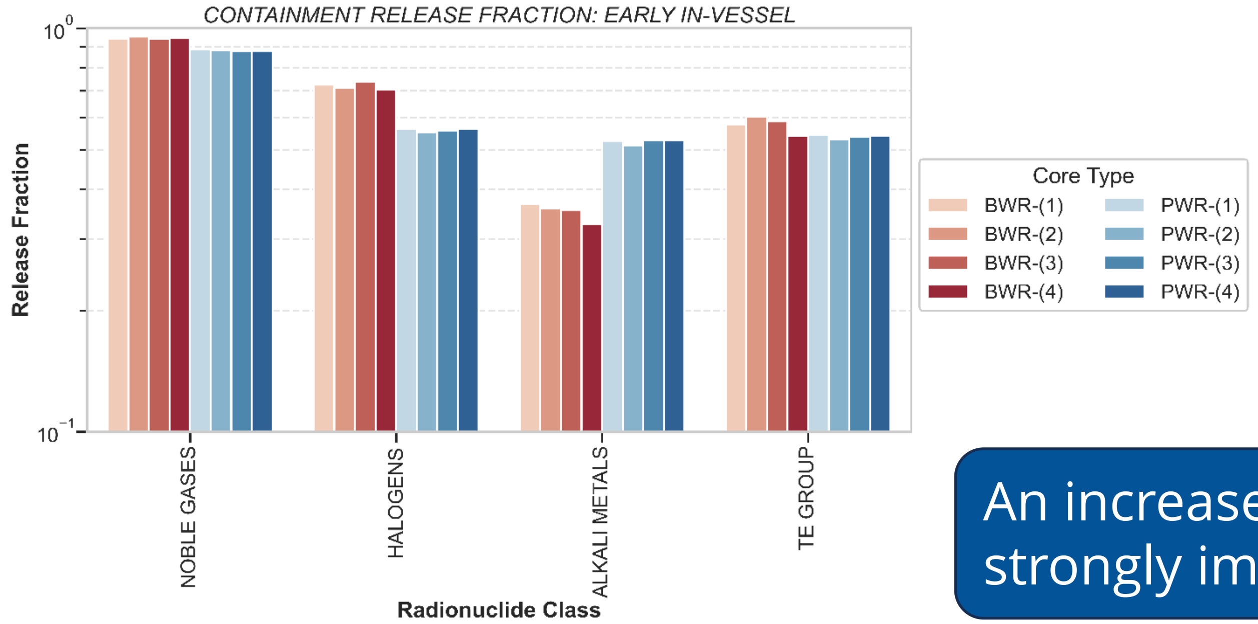
- Objective
  - Extend the NUREG-1465 alternative source term to address LWRs with cores designed to utilize HBU fuel with varied fuel enrichments
- Plants analyzed
  - BWR: Mark I containment (Peach Bottom) and Mark III containment (Grand Gulf)
  - PWR: Ice Condenser containment (Sequoyah) and Large-dry containment (Surry)
- Reactor cores analyzed
  1. Core average burnup of 60GWd/MTU for enrichment of 5 wt%
  2. Core average burnup of 80GWd/MTU for enrichment of 5 wt%
  3. Core average burnup of 60GWd/MTU for enrichment of 8 wt% (peak 10 wt% for BWRs)
  4. Core average burnup of 80GWd/MTU for enrichment of 8 wt% (peak 10 wt% for BWRs)
- Accident scenarios analyzed
  - BWR: SBLOCA, LBLOCA, STSBO, LTSBO, ATWS
  - PWR: SBLOCA, LBLOCA, STSBO

Phase	Onset Criteria	End Criteria
Gap Release	RPV water level below top of active fuel	Release of 5% of initial, total Xe inventory from fuel
Early In-Vessel	Release of 5% of initial, total Xe inventory from fuel	Lower Head Failure
Ex-Vessel	Lower Head Failure	95% of total ex-vessel Cs releases
Late In-Vessel	Lower Head Failure	95% of total late in-vessel Cs releases

Revisiting the Impact of Reactor Core on In-containment Source Term

BWR		Early In-vessel			
	Core Type	(1)	(2)	(3)	(4)
	Phase Duration	6.7	6.3	6.5	6.3
	Noble Gases	0.94	0.96	0.94	0.94
	Halogens	0.71	0.71	0.76	0.71
	Alkali Metals	0.31	0.31	0.31	0.26

PWR		Early In-vessel			
	Core Type	(1)	(2)	(3)	(4)
	Phase Duration	4.0	3.8	4.2	3.8
	Noble Gases	0.93	0.92	0.91	0.92
	Halogens	0.57	0.56	0.57	0.58
	Alkali Metals	0.5	0.5	0.5	0.51



Core Types:

(1) 60 GWd/MTU LEU,

(2) 80 GWd/MTU LEU

(3) 60 GWd/MTU HALEU

(4) 80 GWd/MTU HALEU

An increase in burnup and enrichment does not strongly impact the in-containment source term

BWR In-containment Source Term Evolution

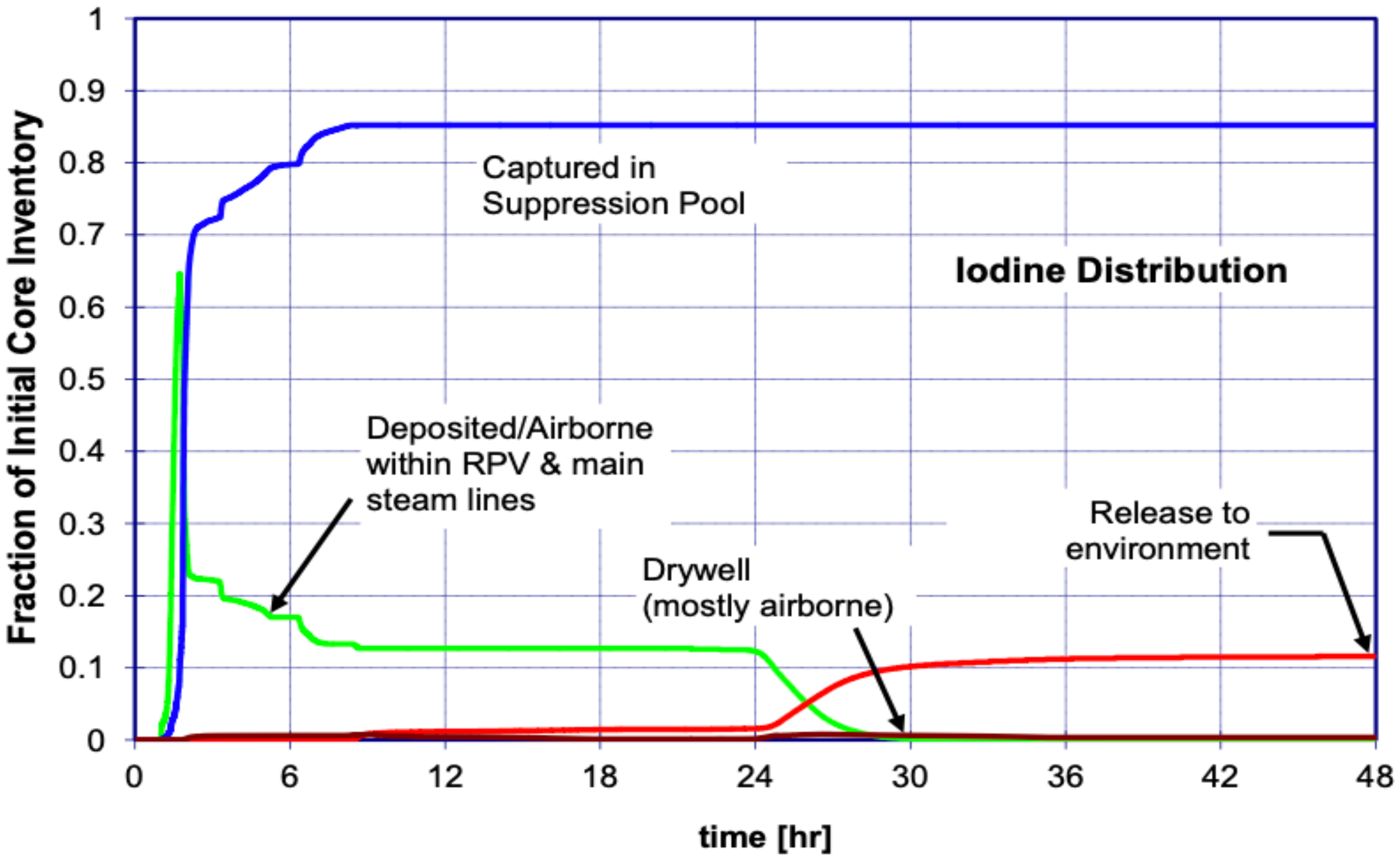
	Gap Release			Early In-vessel		
Study	**2023**	**2011**	**NUREG-1465**	**2023**	**2011**	**NUREG-1465**
Phase Duration (hr)	0.70	0.16	0.50	6.7	8.0	1.5
Noble Gases	0.016	0.008	0.050	0.95	0.96	0.95
Halogens	0.005	0.002	0.050	0.71	0.47	0.25
Alkali Metals	0.005	0.002	0.050	0.32	0.13	0.20
Te Group	0.003	0.002	0.0	0.56	0.39	0.050
Ba/Sr Group	0.0006	0.0	0.0	0.005	0.005	0.020
Ru Group	<1.0e-6	0.0	0.0	0.006	0.003	0.003
Mo Group	1.9E-05	0.0	0.0	0.12	0.020	0.003
Lanthanides	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0002
Ce Group	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0005

- SAND2023-01313 and SAND2011-0128 utilized MELCOR
- Accident scenarios and modeling best-practices lead to tendency for increased early in-vessel halogen releases
- Peach Bottom and Grand Gulf modeling best-practices in SAND2023-01313 represent improvements due to SOARCA

BWR In-Containment Source Terms Consistent with SOARCA

- SOARCA found limited in-vessel halogen retention during early-in vessel phase

PB SOARCA halogen releases (STSBO without RCIC blackstart)



*In-containment source terms reported in SAND2023-01313 characterize **total** radioactive inventory in containment

PWR In-containment Source Term Evolution

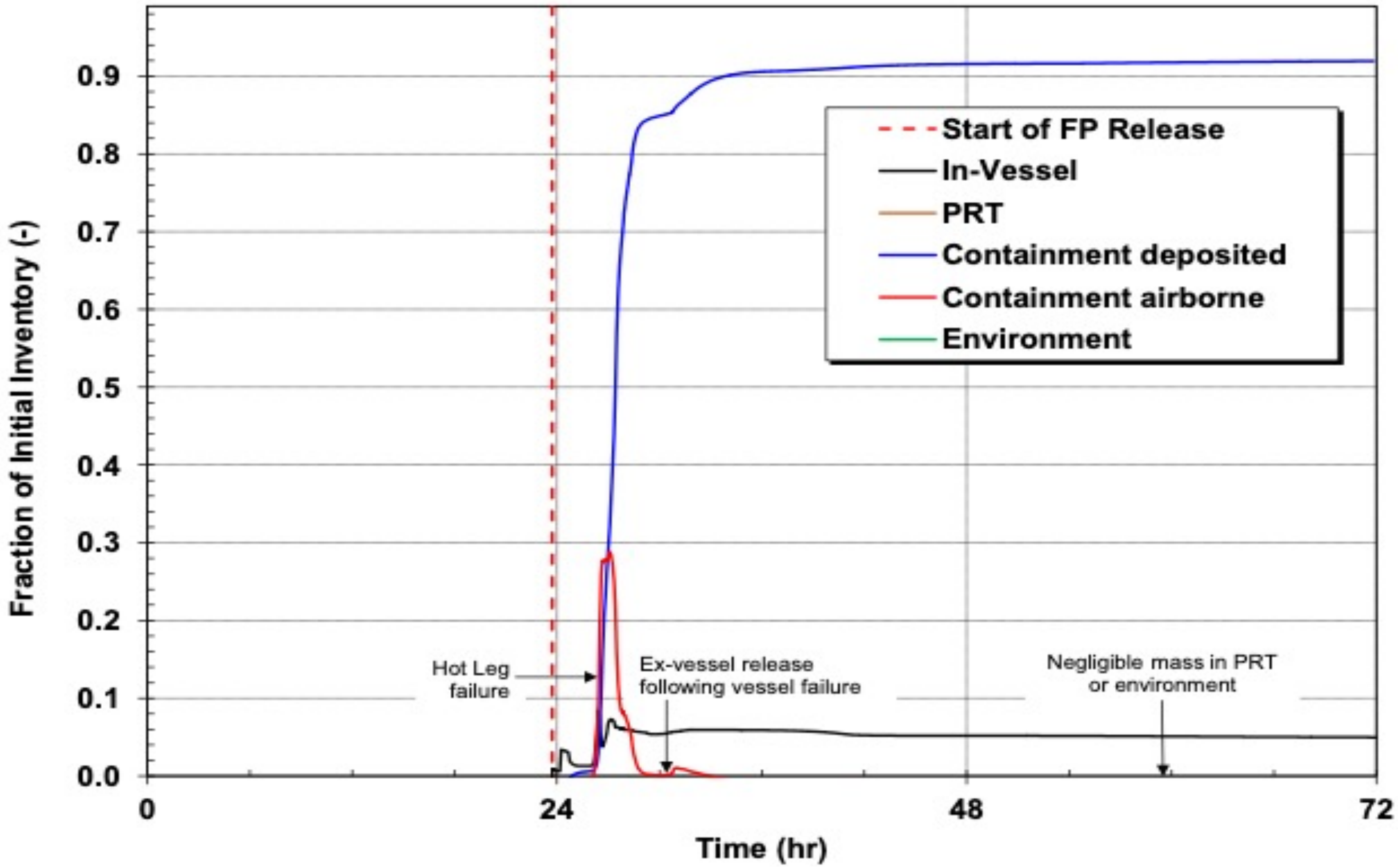
	Gap Release			Early In-vessel		
Study	**2023**	**2011**	**NUREG-1465**	**2023**	**2011**	**NUREG-1465**
Phase Duration	1.3	0.22	0.50	4.0	4.5	1.3
Noble Gases	0.026	0.017	0.050	0.93	0.94	0.95
Halogens	0.007	0.004	0.050	0.58	0.37	0.35
Alkali Metals	0.003	0.003	0.050	0.50	0.23	0.25
Te Group	0.006	0.004	0.0	0.55	0.30	0.050
Ba/Sr Group	0.001	0.0006	0.0	0.002	0.004	0.020
Ru Group	<1.0e-6	0.0	0.0	0.008	0.006	0.003
Mo Group	2.0E-05	0.0	0.0	0.15	0.080	0.003
Lanthanides	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0002
Ce Group	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0005

- SAND2023-01313 and SAND2011-0128 utilized MELCOR
- Accident scenarios and modeling best-practices lead to tendency for increased early in-vessel halogen releases
- Surry and Sequoyah modeling best-practices in SAND2023-01313 represent improvements due to SOARCA

PWR In-Containment Source Terms Consistent with SOARCA

- SOARCA found limited halogen in-vessel retention after hot leg creep rupture

SQN SOARCA halogen releases (LTSBO)



*In-containment source terms reported in SAND2023-01313 characterize **total** radioactive inventory in containment

In-containment Release Rate Evolution

MELCOR

	BWR			PWR		
	Early In-vessel			Early In-vessel		
Study	**2023**	**2011**	**NUREG-1465**	**2023**	**2011**	**NUREG-1465**
Noble Gases	0.14	0.12	0.63	0.21	0.21	0.73
Halogens	0.11	0.059	0.17	0.16	0.082	0.27
Alkali Metals	0.047	0.016	0.13	0.15	0.051	0.19
Te Group	0.091	0.049	0.033	0.15	0.067	0.038
Ba/Sr Group	0.0009	0.0006	0.013	0.0007	0.0009	0.015
Ru Group	0.0009	0.0003	0.002	0.002	0.001	0.002
Mo Group	0.017	0.003	0.002	0.045	0.018	0.002
Lanthanides	<1.0e-6	<1.0e-6	0.0001	<1.0e-6	<1.0e-6	0.0002
Ce Group	<1.0e-6	<1.0e-6	0.0003	<1.0e-6	<1.0e-6	0.0004

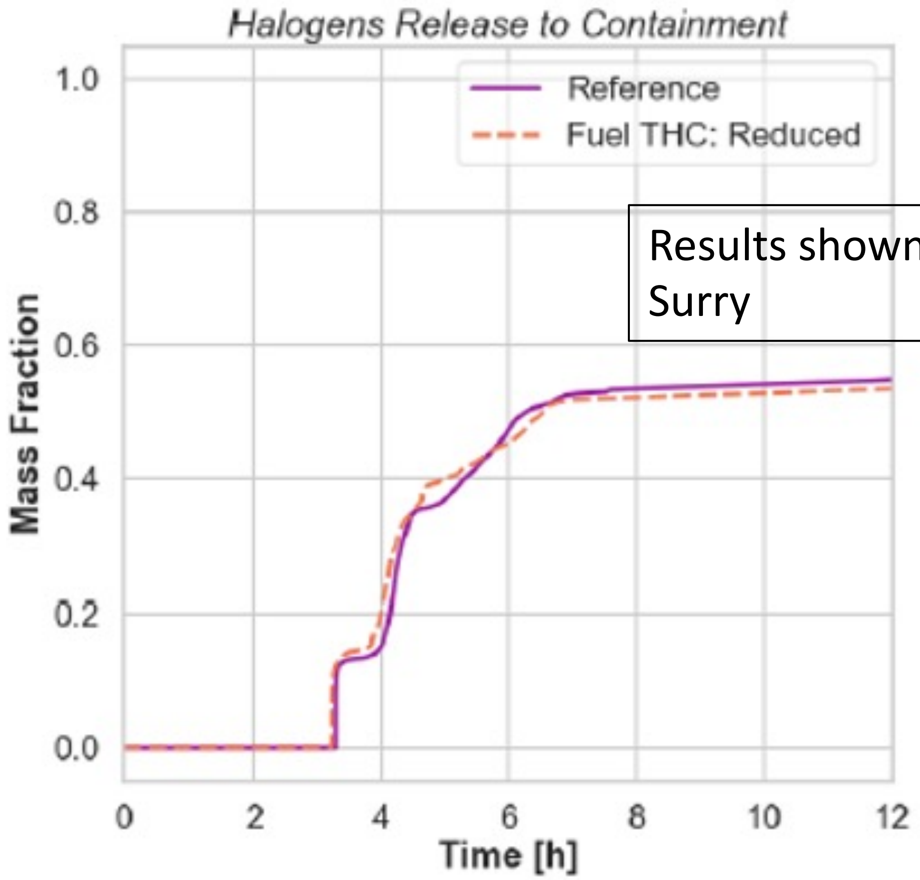
Reported as [release fraction/hour]

Fuel Thermal Conductivity Sensitivity

MELCOR

Increased burnup leads to decrease of fuel thermal conductivity

Sensitivity Case	Fuel Thermal Conductivity [W/m-K]
Reference	4.92
Reduced	2.02
Low	0.2



Results shown for Surry

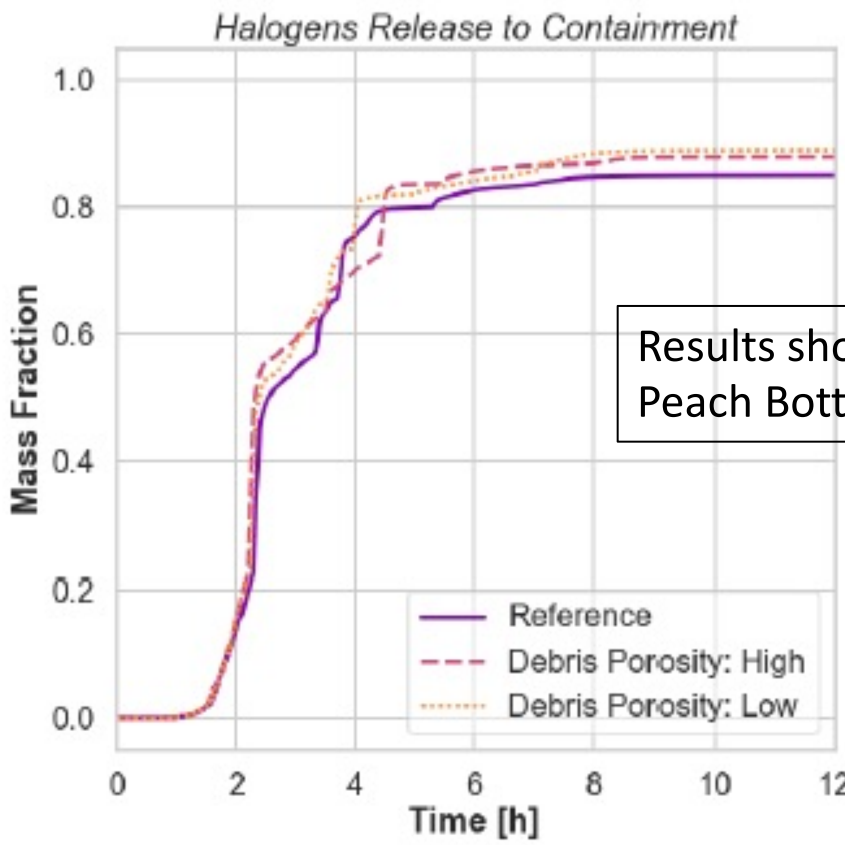
No impact from variation of fuel thermal conductivity

In-vessel Particulate Debris Porosity

Very high burnups have been postulated to promote disintegration of the fuel material

Three sensitivity cases to assess impact on in-containment source term

Sensitivity Case	In-Vessel Particulate Debris Porosity
Reference	0.4
High	0.6
Low	0.2



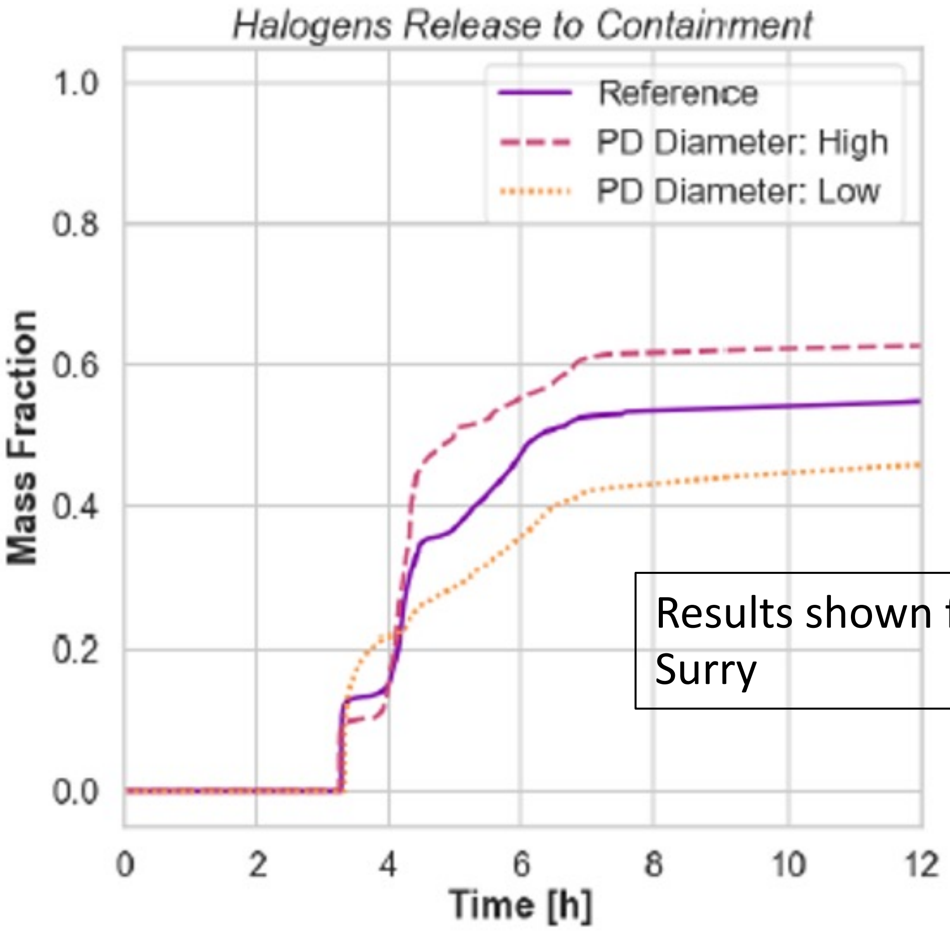
Results shown for Peach Bottom

No impact from variation of in-vessel particulate debris porosity

Diameter of In-vessel Particulate Debris Sensitivity

Higher burnups result in a greater degree of fuel breakup

Sensitivity	In-core Particulate Debris Diameter [cm]	Lower Plenum Particulate Debris Diameter [cm]
Reference	1.0	0.2
High	1.5	0.5
Low	0.5	0.1



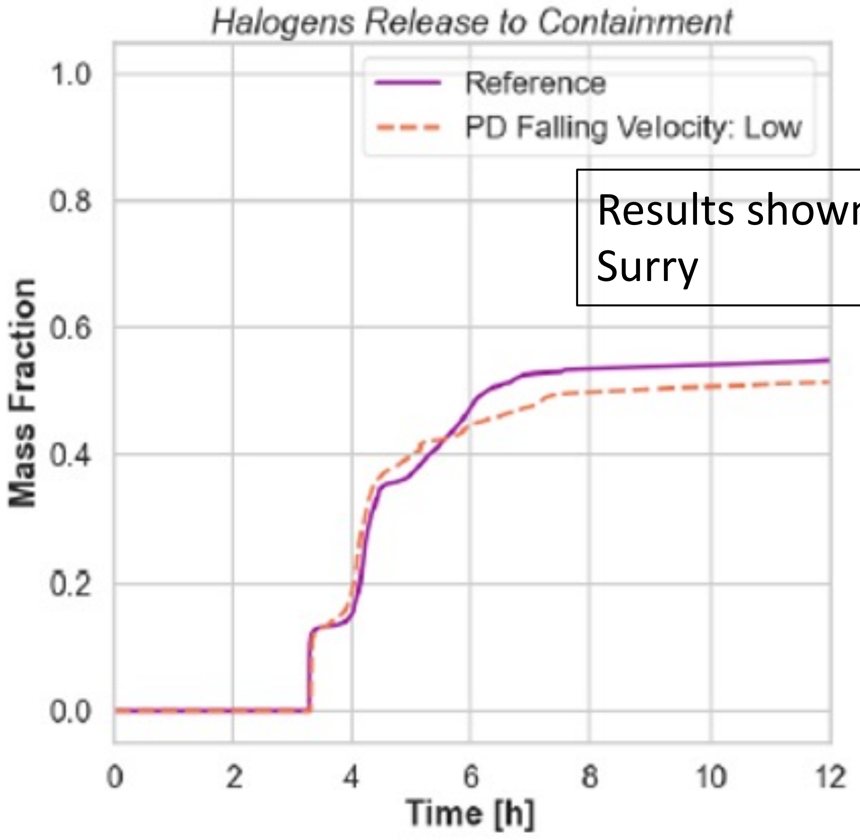
Results shown for Surry

Variation in particulate debris diameter impacts in-containment source term
Impact smaller than changes across accident scenarios

Particulate Debris Falling Velocity Sensitivity

Particulate debris sizes could impact particulate debris fall velocity into lower plenum

Sensitivity	In-Vessel Particulate Debris Fall Velocity [m/s] *Peach Bottom*	In-Vessel Particulate Debris Fall Velocity [m/s] *Surry*
Reference	0.94	0.094
Low	0.094	0.064



Results shown for Surry

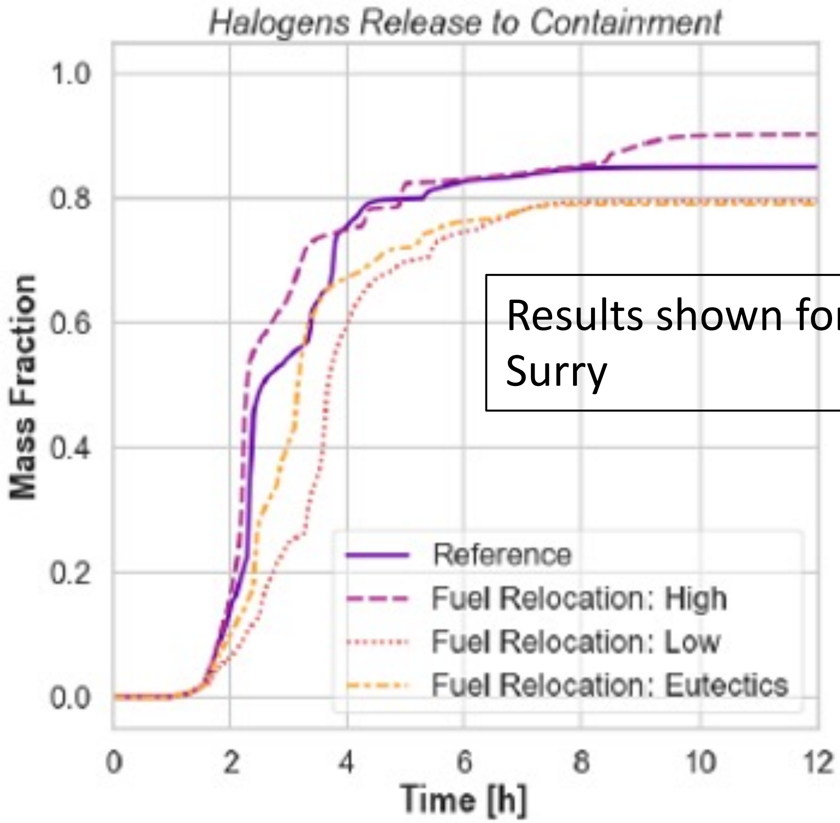
No impact on source term due to variation in particulate debris fall velocity

Fuel Relocation Temperature Sensitivity

Material interactions can cause early failure of fuel assemblies and other core components

- MELCOR uses either the interactive materials model or eutectics model to represent material interactions

Sensitivity	Fuel Relocation Temperature [K]
Reference	2479
High	2728
Low	2230
Eutectics	Eutectics model



Results shown for Surry

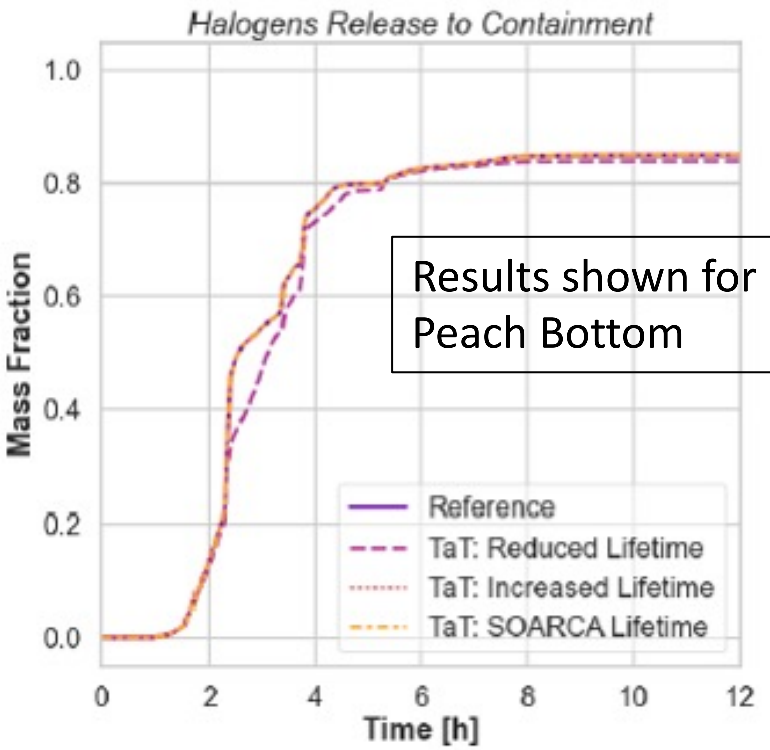
Material interactions that cause early fuel failure and can impact accident progression timings and in-containment source terms based on SOARCA uncertainty studies

Fuel Rod Lifetime Sensitivity

Fuel assemblies at high temperatures exhibit early failures

- Early failures captured in MELCOR simulations using a lifetime function

Sensitivity	Fuel Rod Lifetime Model
Reference	Default time-at-temperature model
Increased Lifetime	Lifetime function that accrues damage from 22.2 hours to 20 minutes at temperatures from 2100K – 2600K
Reduced Lifetime	Lifetime function that accrues damage from 1.67 hours to 3.3 minutes at temperatures from 2100K – 2600K
SOARCA Lifetime	Lifetime function that accrues damage from 10 hours to 5 minutes at temperatures from 2100K – 2600K



Results shown for Peach Bottom

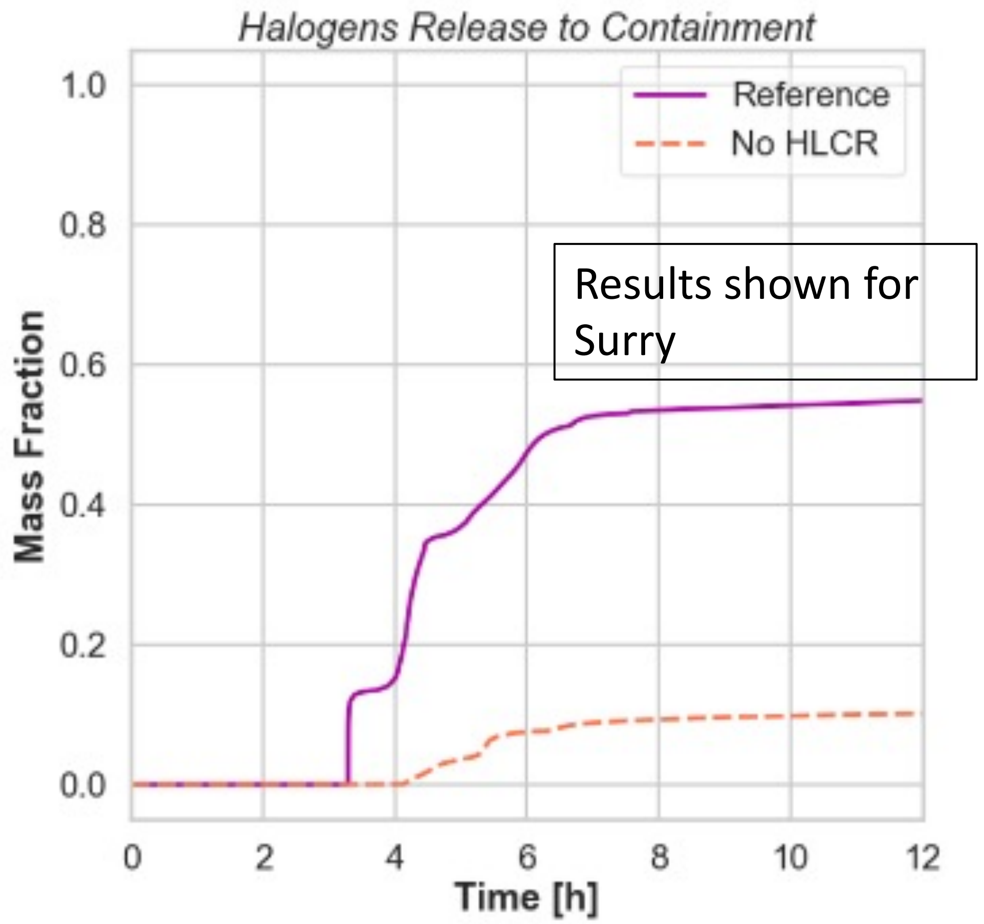
No impact due to variation of the fuel rod lifetime modeling on source term
Oxidized fuel assembly temperature failure model generally dominates

Hot Leg Creep Rupture Sensitivity

Key insight from SOARCA is potential for induced RPV pressure boundary failures

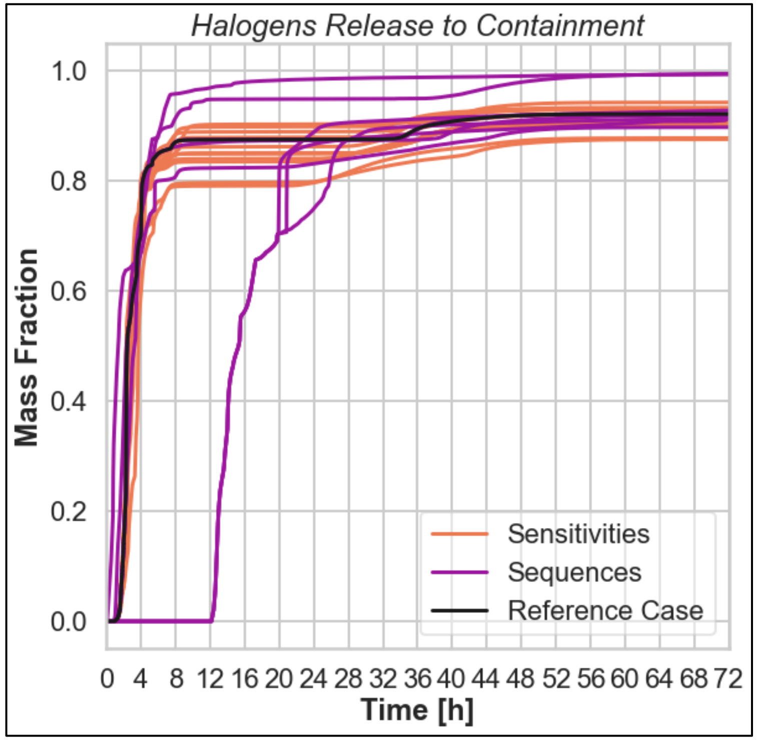
- Severe accident conditions lead to high pressure and temperature conditions at RPV boundary
- Thermally-induced hot leg creep rupture found likely for PWRs
- BWRs exhibited thermally-induced seizure of cycling SRVs

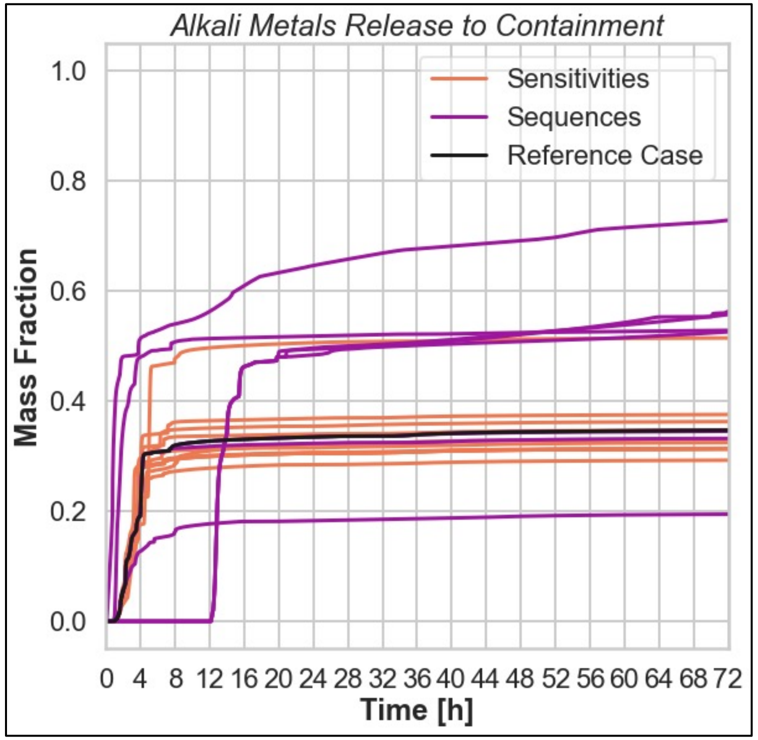
Sensitivity	RPV Induced Pressure Boundary Failure Modeling
Reference	Hot leg creep rupture enabled
No HLCR	Hot leg creep rupture disabled



Significant increase in early in-vessel source term for induced RPV failure for SBOs

In-containment Source Term Variability



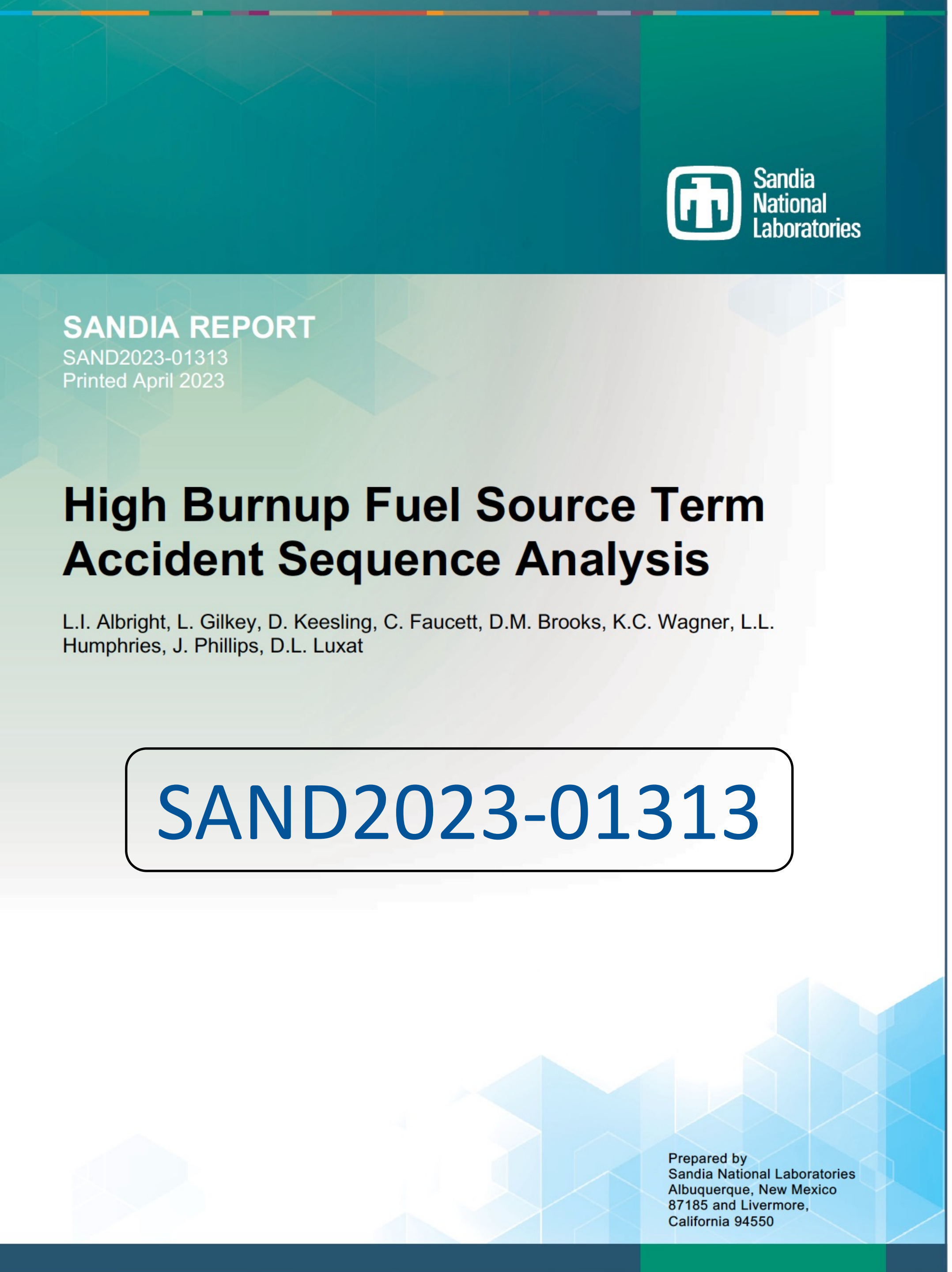


Peer Review Finding

- Potential for combined effects of various sensitivity studies to be larger than separate effects

Nonlinear processes in severe accidents tend to limit amplification of response variability in multi-parameter sensitivity studies such that single scenario variability is less than variation across scenarios

In-containment source term variation dominated by variation across sequences


Sandia National Laboratories

SANDIA REPORT
SAND2023-01313
Printed April 2023

High Burnup Fuel Source Term Accident Sequence Analysis

L.I. Albright, L. Gilkey, D. Keesling, C. Faucett, D.M. Brooks, K.C. Wagner, L.L. Humphries, J. Phillips, D.L. Luxat

SAND2023-01313

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico
87185 and Livermore,
California 94550

Summary

- Increased burnup or extended enrichment does not significantly impact source term
  - Most significant variation in source term arises due to differences between accident scenarios
- Status of RPV has significant impact on early in-vessel releases
  - Low pressure scenarios exhibit more significant releases to containment
  - NUREG-1465 prescribed larger number of high pressure scenarios than SAND2023-01313
- Early in-vessel source term greatly reduced if RPV pressure boundary intact

Independent Peer Review

Focus of SAND2023-01313 Peer Review

MELCOR

ERI/NRC 23-201

PEER REVIEW OF THE IN-CONTAINMENT SOURCE TERM STUDY FOR HIGH-BURNUP AND HIGH-ASSAY LOW ENRICHED URANIUM FUELS

ERI/NRC 23-201

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

ENERGYRESEARCHINC
P. O. Box 2034
Rockville, Maryland 20847

April 2023

- Review technical basis of SAND2023-01313
- Recommend improvements to SAND2023-01313
- Assess suitability of SAND2023-01313 source terms for regulatory applications

Peer Review Organization

Panel Membership

- Dr. Mohsen Khatib-Rahbar – Panel Chair
  - Energy Research, Inc. (ERI)
- Dr. Richard S. Denning
  - Consultant
- Mr. Jeff Gabor
  - Jensen Hughes
- Dr. Didier Jacquemain
  - Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA)
- Dr. Luis E. Herranz
  - Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT)
- Dr. Yu Maruyama
  - Japan Atomic Energy Agency (JAEA)

Panel Objectives

- Assess technical adequacy with respect to:
  - Overall analysis approach
  - Specific applications of MELCOR to development of in-containment source terms
- Assess appropriateness of severe accident sequences selected
- Assess applied models and assumptions in terms of
  - Current understanding of severe accidents and source terms
  - Adequacy considering available experimental data, and observations
- Assess that source terms are representative, rather than conservative or bounding
- Assess adequacy of documentation against
  - Completeness of technical bases specification
  - Approach to analysis of uncertainties

Peer Review Process

MELCOR

- Draft High Burnup Fuel Source Term Accident Sequence Analysis (Completed 2021)
- Virtual Meetings (began in 2022)
  1. Briefing on the peer review objectives and the draft report by NRC and SNL
     - Panelist review reports delivered to SNL
     - Preliminary resolution of comments by SNL
     - Preparation of the draft peer review report
  2. Discussion of draft peer review report, comment resolution, and summary of unresolved comments
     - Final resolution of comments by SNL
     - Revision of High Burnup Fuel Source Term Accident Sequence Analysis report
  3. Discussion of revised report, peer review panel findings, and conclusions
     - Final High Burnup Fuel Source Term Accident Sequence Analysis report released (2023)
     - Final peer review report released (2023)

Acceptability of the SAND2023-01313 Source Term

- “[The peer review panel] **endorses the approach** taken in [SAND2023-01313]”
- “[SAND2023-01313] provides a **defendable technical basis** for the proposed source terms”
- “The peer review panel finds that the four nuclear power plants considered in the [SAND2023-01313] **reasonably represent the U.S. nuclear fleet**”
- “The **spectrum of accidents is sufficient** to satisfy the following stated attributes of an acceptable alternative accident source term (RG 1.183):

  The accident source term must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

Qualities of the SAND2023-01313 Source Term

- Study is a significant technical improvement using state-of-the-art methods implemented in latest version of MELCOR
- In-containment source terms for HBU/HALEU fuels are representative MELCOR estimates, rather than conservative or bounding estimates
- No bias in the approach identified that could overestimate in-containment source terms
- Sensitivity studies documented in SAND2023-01313 valuable in supporting applications
    - Sensitivities explored limitations in understanding of HBU/HALEU fuel response under severe accident conditions
    - Results demonstrated impact of thermally induced (creep) depressurization of RCS for PWRs on in-containment source terms

Peer Review Report Recommendations

- Gap release phase incorporated into the early in-vessel phase
  - *The panel considers the current approach of separating the gap and early in-vessel release phases, a product of the simplified single channel treatment of the STCP models of circa 1980s that is reflected in the NUREG-1465 source terms, outdated. During severe accidents, the gap and in-vessel releases from the fuel overlap to the extent that it is not possible to truly separate the two as distinct phases. Therefore, it is recommended that the gap release be incorporated into the early in-vessel release phase.*
- More appropriate to represent impact of burnup using core inventories for HBU expressed in terms of radiological activities
  - *The implication of comparison of mass inventories in kilogram [SAND2023-01313] is to incorrectly conclude that at higher fuel burnups, off-site doses would likely be substantially higher for high burnup fuels as the direct result of larger core mass inventories of radionuclides. In fact, when compared on the basis of integrated radiological activity, there would not be any significant differences for the two levels of fuel burnup.*
  - Examples shown in the next presentation: “Follow-on Calculations”

Other Comments And Recommendations

- Panelists requested additional clarification (reflected in final report) that
  - Containment bypass scenarios and air ingression not considered in development of tabular source terms
  - Fission product removal mechanisms in containment not included in tabular source terms
    - Captured in MELCOR simulations, but post-processed out of reported MELCOR source terms
- Peer reviewers acknowledged more recent PRA studies could have different contributors to core damage
  - For the intended applications the scenarios used in the current [SAND2023-01313] appropriate with regards to the progression of severe accidents, radionuclide release and transport
- Panelists noted for most radionuclides no increase in activity with burnup sufficient to impact siting calculations
- Peer reviewers noted the uncertainty in Iodine speciation based on experiments (FPT3, DF-4, and BECARRE)
- Peer review noted that current Fukushima Daiichi post-accident analyses confirm the assumption that Cs_2MoO_4 is dominant chemical form of Cs
- Peer review panel considered the use of median estimates appropriate to avoid bias due to potential outliers

Other Comments And Recommendations

Finally, even though tabular severe accident in-containment source terms provide a simplified tool for regulatory applications and analyses, it is important to recognize their limitations and the panel encourages the direct application of a state-of-the art severe accident code to specific issues when appropriate.

Fission Product Retention in Suppression Pools

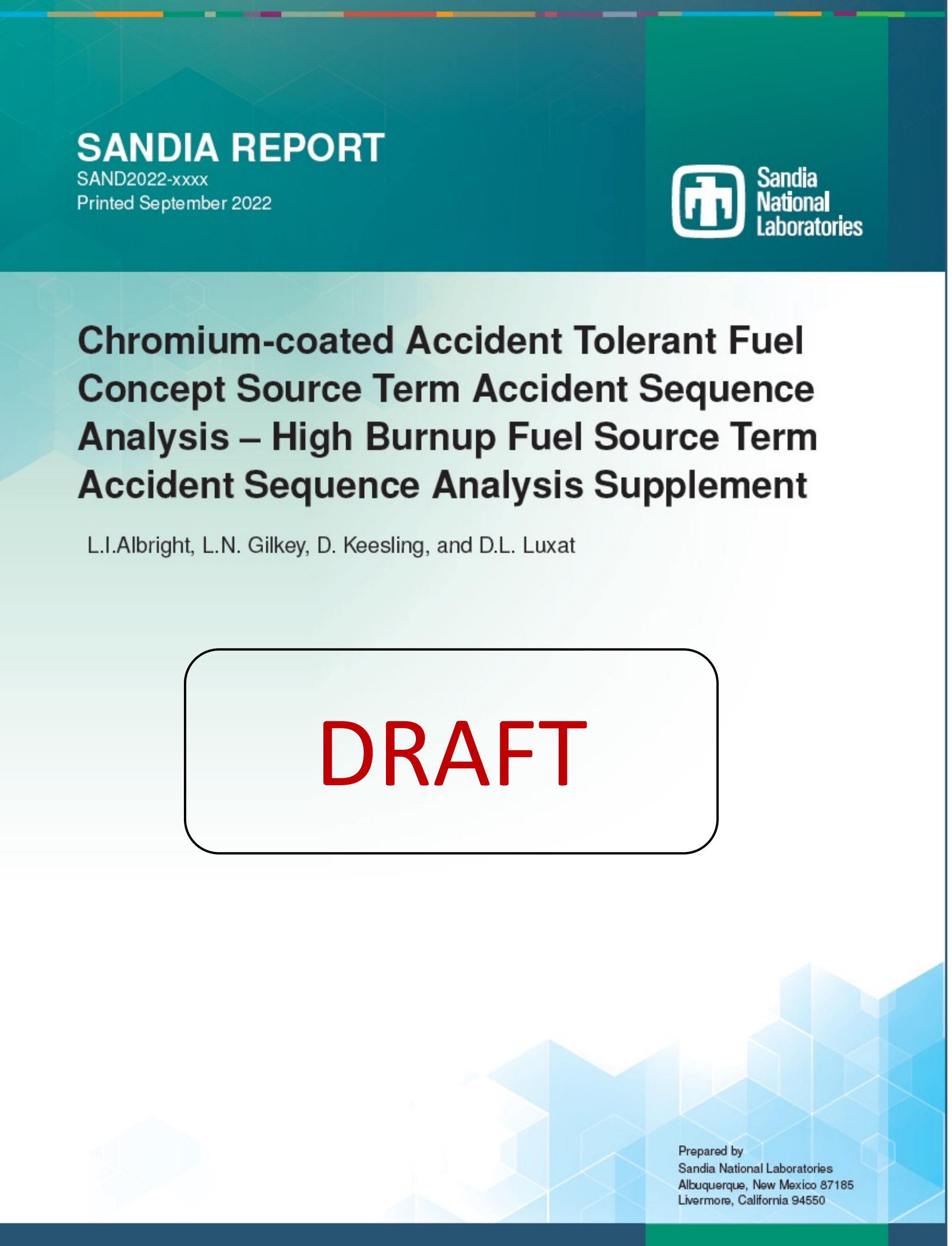
	Gap Release		Early In-vessel	
Release Category	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory
Noble Gases	0.016	0.016	0.95	0.95
Halogens	0.005	1.30E-06	0.71	0.06
Alkali Metals	0.005	1.20E-06	0.32	0.006
Te Group	0.003	<1.0e-6	0.56	0.038
Ba/Sr Group	0.0006	<1.0e-6	0.005	0.0003
Ru Group	<1.0e-6	<1.0e-6	0.006	7.40E-06
Mo Group	1.90E-05	<1.0e-6	0.12	0.0001
Lanthanides	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6
Ce Group	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6

Peer Review Findings

- In-containment source terms should consider the impact of retention in suppression pools, especially for SBO scenarios that discharge directly into the suppression pool
- Estimates of retention in suppression pools provided in SAND2023-01313 could be used in regulatory guidance to establish suppression pool decontamination factors

Significant effect of retention in suppression pool on key radionuclide groups

Upcoming Work



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Sandia National Laboratories

Chromium-coated Accident Tolerant Fuel Concept Source Term Accident Sequence Analysis – High Burnup Fuel Source Term Accident Sequence Analysis Supplement

L.I.Albright, L.N. Gilkey, D. Keesling, and D.L. Luxat

DRAFT

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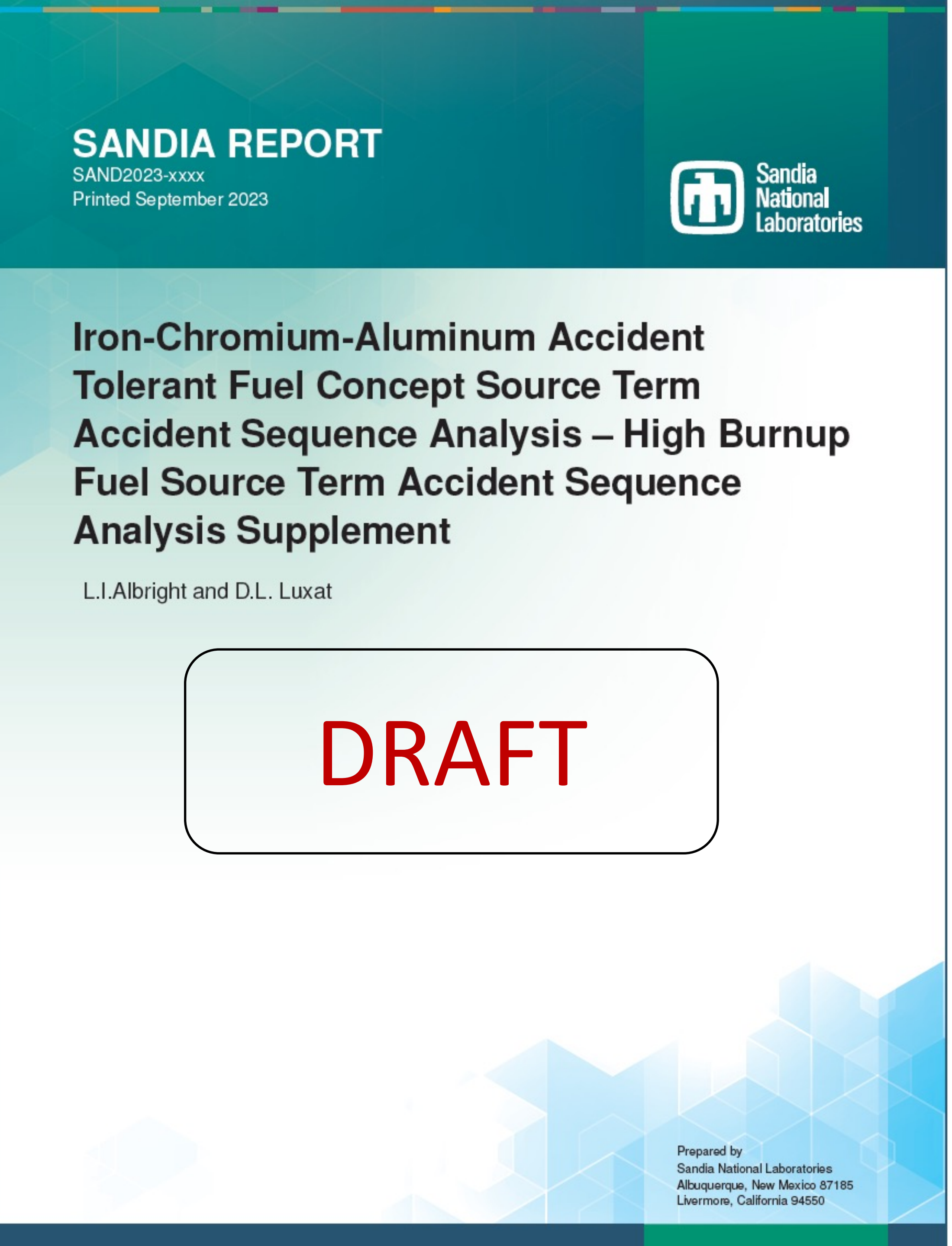
Cr-Coated ATF Concept

- Cr-coated ATF concept most similar to conventional fuels
  - Thin, protective chromium coating on Zircaloy fuel cladding delays exothermic Zircaloy oxidation onset
- Cr-coated analysis informed by ATF severe accident PIRT (NUREG/CR-7283) findings

MELCOR


SANDIA REPORT
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Sandia National Laboratories

Iron-Chromium-Aluminum Accident Tolerant Fuel Concept Source Term Accident Sequence Analysis – High Burnup Fuel Source Term Accident Sequence Analysis Supplement

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FeCrAl ATF Concept

- FeCrAl ATF concept utilizes a novel fuel cladding material
  - Substitution of Zr-based alloy with an FeCrAl alloy
  - Intended to reduce both oxidation in the core and associated hydrogen production
- FeCrAl analysis informed by ATF severe accident PIRT (NUREG/CR-7283) findings
  - Sensitivity analyses deployed to interrogate FeCrAl cladding knowledge uncertainties

Thank you for your attention!

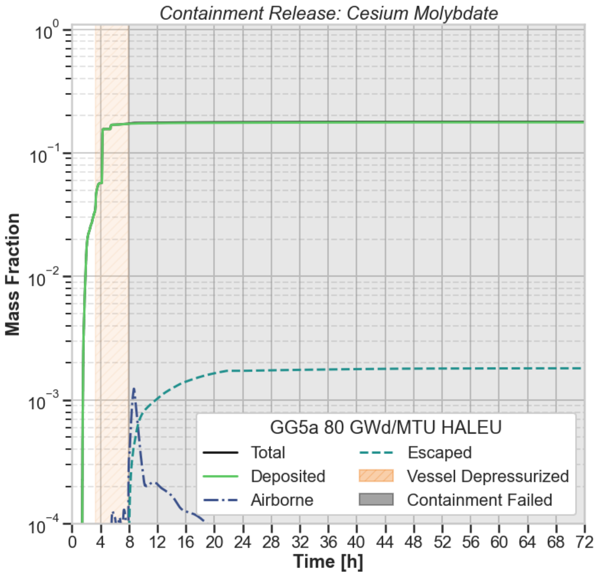
Backup Slides

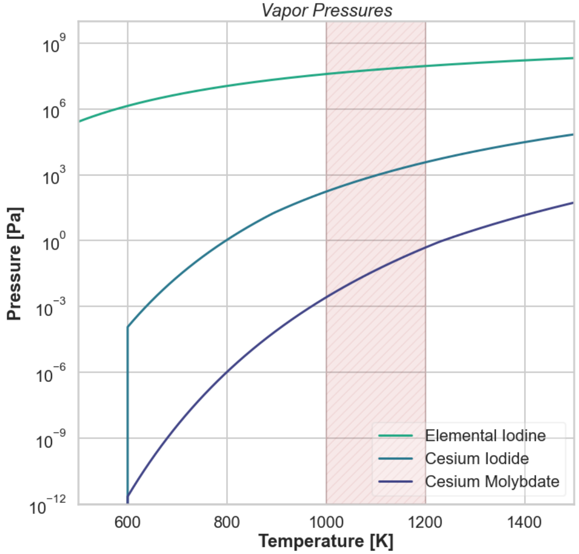
Acronyms

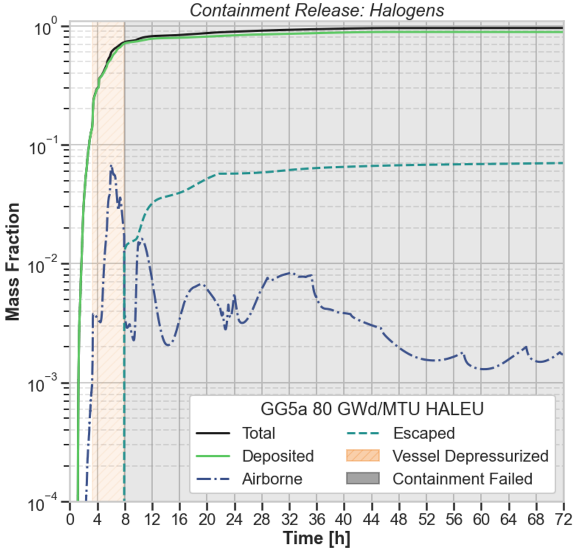
Acronym	Definition	Acronym	Definition
AC	Alternating current	NRC	Nuclear Regulatory Commission
ADS	Automatic depressurization system	ORNL	Oak Ridge National Laboratories
AFW	Auxiliary Feedwater	PB	Peach Bottom
AST	Alternative source term	PIRT	Phenomena Identification and Ranking Table
ATF	Accident tolerant fuel	PORV	Pilot-operated relief valve
ATWS	Anticipated transient without scram	PRA	Probabilistic risk assessment
BWR	Boiling water reactor	PRT	Pressurizer relief tank
CCFL	Counter current flow	PWR	Pressurized water reactor
CDF	Core damage frequency	QoI	Quantity of interest
DC	Direct current	RCIC	Reactor core isolation cooling system
ECCS	Emergency core cooling system	RCP	reactor coolant pump
ECDF	Empirical cumulative distribution function	RCS	Reactor coolant system
GG	Grand Gulf	RHR	Residual heat removal
HALEU	High-assay low-enriched uranium	SBLOCA	Small-break loss of coolant accident
HBU	High burnup	SBO	Station blackout
HLCR	Hot leg creep rupture	SOARCA	State-of-the-Art Reactor Consequence Analyses
HPCI	High pressure coolant injection system	SQN	Sequoyah
HPSI	High-pressure safety injection	SRV	Safety relief valve
LBLOCA	Large-break loss of coolant accident	STCP	Source Term Code Package
LEU	Low-enriched uranium	STSBO	Short-term station blackout
LOCA	Loss of coolant accident	SU	Surry
LPCI	Low-pressure coolant injection	TDAFW	Turbine-driven auxiliary feedwater
LPSI	Low-pressure safety injection	TID	Technical information document
LTSBO	Long-term station blackout	TMI-2	Three Mile Island Unit-2
LWR	Light water reactor		

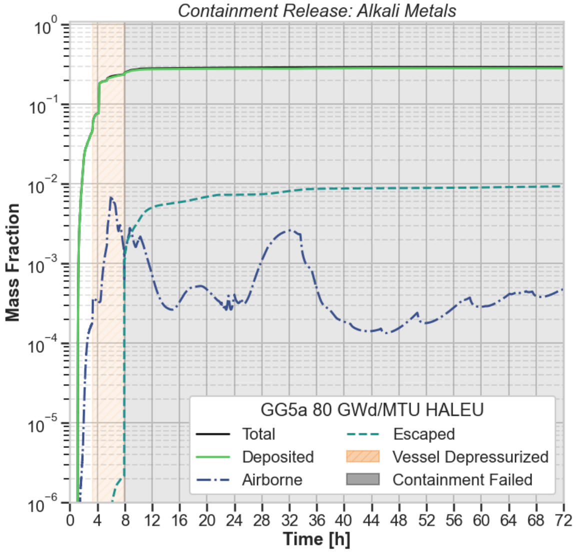
Cs and I Releases

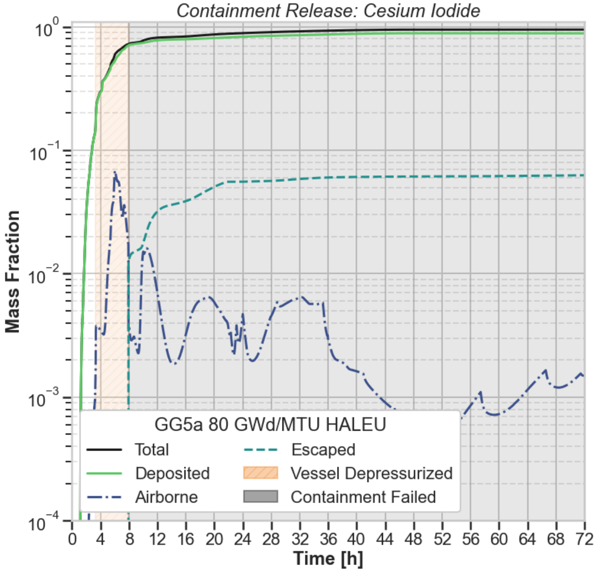
- SAND2011-0128 considered deposition of radionuclides on the lower head, leading to significantly decreased in-vessel phase releases.
  - This consideration delays a significant fraction of radionuclide release to containment until after lower head failure during the ex-vessel phase (employed for Peach Bottom and Sequoyah)
  - This practice is no longer considered appropriate, and was not employed in SAND2023-01313
- CsI (all original I inventory and ~10% original Cs inventory) transports readily from the primary system to containment during core damage due to the relatively large CsI vapor pressures at elevated primary system temperatures
  - Consistent with Peach Bottom SOARCA results











NUREG-1465 Accident Selection

- Dominant sequences were chosen based on impact on source term
  - PWRs are predominantly LOCA accidents
  - BWRs are predominantly SBO/ATWS accidents

PWR Plants	Sequence	Description
Surry	AG	LOCA (hot leg), no containment hear removal systems
	TMLB	LOOP, no PCS and no AFWS
	V	Intefacing system LOCA
	S3B	SBO with RCP seal LOCA
	S2D-δ	SBLOCA, no ECCS and H2 combustion
	S2D-β	SBLOCA w/ 6" hole in containment
Oconee 3	TMLB	SBO, no active ESF systems
	S1DCF	LOCA (3"), no ESF systems
Sequoyah	S3HF1	LOCA RCP, no ECCS, no CSRS w/ reactor cavity flooded
	S3HF2	S3HF1 w/ hot leg induced LOCA
	3HF2	S3HF1 w/ dry reactor cavity
	S3B	LOCA (1/2") w/ SBO
	TBA	SBO induces hot leg LOCA - H2 burn fails containment
	ACD	LOCA (hot leg), no ECCS no CS
	S3B1	SBO delayed 4 RCP seal failures, only steam driven AFW operates
	S3HF	LOCA (RCP seal), no ECCS no CSRS
	S3H	LOCA (RCP seal) no ECCS recirculation

BWR Plants	Sequence	Description
Peach Bottom	TC1	ATWS w/ reactor depressurized
	TC2	ATWS w/ reactor pressurized
	TC3	TC2 with wetwell venting
	TB1	SBO with battery depletion
	TB2	TB1 with containment failure at vessel failure
	S2E1	LOCA (2"), no ECCS and no ADS
	S2E2	S2E1 with basaltic concrete
	V	RHR pipe failure outside containment
	TBUX	SBO with loss of all DC power
LaSalle	TB	SBO with late containment failure
Grand Gulf	TC	ATWS early containment failure fails ECCS
	TB1	SBO with battery depletion
	TB2	TB1 w/ H2 burn fails containment
	TBS	SBO, no ECCS but reactor depressurized
	TBR	TBS with AC recovery after vessel failure