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MEMORANDUM TO: Robert G. Lukes, Chief
Nuclear Methods and Fuel Analysis Branch
Division of Safety Systems
Office of Nuclear Reactor Regulation

FROM: Paul M. Clifford, Senior Technical Advisor for Reactor Fuel ***/RA/***
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SUBJECT: TECHNICAL BASIS FOR DRAFT RG 1.183 REVISION 1 (2021)
NON-LOCA FISSION PRODUCT RELEASE FRACTIONS

The purpose of this memorandum is to document the technical basis for recommended updates to existing guidance related to fission product release fractions used to assess non-loss-of-coolant accident (non-LOCA) radiological consequences. Table 3 of Regulatory Guide (RG) 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, provides steady-state radionuclide release fractions residing in the fuel rod void volume (i.e., plenum plus pellet-to-cladding gaps) and available for release upon cladding breach. Currently, Table 3 is limited in applicability to a maximum rod average burnup of 62 GWd/MTU and a peak rod average power of 6.3 kw/ft above 54 GWd/MTU rod average burnup. To support modern fuel design and utilization, the staff is extending the applicability up to a rod average burnup of 68 GWd/MTU along with an expanded operating regime. The NRC's fuel rod thermal-mechanical Fuel Analysis Under Steady-State and Transient model (FAST-1.0), along with the ANS-5.4 Standard Fission Product Release Model (2011), was used to predict high-confidence steady-state radionuclide release fractions.

In addition to the steady-state release inventory, additional fission products may be released from the fuel pellets due to local transient conditions. Grain boundary separation and pellet fragmentation promote a burst release of fission products which is being referred to as transient fission gas release (TFGR). This memorandum also includes recommended updates associated with TFGR.

In addition, this memorandum includes several attachments which support the development of the revised guidance, provide recommendations on past guidance, and provide input for an environmental assessment of high burnup fuel.

Attachment 1: Alternative Treatment for Cesium Release Fraction
Attachment 2: Analytical Technique for Calculating Steady-State Release Fractions
Attachment 3: Sample Calculation of Steady-State Release Fractions
Attachment 4: Reconstruction of Technical Bases and Recommendations for RG 1.183 Revision 00 Non-LOCA Release Fractions

Attachment 5: Best-Estimate Release Fractions for Environmental Assessment

Steady-State Radionuclide Release Fractions

PNNL Report PNNL-18212, Revision 1, *Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard* (ML111890397), documents an analytical technique for calculating radionuclide release fractions using an approved fuel rod thermal-mechanical model along with the ANS-5.4 standard fission product release model (NUREG/CR-7003). Attachment 2 of this memorandum captures the analytical procedure developed in PNNL-18212, Revision 1. Following this analytical procedure, steady-state release fractions were calculated using the latest version of the FAST fuel rod thermal-mechanical model (FAST-1.0.1). As described below, the only deviation to the original procedure was the treatment of cesium (Cs)-134 and Cs-137.

Inputs and Assumptions:

Steady-state radionuclide release fractions are sensitive to fuel rod design parameters and power history. To ensure a broad range of applicability, the following inputs and assumptions were used to generate the release fractions.

1. For the pressurized-water reactor (PWR) fleet, the Combustion Engineering 14x14 fuel rod design is more limiting than remaining Babcock & Wilcox, Combustion Engineering, and Westinghouse 14x14, 15x15, 16x16, and 17x17 fuel rod configurations.
2. For the boiling-water reactor (BWR) fleet, the General Electric 10x10 fuel rod design is more limiting than the remaining 10x10 and 11x11 fuel rod configurations. Note that the legacy 8x8 and 9x9 bundle designs are no longer in operation.
3. Figure 1 shows PWR and BWR peak fuel rod average powers versus burnup. Figure 2 shows PWR and BWR fuel rod axial power distributions. The product of these rod power components, peak linear heat generation rate (PLHGR), is 14.0 KW/ft and 15.0 KW/ft for PWR and BWR rod designs, respectively. The combination of a larger radial power profile and relatively flat axial power profile tends to maximize releases along the entire fuel column.
4. Cesium releases are not calculated directly in most fuel rod thermal-mechanical models. Instead, cesium releases are inferred based upon predictions of long-lived noble gas releases and established relationships between cesium and noble releases. Attachment 1 provides an alternative treatment for predicting the fraction of cesium residing in the fuel rod void volume and available for release upon cladding breach.
5. According to the FAST-1.0 Integral Assessment (PNNL-29727), the standard deviation for the steady-state UO_2 FGR prediction is 2.6% absolute up to 70 GWd/MTU rod average. Following the uncertainty treatment described in PNNL Report PNNL-18212, a 6.14% absolute uncertainty term will be added to the predicted, best-estimate Kr-85 release fractions to obtain upper bound values.
6. The ANS-5.4 Standard Fission Product Release Model (NUREG/CR-7003) recommends multiplying the best-estimate, short-lived radionuclide release predictions by a factor of 5.0 to obtain upper bound release fractions.

Figure 1. Power Operating Envelope for Steady-State Release Fractions

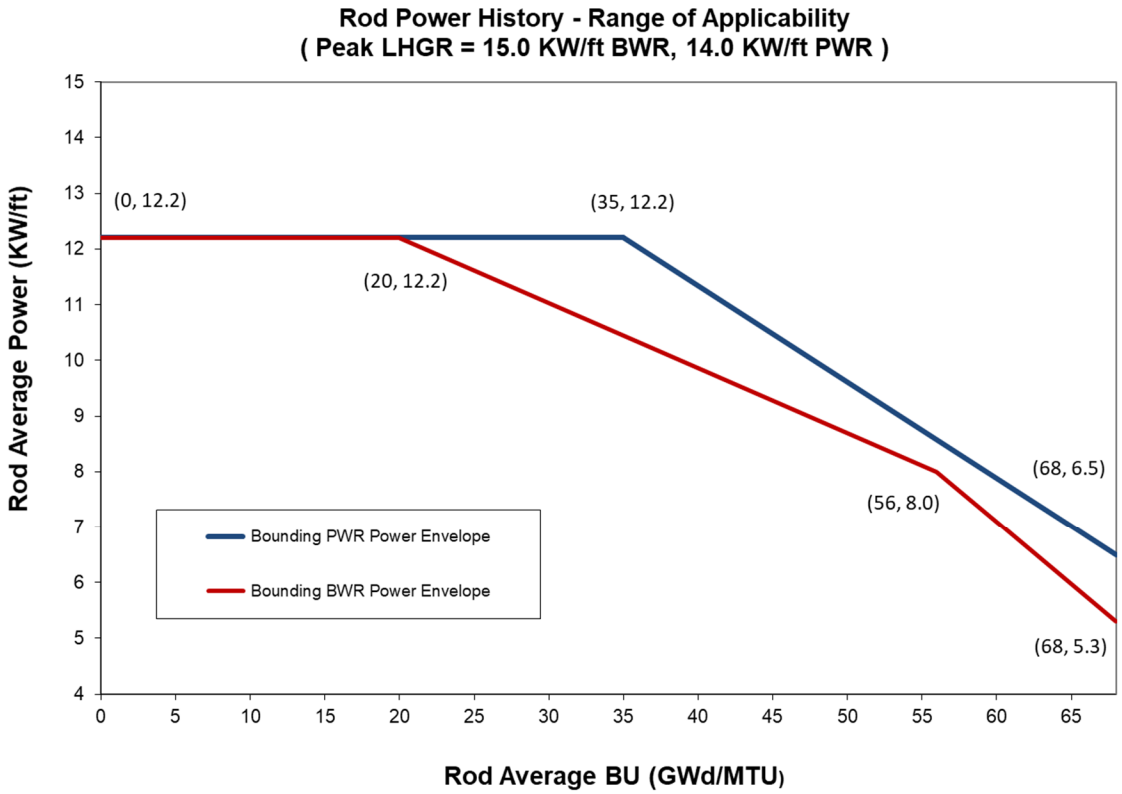
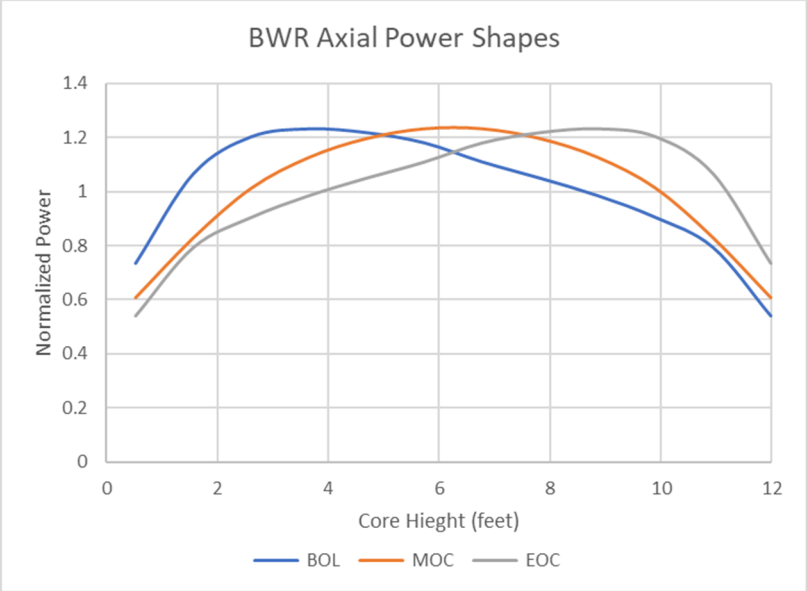
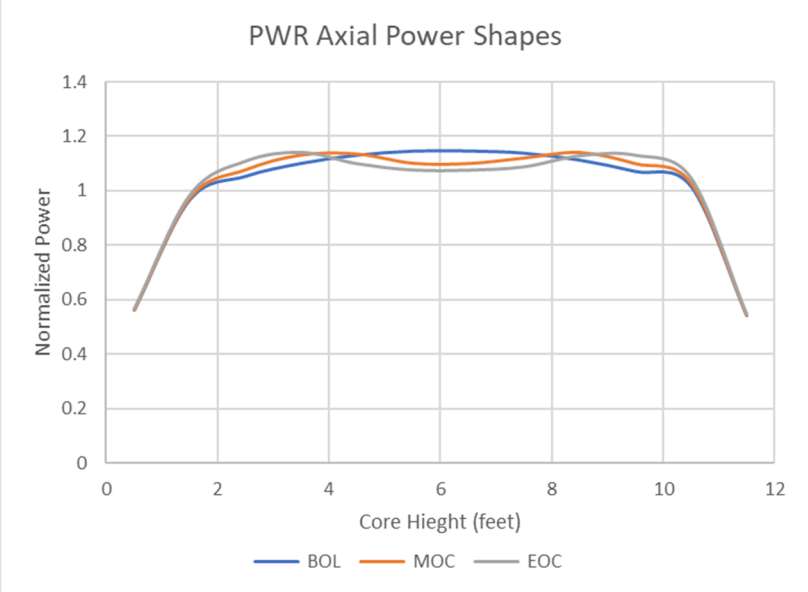


Figure 2. Axial Power Distributions for Steady-State Release Fractions





FAST Calculations:

Following the analytical procedure in Attachment 2 and above inputs and assumptions, FAST-1.0.1 was employed along with ANS-5.4 (2011) standard release models to calculate bounding PWR 14x14 and BWR 10x10 radionuclide release fractions. Steady-state radionuclide releases are sensitive to both fuel design and operating history. As described above, limiting fuel rod designs and power operating histories were selected to expand the applicability of the results to ensure that the regulatory guidance remains useful. While not documented within this memo, earlier sensitivity studies informed these decisions on fuel rod design and power operating history.

Table 1 lists the results from the PWR 14x14 FAST calculations and upper bound release fractions. The PWR power operating envelope and axial power distributions shown in Figures 1 and 2 yielded a maximum rod average burnup of 68.1 GWd/MTU, local burnup of 76.5 GWd/MTU, and irradiation residence time up to 1436 days. Two types of fuel rod power calculations were performed: (1) a single fuel rod operates along the Figure 1 operating envelope and (2) a segmented power history (SPH) where 7 fuel rod power histories comprise the Figure 1 operating envelop. For the SPH cases, each fuel rod operates at the peak power for approximately $1/7^{th}$ of its lifetime. During the remaining SPH lifetime, rods operate at 10% below the peak rod average power profile. The goal of the SPH approach is to remove excess conservatism in the prediction of long-lived stable releases since it is not possible for a single fuel rod to lead the core at all exposures. No benefit is expected for the short-lived radionuclides.

Table 2 provides a comparison of the FAST predictions for the full envelope and SPH radionuclide release fractions. Examination of this table reveals that the Kr-85 release fraction decreased from 0.3994 to 0.3834. This change was less than expected. As anticipated, the short-lived release fractions were essentially unchanged. Hence, the overall benefit of the SPH approach with respect to radiological consequences was not significant. Based on these results, it was decided to base the radionuclide release fraction guidance on the full envelope.

Figure 3 shows trends in predicted release rates for long-lived Kr-85 and short-lived I-131. Examination of this figure reveals that the maximum Kr-85 release fraction occurs at approximately 50 GWd/MTU. Long-lived radionuclide releases into the fuel rod void volume continue to accumulate and the maximum concentration (i.e. moles of gas) will be at end-of-life (EOL). However, the decreasing trend in rod power toward EOL results in lower fuel temperatures and a slower rate of fission gas release (FGR). Long-lived radionuclide release fraction is expressed as the ratio of total release over total production. Given that total production of Kr-85 continues to increase, Figure 3 shows that the predicted release fraction begins to slightly decrease at time-steps near EOL where release rates decrease with reducing fuel temperatures.

As shown in Figure 3, the maximum short-lived I-131 release fraction occurs at 35 GWd/MTU. Given that short-lived radionuclides decay and do not accumulate with power operation, the ANS-5.4 standard release model considers production rate, diffusion coefficient, and half-life. These factors tend to maximize short-lived release fractions and void volume concentrations where fuel rod power (i.e., rate of production) and fuel temperatures (i.e., rate of diffusion) are at maximum values. For a given rod power, fuel temperatures tend to increase with burnup due to a progressively decreasing fuel thermal conductivity (TCD). The effects of TCD is shown in Figure 3 as an increasing trend of predicted release fraction at constant rod power up to 35 GWd/MTU. At this point along the rod power profile, the fuel pellet is at both maximum power (production rate) and maximum temperature (diffusion rate). The maximum predicted release

fraction occurs just prior to a decrease in rod power, with subsequent decrease in fuel temperature. Table 3 provides the recommended PWR steady-state release fractions.

Examination of the PWR full power envelope calculation reveals that the predicted EOL rod internal pressure was 3501 psia. At a differential pressure of 1251 psid, the fuel rod cladding is approaching a lift-off condition. However, predicted fuel temperatures continue to diminish which indicates that any potential feed-back effects associated with a widening fuel-to-cladding gap are insignificant. Note that FGR model uncertainties are applied outside the FAST simulation and therefore do not contribute to predicted rod internal pressure. Given that fuel rod design requirements preclude cladding lift-off, this shows that the recommended release fractions are conservative relative to how PWR fuel is designed and operated and hence appropriate for the purpose of guidance development.

Table 4 lists the upper bound results from the BWR 10x10 FAST calculations. The BWR peak rod average power and axial power distributions shown in Figures 1 and 2 yielded a maximum rod average burnup of 68.1 GWd/MTU, local burnup of 78 GWd/MTU, and irradiation residence time of 1261 days. Figure 4 shows trends in predicted release rates for long-lived Kr-85 and short-lived I-131. As with the PWR trends discussed above, maximum long-lived Kr-85 release fraction occurs at approximately 50 GWd/MTU, maximum long-lived Kr-85 concentration (i.e. moles of gas) will be at EOL, and maximum short-lived I-131 release fractions and fuel rod void inventory will be at 20 GWd/MTU where fuel rod power (rate of production) and fuel temperatures (rate of diffusion) are at maximum values. Table 5 provides the recommended BWR steady-state release fractions.

Examination of the BWR full power envelope calculation reveals that the predicted EOL rod internal pressure was 2682 psia. At a differential pressure of 1627 psid, the fuel rod cladding is approaching a lift-off condition. However, predicted fuel temperatures continue to diminish which indicates that any potential feed-back effects associated with a widening fuel-to-cladding gap are insignificant. Note that FGR model uncertainties are applied outside the FAST simulation and therefore do not contribute to predicted rod internal pressure. Given that fuel rod design requirements preclude cladding lift-off, this shows that the recommended release fractions are conservative relative to how BWR fuel is designed and operated and hence appropriate for the purpose of guidance development.

Table 1: FAST PWR Steady-State Radionuclide Release Fractions

Long-Lived Stables

Case	FAST Peak FGR	FAST Unc. Term	Kr-85 Upper Bound	Cs-134 Upper Bound	Cs-137 Upper Bound	
Full Envelope	0.3380	0.0614	0.3994	0.1997	0.1997	< - Peak Release at 51.7 GWd/MTU
PWR Segment 1	0.2660	0.0614	0.3274	0.1637	0.1637	
PWR Segment 2	0.2670	0.0614	0.3284	0.1642	0.1642	
PWR Segment 3	0.2690	0.0614	0.3304	0.1652	0.1652	
PWR Segment 4	0.3030	0.0614	0.3644	0.1822	0.1822	
PWR Segment 5	0.3220	0.0614	0.3834	0.1917	0.1917	< - Peak Release at 46.5 GWd/MTU
PWR Segment 6	0.3040	0.0614	0.3654	0.1827	0.1827	
PWR Segment 7	0.2710	0.0614	0.3324	0.1662	0.1662	

Short-Lived Volitiles

	Nobles (Kr, Xe) - Release/Birth								Halogen (I) - Release/Birth			
	Kr-85m		Kr-87		Kr-88		Xe-133	Xe-135	I-131	I-132	I-133	I-135
Case	FAST	Upper Bound	FAST	Upper Bound	FAST	Upper Bound	Upper Bound	Upper Bound	Upper Bound	Upper Bound	Upper Bound	Upper Bound
Full Envelope	0.0051	0.0254	0.0026	0.0132	0.0036	0.0179	0.0577	0.0330	0.0607	0.0685	0.0365	0.0261
PWR Segment 1	0.0023	0.0114	0.0012	0.0059	0.0016	0.0081	0.0259	0.0148	0.0273	0.0308	0.0164	0.0117
PWR Segment 2	0.0023	0.0115	0.0012	0.0060	0.0016	0.0081	0.0261	0.0149	0.0274	0.0309	0.0165	0.0118
PWR Segment 3	0.0036	0.0178	0.0019	0.0093	0.0025	0.0126	0.0405	0.0232	0.0426	0.0481	0.0256	0.0183
PWR Segment 4	0.0050	0.0252	0.0026	0.0131	0.0036	0.0178	0.0573	0.0328	0.0603	0.0681	0.0363	0.0259
PWR Segment 5	0.0042	0.0208	0.0022	0.0108	0.0029	0.0147	0.0472	0.0270	0.0497	0.0561	0.0299	0.0213
PWR Segment 6	0.0023	0.0114	0.0012	0.0060	0.0016	0.0081	0.0259	0.0148	0.0273	0.0308	0.0164	0.0117
PWR Segment 7	0.0023	0.0114	0.0012	0.0060	0.0016	0.0081	0.0259	0.0148	0.0273	0.0308	0.0164	0.0117

Table 2: PWR Full Envelope Versus SPH Release Fractions

Nuclide	Half-Life	Full Envelope	Maximum SPHs	Difference (absolute)
Xe-133	5.243d	0.0577	0.0573	0.0003
Xe-135	9.10h	0.0330	0.0328	0.0002
Xe-135m	15.3m	- -	- -	
Xe-137	3.82m	- -	- -	
Xe-138	14.1m	- -	- -	
Xe-139	39.7s	- -	- -	
Kr-85	10.76y	0.3994	0.3834	0.0160
Kr-85m	4.48h	0.0254	0.0252	0.0001
Kr-87	1.27h	0.0132	0.0131	0.0000
Kr-88	2.84h	0.0179	0.0178	0.0001
Kr-89	3.15m	- -	- -	
Kr-90	32.3s	- -	- -	
I-131	8.04d	0.0607	0.0603	0.0004
I-132	2.28h	0.0685	0.0681	0.0004
I-133	20.8h	0.0365	0.0363	0.0002
I-134	52.6m	- -	- -	
I-135	6.57h	0.0261	0.0259	0.0002
Cs-134	2.07y	0.1997	0.1917	0.0080
Cs-137	30.1y	0.1997	0.1917	0.0080

Table 3: Recommended PWR Steady-State Radionuclide Release Fractions

Group	Release Fraction
I-131	0.07
I-132	0.07
Kr-85	0.40
Other Noble Gases	0.06
Other Halogens	0.04
Alkali Metals	0.20

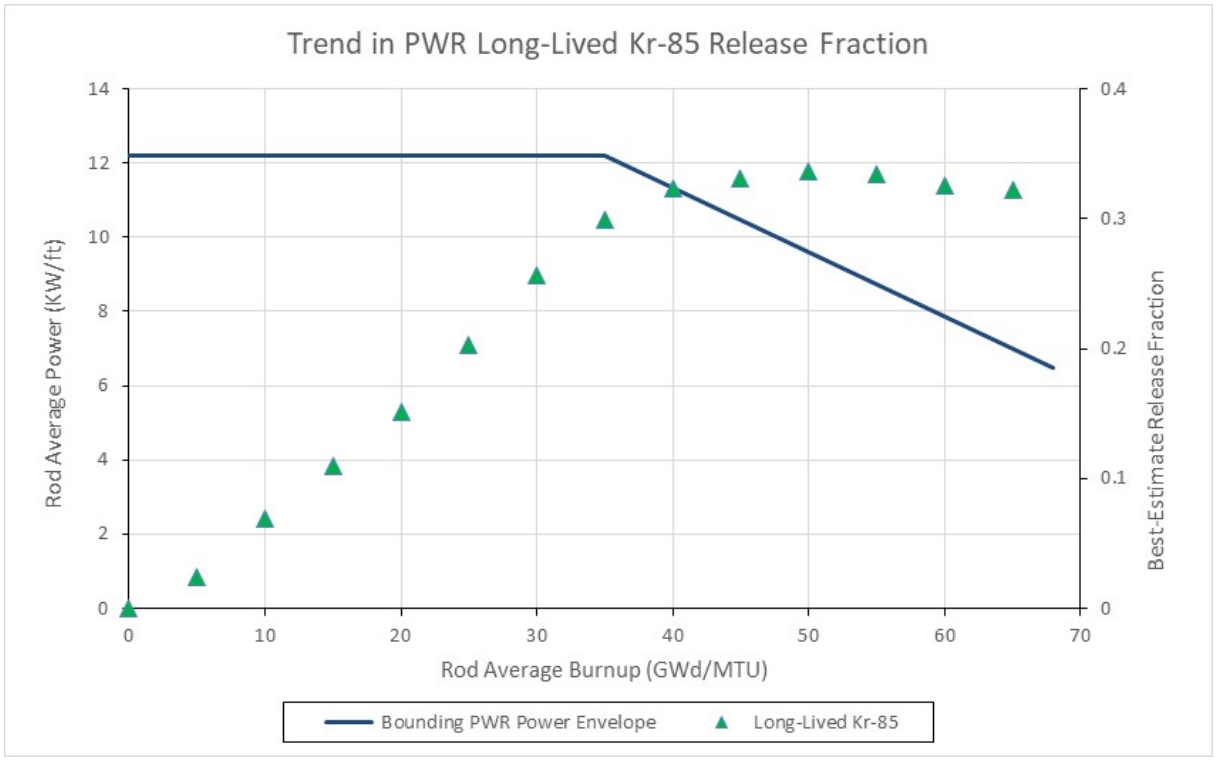
Table 4: FAST BWR Steady-State Radionuclide Release Fractions

Nuclide	Half-Life	Full Envelope
Xe-133	5.243d	**0.0225**
Xe-135	9.10h	**0.0129**
Xe-135m	15.3m	**- -**
Xe-137	3.82m	**- -**
Xe-138	14.1m	**- -**
Xe-139	39.7s	**- -**
Kr-85	10.76y	**0.3174**
Kr-85m	4.48h	**0.0099**
Kr-87	1.27h	**0.0052**
Kr-88	2.84h	**0.0070**
Kr-89	3.15m	**- -**
Kr-90	32.3s	**- -**
I-131	8.04d	**0.0237**
I-132	2.28h	**0.0267**
I-133	20.8h	**0.0142**
I-134	52.6m	**- -**
I-135	6.57h	**0.0102**
Cs-134	2.07y	**0.1587**
Cs-137	30.1y	**0.1587**

Table 5: Recommended BWR Steady-State Radionuclide Release Fractions

Group	**Release Fraction**
I-131	0.03
I-132	0.03
Kr-85	0.32
Other Noble Gases	0.03
Other Halogens	0.02
Alkali Metals	0.16

Figure 3: Trends in PWR Radionuclide Release Fractions



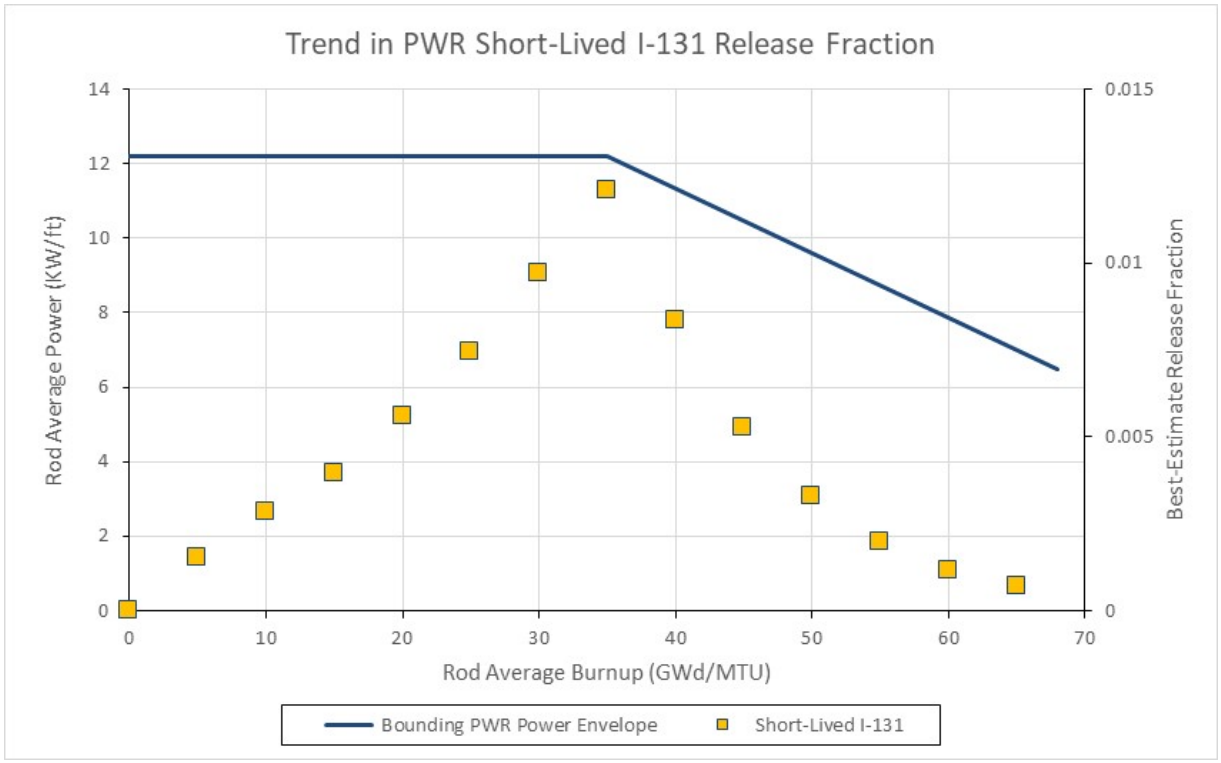
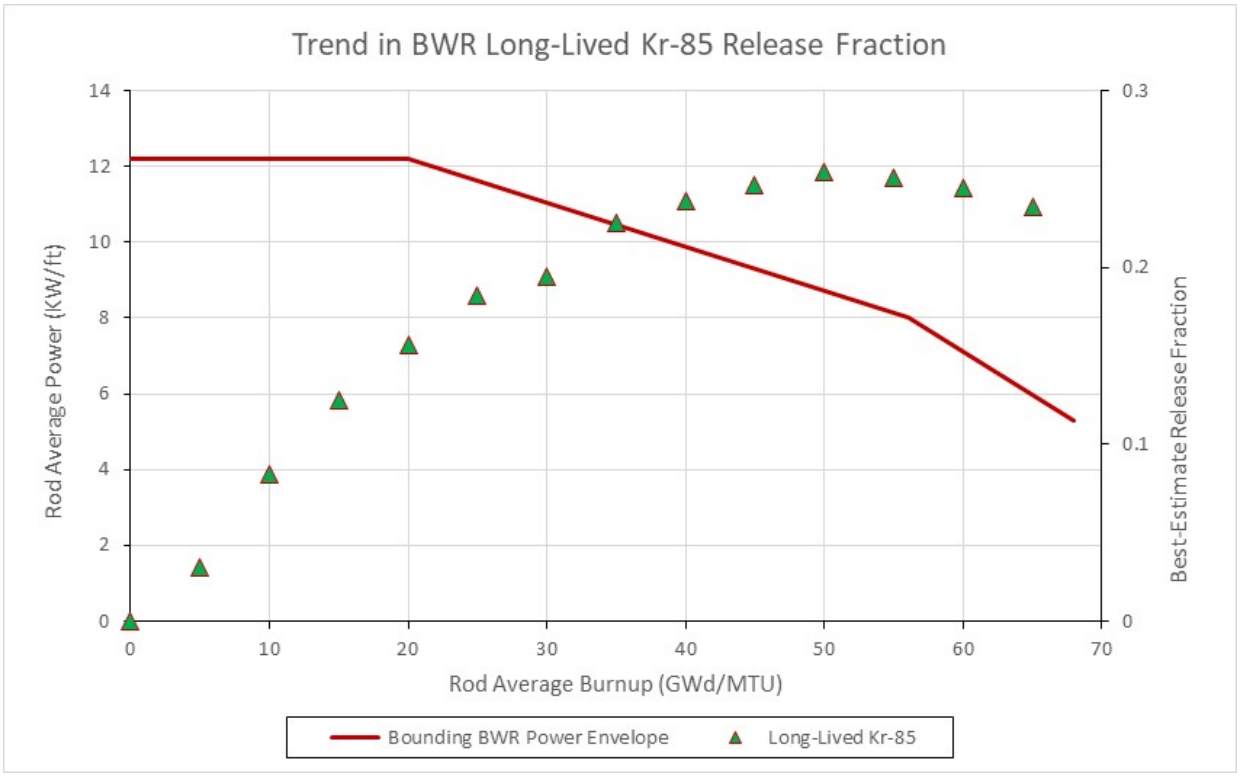
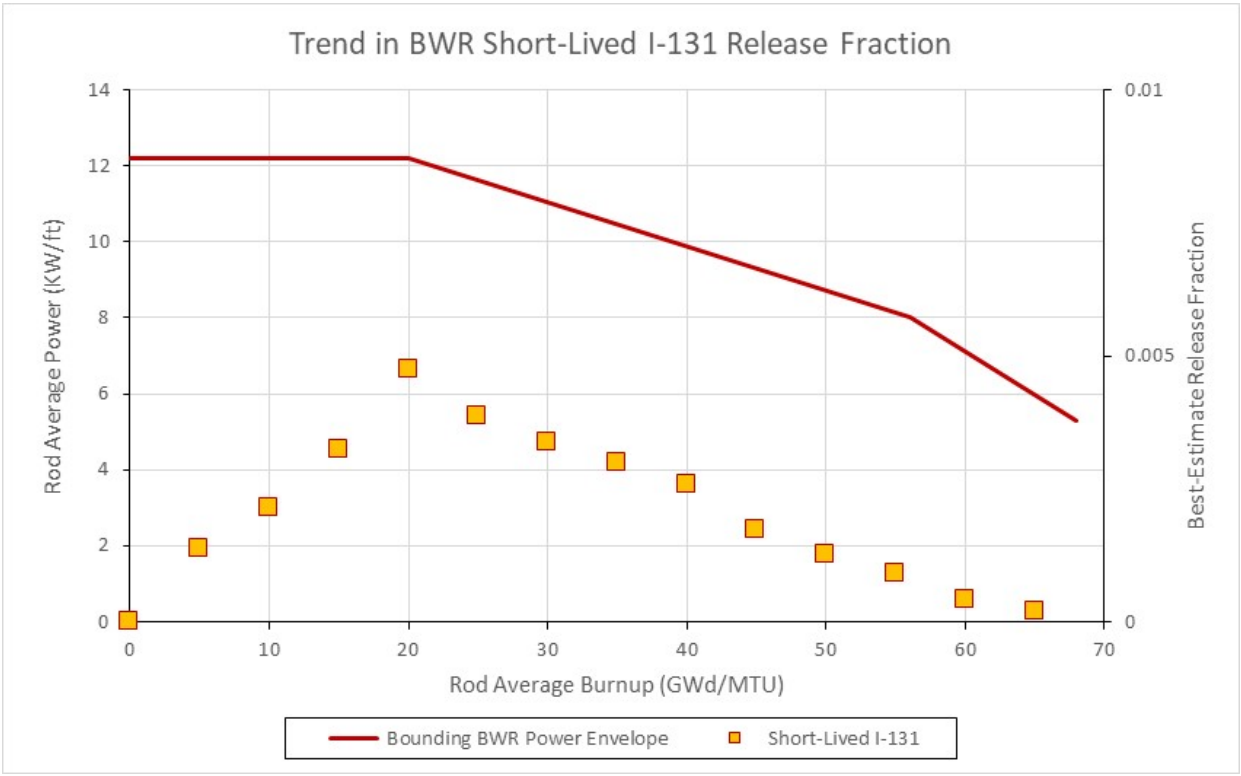


Figure 4: Trends in BWR Radionuclide Release Fractions





PWR Control Rod Ejection and BWR Control Rod Drop Accident Transient Releases

For non-LOCA DBAs involving a rapid increase in fuel rod power, such as the BWR control rod drop accident and PWR control rod ejection accident, additional fission product releases may occur due to pellet fracturing and grain boundary separation. This transient fission gas release (TFGR) increases the amount of activity available for release into the reactor coolant system for fuel rods that experience cladding breach.

Fission gas released during the experiments was measured for many of the prompt power excursion tests which comprise the reactivity-initiated accident (RIA) empirical database. This transient FGR database was originally documented in PNNL Report PNNL-18212, Revision 1, *Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard* (ML111890397). The database was revised, expanded, and captured in Regulatory Guide 1.236, *Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents*, and its technical bases document (ML14188C423).

Figure 5 illustrates the revised TFGR database which is best represented as a function of peak fuel enthalpy rise (∆cal/g). Burnup-dependent TFGR correlations, derived in the RG 1.236 technical bases document, were provided for estimating rod internal pressure and predicting cladding balloon/rupture failure. These same correlations also provide the additional source term for estimating activity released upon cladding breach.

The empirical database suggests that TFGR is sensitive to both local fuel burnup and peak radial average fuel enthalpy rise. As a result, separate low-burnup and high-burnup TFGR correlations are provided, as follows:

pellet burnup < 50 GWd/MTU

TFGR = maximum [(0.26 * ΔH) – 13) / 100, 0]

pellet burnup > 50 GWd/MTU

TFGR = maximum [(0.26 * ΔH) – 5) / 100, 0]

where:

TFGR = transient fission gas release, fraction, and

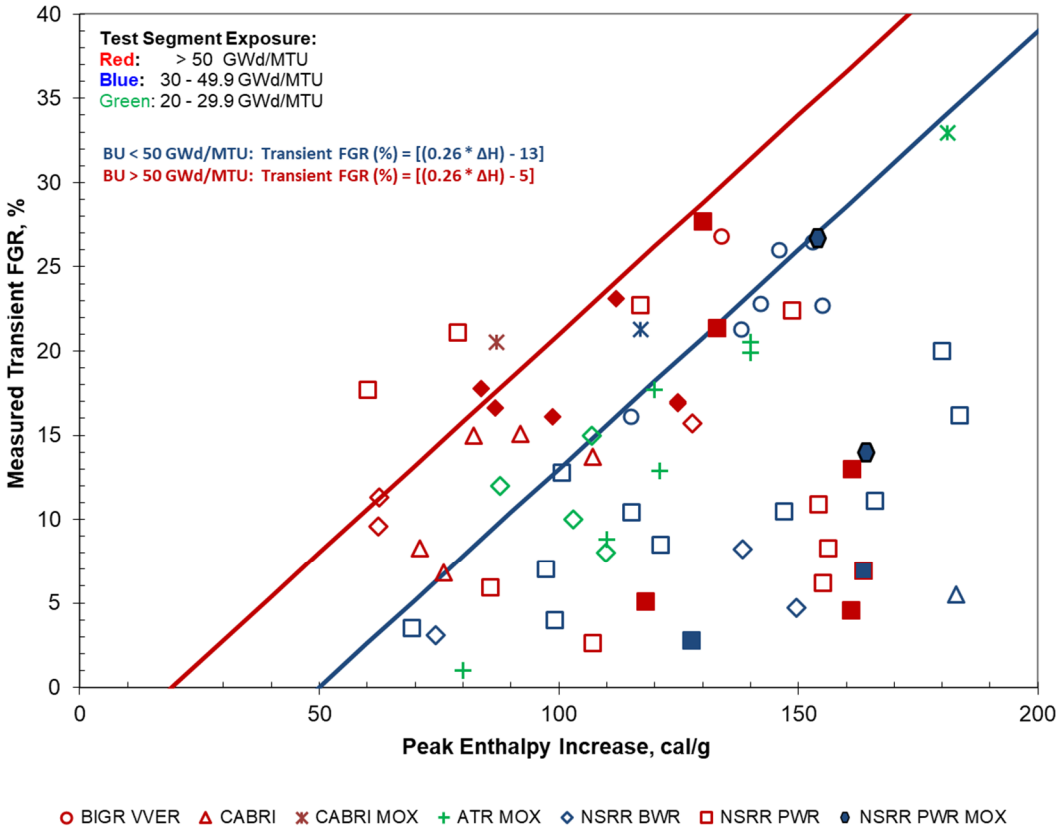
ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

PNNL Report PNNL-18212, Revision 1 provided an investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release. This investigation concluded that adjustments to the above empirically based correlations are needed for different radionuclides. For the stable, long-lived Kr-85 noble gas, the transient fission product release is equivalent to the above burnup-dependent correlations. For volatile, short-lived radioactive isotopes such as halogens (e.g., iodine (I) 131, I 132, I 133, I 135) and xenon (Xe) and Kr noble gases except Kr-85 (e.g., Xe 133, Xe 135, Kr 85m, Kr 87, Kr 88), the transient fission product release correlations should be multiplied by a factor of 0.333.

Attachment 1 to this memorandum documents an alternative treatment for cesium releases and recommends that the transient fission product release for alkali metals is equivalent to the above burnup-dependent correlations with no adjustment.

Figure 5: Transient FGR Database from Prompt Power Testing on Irradiated Fuel Rods

(Solid symbols indicate failed rods; clear symbols indicate intact rods)



SUBJECT: TECHNICAL BASIS FOR DRAFT RG 1.183 REVISION 1 (2021) NON-LOCA FISSION PRODUCT RELEASE FRACTIONS Dated: July 28, 2021

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Attachment 1: Alternative Treatment for Cesium Release Fractions

As recommended by the 1982 and 2011 ANS-5.4 standard, the cesium diffusion coefficient should be assumed to be a factor of 2.0 higher than for the noble gas nuclides. Because release fraction is approximately proportional to the square root of the diffusion coefficient, PNNL Report PNNL-18212, Revision 1, *Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard* (ML111890397) developed the following treatment for cesium releases:

$$(\text{Release Fraction})_{\text{Cs-134, Cs-137}} = (\text{Release Fraction})_{\text{Kr-85}} * (2.0)^{0.5}$$

where: $(\text{Release Fraction})_{\text{Kr-85}}$ is calculated by the fuel performance code.

However, as described in a recent memorandum from the Office of Research (ML20126G376), cesium is expected to behave very differently than krypton once it reaches the grain boundaries. At this point, it can react with other constituents in the fuel to form less volatile compounds that may then accumulate on the grain boundaries as solids or liquids. Cesium released from the fuel may also react with the zirconium in the cladding to form more stable (i.e., non-gaseous) compounds. These effects tend to decrease the inventory of gaseous cesium available for release in the event of a cladding breach during a non-LOCA event. Based upon these chemical interactions and experimental evidence which shows that cesium releases are much lower than noble gas releases during normal operations and power ramps, the Office of Research recommended maintaining the existing cesium release fraction of 0.12 in updated guidance.

Peak rod steady-state radionuclide release fractions are sensitive to fuel design and fuel utilization. One potential issue with defining a constant cesium release fraction (0.12) is that it would remain independent of fuel design and fuel utilization. At the request of the industry, RG 1.183 Table 3 is being revised to support more aggressive fuel utilization. As expected, the empirically tuned release models predict large increases for long-lived noble gas because of these more aggressive fuel rod operating envelopes. And while the above physical properties and chemical effects need to be considered, more aggressive fuel utilization will likely promote larger cesium releases. Therefore, it was decided to develop an alternative treatment for cesium release fraction. One tied to predicted noble gas release; but informed by physical properties and chemical interactions.

The gaseous inventory of cesium residing in the gap and available for release upon cladding breach may be represented by the following equation:

$$(\text{Gap Inventory})_{\text{Cs-134, Cs-137}} = (\text{Release Fraction})_{\text{Kr-85}} * (0.5)$$

Where,

$(\text{Gap Inventory})_{\text{Cs-134, Cs-137}}$ is the amount of gaseous cesium available for release, and

$(\text{Release Fraction})_{\text{Kr-85}}$ is calculated using an approved fuel performance code.

The technical bases for this relationship are described below, starting with key observations from a literature search.

Journal of Nuclear Materials, C.T. Walker et.al., “Observations of the release of cesium from UO2 fuel,” JNM 240 (1996) 32-42.

- Based on Electron Probe Microanalysis (EPMA) and X-ray fluorescence analysis (XRF) of irradiated fuel segments exposed to power ramp and hold conditions, research findings suggest that above 1200°C, cesium and xenon exhibit similar release patterns and effective diffusion coefficients.

- At the rim of high burn-up fuel, cesium and xenon behave differently. Here, cesium appears to be completely retained. Difference in the release patterns exhibited by cesium and xenon at the rim region can be attributed to their physical state. Below 1200°C, cesium is a liquid, is relatively immobile, and forms a film on the grain faces. Whereas, xenon is a gas, forms interconnected bubbles, and is able to be released from the pellet surface.
- Microanalysis of the cladding inner diameter surface reveals a relatively large concentration of cesium in the fuel-to-clad bond layer. Whereas, nobles were not identified.

National Cooperative for the Disposal of Radioactive Waste Technical Report, L. Johnson et.al., "Estimates of the Instant Release Fraction of UO_2 and MOX Fuel at t=0," Technical Report 04-08, November 2004.

- A large database was compiled from various leaching tests. Data suggests the gap release of ^{137}Cs are about one-third that of fission gas release (i.e., noble gases) when FGR exceeds approximately 1.0%. Below this threshold, ^{137}Cs releases to the gap are very similar to FGR.

National Nuclear Laboratory Report, Glyn Rossiter and Mike Mignanelli, "The characteristics of LWR fuel at high burnup and their relevance to AGR spent fuel," NNL (10) 10930, Issue 2, June 2011.

- Since ^{129}I and ^{137}Cs are volatile fission products, the measured rod free volume inventories of these nuclides are relatively large at between 0.002% and 15% of the total amount for ^{129}I, and between 0.2% and 6.5% of the total amount for ^{137}Cs.
- For ^{129}I and ^{137}Cs, the combined rod free volume and grain boundary inventory is correlated to fission gas release. In the case of ^{129}I there is a 1:1 correspondence between the combined inventory and fission gas release over the whole fission gas release (and hence burnup) range, whereas for ^{137}Cs the ratio is roughly 1:1 for fission gas release below ~ 1%, and tends towards 1:3 as fission gas release increases.
- These ratios are consistent with iodine and cesium diffusion coefficients in UO_2 which are approximately equal to (iodine) and one third (cesium) of those of fission gas.
- The release fraction of cesium, together with its radial distribution of cesium and xenon in the fuel, can also be inferred from EPMA measurements. The results are for fuel sample STR013-8R-3 from the Risø transient fission gas release project; this is ~ 20 MWd/kgU BWR fuel which was base irradiated in the Millstone-1 BWR and then subjected to a short hold power ramp in the Risø DR3 reactor. The fission gas release fraction, as measured during rod puncture, was 11.1%. The cesium and xenon concentration profiles are very similar in shape, indicating similar release behavior for the two elements. The release fractions estimated from the EPMA data are 24% for xenon and 11% for cesium. Thus, the fission gas release fraction is approximately 2.2 times the cesium release fraction, in broad agreement with the results from the leaching tests.

Y. Pontillon et.al., "New Insight on Volatile Fission Products (I and Cs) Release from High Burnup UO_2 Fuel Under LOCA Type Conditions.

- Irradiated fuel rod segments were exposed to simulated LOCA heating rates to measure the release of noble gas (^{85}Kr and ^{133}Xe), ^{137}Cs, and ^{131}I. The measured releases for ^{85}Kr were more than 10x higher than ^{137}Cs releases up to 1200°C.

The Information presented above suggests that steady-state cesium releases from the fuel pellet to the fuel rod void volume will be lower than noble gas releases. These observations are often tied to the behavior of cesium, relative to noble gas, below 1200°C (2192°F) where physical properties and chemical effects are notably different. Examination of the FAST calculations for the bounding PWR and BWR fuel rod power profiles reveal that fuel pellet average temperatures remain below 1200°C and pellet surface temperatures remain below 594°C (1100°F). Hence, these observations are applicable to the release fractions in this guidance.

The relationship of noble gas to cesium releases varies between 2.2:1 to 10:1 from the different studies. Given this variability, a conservative lower bound of 2:1 was selected. This relationship accounts for differences in both physical properties and chemical effects and likely bounds the relative inventory available for release upon cladding breach.

With respect to transient fission gas release (TFGR), the above literature search found evidence that (1) cesium and xenon release patterns and effective diffusion coefficients were similar above 1200°C and (2) cesium becomes immobile and gets trapped along grain boundaries and in the porous rim structure below 1200°C. CRE and CRD accidents may experience fuel temperature above 1200°C and the data suggests similar release patterns and diffusion coefficients. However, these prompt power excursions occur over a very short time interval, so diffusion-based releases, even at elevated temperatures, are not anticipated. With respect to the second finding, cesium located in grain boundaries and within the porous rim structure may become mobile at the elevated temperatures and susceptible to release due to pellet fracturing and grain boundary separation during the transient. Given its long half-life and susceptibility to transient release, it is recommended that the transient fission product release for alkali metals is equivalent to the burnup dependent TFGR correlations with no adjustment.

Attachment 2:

Analytical Technique for Calculating Steady-State Release Fractions

This attachment provides an acceptable analytical technique for calculating steady-state non-loss-of-coolant accident (non-LOCA) fission product release fractions residing in the fuel rod void volume (plenum and pellet-to-cladding gap) based on specific fuel rod designs or more realistic fuel rod power histories. The bounding release fractions documented in this memo followed the same analytical procedure. Lower release fractions are achievable using less aggressive rod power histories or less limiting fuel rod designs (e.g., 17x17 versus 14x14 fuel rod design).

Steady-state gap inventories represent radioactive fission products generated during normal steady-state operation that have diffused within the fuel pellet, have been released into the fuel rod void space (i.e., rod plenum and pellet-to-cladding gap), and are available for release upon fuel rod cladding failure. Given the continued accumulation of long-lived radioactive isotopes and the inevitable decay of short-lived radioactive isotopes, the most limiting time-in-life (i.e., maximum gap fraction) for a particular radioactive isotope varies with fuel rod exposure and power history. The analytical technique described in this attachment prescribes the use of fuel rod power profiles based on core operating limits or limiting fuel rod power histories. In addition, this analytical technique produces a composite worst time-in-life (i.e., maximum gap fraction for each radioactive isotope). Therefore, the steady-state fission product gap inventories calculated using this analytical approach will be significantly larger than realistic fuel rod or core-average source terms.

As described in this analytical technique, the long-lived radionuclide release fractions are calculated using an NRC approved fuel rod thermal-mechanical performance code with established modeling uncertainties. As such, the range of applicability of this analytical technique for calculating long-lived radionuclide release fractions is tied directly to the range of applicability of the approved fuel rod thermal-mechanical performance code, along with any limitations and conditions imposed in the NRC's safety evaluation.

The NRC maintains the FAST (formerly FRAPCON and FRAPTRAN) fuel rod thermal-mechanical fuel performance code to perform independent audit calculations for licensing activities. While calibrated and validated against a large empirical database, FAST and its predecessors are not NRC-approved codes and may not be utilized to calculate plant-specific, fuel-specific, or cycle-specific gap inventories without further justification.

As described in this analytical technique, the short-lived radionuclide release fractions are calculated using the ANS-5.4 standard release model documented in NUREG/CR-7003 (2011). Section 4.5 of NUREG/CR-7003 documents key assumptions used in the development of the standard model. These assumptions, listed below, are reasonable and consistent with this application.

1. Gas-atom diffusion from where the atoms are born in the fuel matrix is the mechanism for diffusion to the grain boundaries where the gas precipitates as bubbles.
2. The gas bubbles on the grain boundaries reach a saturation point where the bubbles interlink, releasing the gas to the fuel rod void volume.
3. The criterion for grain boundary interlinkage is a function of only temperature and burnup.
4. Once the grain boundary interlinkage has occurred, the grain boundary remains linked to the rod internal void volume for the rest of the irradiation (i.e., no re-sintering of the grain boundary is experienced).

5. The nuclides are in radioactive equilibrium in terms of release and decay, which is approximately correct if fuel temperatures remain constant for at least three half-lives of the nuclide.
6. Precursor effects can be accounted for by a multiplier, α, on the diffusion coefficient, per Friskney and Speight. The diffusion coefficients for the bromine and tellurium precursors are assumed to be a factor of 4 and 20 greater, respectively, than the noble gases and iodines.

Section 4.5 of NUREG/CR-7003 also documents limitations on the standard model. These limitations, listed below, must be observed with future application.

1. The model does not account for burst releases due to large sudden temperature (power) increases greater than 300°K.
2. The model does not account for transport of fission products after release from the fuel rod due to a cladding defect.
3. The model does not apply to release of stable nuclides or nuclides with half-lives greater than one year. Other models are available to determine release of stable nuclides.
4. Application of the model to predict release with temperature changes on the order of less than three half-lives for an isotope can be performed in a conservative manner by assuming the maximum temperature during the time period has been constant for the three half-lives of the isotope.
5. The model does not apply to release from failed fuel where fuel oxidation is possible. Fuel oxidation can enhance the radiological release.
6. The model does not apply to commercial fuel operation with rod-average burnups greater than 70 MWd/kgU. Data used for model development extend to 90 MWd/kgU but are at relatively low linear heat generation rates between 70 to 90 MWd/kgU.
7. The model is based on fuel with densities between 95% to 98% of theoretical density, grain sizes between 6 to 15 μm (mean linear intercept), and open porosities between 0.1% to 3.0% theoretical density. Therefore, the model should only be applied to fuel within these ranges.
8. The model is based on release from UO2 fuel only; therefore, it currently does not apply to (U,Pu)O2 fuel or fuel types other than UO2.

Analytical Technique:

The analytical technique used to calculate steady-state gap inventories should include the following attributes:

1. For stable, long-lived radioactive isotopes, such as krypton (Kr)-85, an NRC-approved fuel rod thermal-mechanical performance code with established modeling uncertainties should be used to predict the integral FGR. The code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable FGR data up to the licensed burnup of the fuel rod design.

    1.1. Long-lived radioactive isotopes will continue to accumulate throughout exposure, with insignificant decay because of their long half-lives. For this reason, maximum gap inventories for long-lived isotopes are likely to occur near or at the end of life of the fuel assembly.

    1.2. Cesium is expected to behave differently than noble gas once it reaches the grain boundaries. At this point, it may react with other constituents in the fuel to form less

volatile compounds that may then accumulate on the grain boundaries as solids or liquids. Cesium released from the fuel may also react with the zirconium in the cladding to form more stable (i.e., non-gaseous) compounds. These effects tend to decrease the inventory of gaseous cesium available for release in the event of a cladding breach. To account for these effects, the following relationship is recommended:

$$(\text{Gap Inventory})_{\text{Cs-134, Cs-137}} = (\text{Release Fraction})_{\text{Kr-85}} * (0.5)$$

Where $(\text{Gap Inventory})_{\text{Cs-134, Cs-137}}$ is the amount of gaseous cesium available for release and $(\text{Release Fraction})_{\text{Kr-85}}$ is calculated using an approved fuel performance code with applied uncertainties.

2. For volatile, short-lived radioactive isotopes, such as iodine (I) (i.e., I-131, I-132, I-133, and I-135) and xenon (Xe) and Kr noble gases (except for Kr-85) (i.e., Xe-133, Xe-135, Kr-85m, Kr-87, and Kr-88), an NRC-approved release model or an NRC-endorsed ANS-5.4 release model should be used to predict the release-to-birth (R/B) fraction using fuel parameters at several depletion time steps from an NRC-approved fuel rod thermal-mechanical performance code. The fuel parameters necessary for use in the NRC-endorsed ANS-5.4 model calculations of the R/B fraction are local fuel temperature, fission rate, and axial node/pellet burnup. Consistent with Item 1, the code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable FGR data up to the licensed burnup of the fuel rod design.

   Because of their relatively short half-lives, the amount of activity associated with volatile radioactive isotopes depends on their rate of production (i.e., fission rate and cumulative yield), rate of release, and rate of decay. Maximum (R/B) ratios for short-lived isotopes are likely to occur at approximately the maximum exposure at the highest power level (i.e., knee in the power operating envelope).

   2.1. NUREG/CR-7003 (PNNL-18490), “Background and Derivation of ANS-5.4 Standard Fission Product Release Model,” issued January 2010, provides guidance on using the NRC-endorsed ANS-5.4 release model to calculate short-lived (R/B) factors.

   2.2. For nuclides with half-lives of less than 1 hour, no gap inventories are provided. Because of their rapid decay (relative to the time for diffusion and transport), these nuclides will be bounded by the calculated gap fractions for longer lived nuclides under the headings “Other Noble Gases” and “Other Halogens” as shown in the example below.

   2.3. For nuclides with half-lives of less than 6 hours, an approved fuel performance code is applied to predict the R/B fraction using Equation 12 in NUREG/CR-7003 and its definitions of terms, as follows:

$$\left(\frac{R}{B}\right)_{i,m} = \left(\frac{S}{V}\right)_{i,m} \sqrt{\frac{\alpha_{nuclide} D_{i,m}}{\lambda_{nuclide}}}$$

   Where

   R is release rate (atoms-cm^{-3}s^{-1}),

   B is production rate (atoms-cm^{-3}s^{-1}),

   S is surface area (cm^2),

   V is volume (cm^3),

   α accounts for precursor enhancements effects and is defined below,

D is diffusion coefficient (cm^2s^{-1}), and

λ is half-life (s^{-1})

2.4. For nuclides with half-lives of greater than 6 hours, the R/B fraction is predicted by multiplying the fractal-scaling factor ($F_{nuclide}$) by the predicted Kr-85m (R/B) using Equation 13 of NUREG/CR-7003, as follows:

$$\left(\frac{R}{B}\right)_{i,nuclide} = F_{nuclide}\left(\frac{S}{V}\right)_i \sqrt{\frac{\alpha_{Kr-85m} D_i}{\lambda_{Kr-85m}}}$$

The R/B fraction for isotope I-132 should be calculated using this fractal equation even though its half-life is less than 6 hours (2.28 hours) because its precursor of tellurium (Te)-132 has a half-life of 3.2 days, which controls the release of I-132.

2.5. Table A2-1 lists the fractal-scaling factors for each nuclide calculated using the following equation from NUREG/CR-7003:

$$F_{nuclide} = \left(\frac{\alpha_{nuclide}\lambda_{Kr-85m}}{\lambda_{nuclide}\alpha_{Kr-85m}}\right)^{0.25}$$

3. High-confidence, upper-tolerance release fractions should be calculated using an NRC-approved fuel rod thermal-mechanical code along with quantified model uncertainties.

3.1. For short-lived isotopes, the 2011 ANS-5.4 release model standard recommends multiplying the best-estimate predictions by a factor of 5.0 to obtain upper-tolerance release fractions.

3.2. For long-lived isotopes, established model uncertainties associated with the NRC-approved fuel rod thermal-mechanical code should be applied, either deterministically or sampled within a statistical application methodology, to obtain high-confidence upper-tolerance release fractions.

4. Nominal fuel design specifications (excluding tolerances) may be used.

5. Actual in-reactor fuel rod power histories may diverge from reload core depletion calculations because of unplanned shutdowns or power maneuvering. As a result, the rod power history or histories used to predict gap inventories should bound anticipated operation. Rod power histories used in the fuel rod design analysis based on core operating limits report thermal-mechanical operating limits or radial falloff curves may be used. The fuel rod power history used to calculate gap inventories should be verifiable.

5.1. A segmented power history which separates the bounding power history into several segments, each running along the peak for a prescribed period, may be used to calculate release fractions.

6. Higher local power density (Fq) promotes more local FGR. Higher rod average power (Fr), along with a flatter axial power distribution (Fz), promotes more FGR along the fuel stack. Sensitivity cases should be conducted to ensure the limiting fuel rod power history is captured.

7. Each fuel rod design (e.g., UO_2, UO_2-Gd_2O_3, part-length, full-length) should be evaluated.

8. The minimum acceptable number of radial and axial nodes as defined in ANS-5.4 should be used, along with the methodology of summing the release for these nodes, to determine the overall release from the fuel pellets to the fuel void volume.

Table A2-1. Fractal Scaling Factors for Short-Lived Nuclides

NUCLIDE	NUREG/CR-7003, TABLE 1			FRACTAL SCALING FACTOR
	Half-Life	Decay Constants	Alpha	
Xe-133	5.243 days	1.53E-06	1.25	2.276
Xe-135	9.10 hours	2.12E-05	1.85	1.301
Xe-135m	15.3 months	7.55E-04	23.50	1.005
Xe-137	3.82 months	3.02E-03	1.07	0.328
Xe-138	14.1 months	8.19E-04	1.00	0.447
Xe-139	39.7 seconds	1.75E-02	1.00	0.208
Kr-85m	4.48 hours	4.30E-05	1.31	1.000
Kr-87	1.27 hours	1.52E-04	1.25	0.721
Kr-88	2.84 hours	6.78E-05	1.03	0.840
Kr-89	3.15 months	3.35E-03	1.21	0.330
Kr-90	32.3 seconds	2.15E-02	1.11	0.203
I-131	8.04 days	9.98E-07	1.00	2.395
I-132	2.28 hours	8.44E-05	137*	2.702
I-133	20.8 hours	9.26E-06	1.21	1.439
I-134	52.6 months	2.20E-04	4.40	0.900
I-135	6.57 hours	2.93E-05	1.00	1.029

* The I-132 alpha term accounts for significant contribution from precursor Te-132.

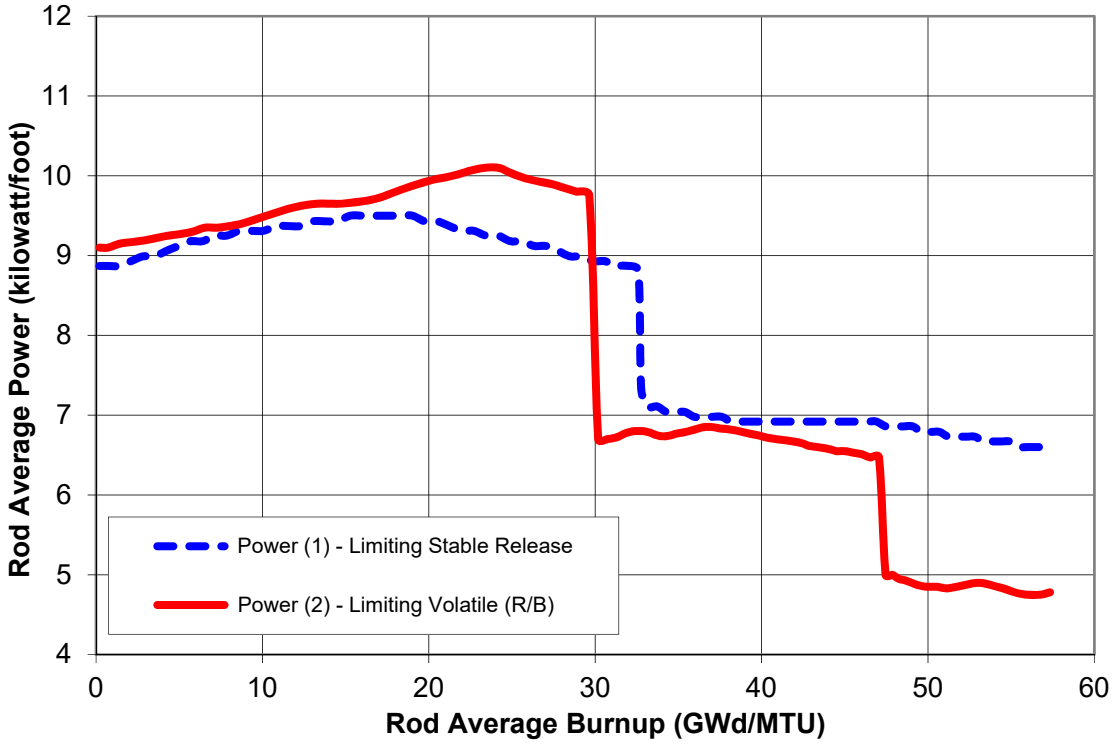
Attachment 3: Sample Calculation of Steady-State Release Fractions

In this sample calculation, the licensee elects to perform cycle-specific calculations to verify the continued applicability of plant-specific steady-state release fractions docketed in the UFSAR Chapter 15 radiological consequences.

For this cycle, the licensee surveys the reload depletion and identifies the limiting fuel rod power histories for long-lived stable isotopes and short-lived volatile isotopes. Examples are shown in Figure A3-1. The licensee then makes adjustments to account for power uncertainties and plant maneuvering. The figure below illustrates the limiting fuel rod power history for the calculation of stable releases and volatile (R/B) ratios. The licensee has verified that the uranium dioxide (UO_2) fuel rod design is more limiting than co-resident $(U,Gd)O_2$ fuel rods.

As discussed in the analytical procedure, maximum stable releases occur near end of life in fuel rods with a relatively high power during their second cycle of operation. Maximum volatile (R/B) ratios occur near the highest rod power in the low to mid-burnup range. A high probability exists that the limiting fuel rod design and power history identified for the fuel rod thermal-mechanical rod internal pressure analysis will coincide with that for maximum stable releases. Similarly, the limiting fuel rod design and power history identified for the fuel rod thermal-mechanical anticipated operational occurrence fuel centerline melt analysis will likely coincide with that for maximum (R/B) ratios.

Figure A3-1: Limiting Cycle-Specific Fuel Rod Power Histories



In this example, an earlier version of the FRAPCON code with the ANS-5.4 release model was used to calculate the release fraction for stable nuclide Kr-85 and (R/B) ratios for volatile Kr-85m, Kr-87, and Kr-88 at each depletion time step for the two limiting fuel rod power histories. While this example employs the FRAPCON code, licensees should use an NRC-approved fuel performance code with established modeling uncertainties.

Long-Lived Stable Release:

$Kr\text{-}85_{Upper\ Bound}$ = $[(Fission\ gas\ release)_{FRAPCON} + (Uncertainty\ Term)_{FRAPCON}]$
Power 1 = [(0.0614) + (2.36*0.028)] = 0.1275
Power 2 = [(0.0474) + (2.36*0.028)] = 0.1135

$Cs\text{-}137_{Upper\ Bound}$ = $[(Kr\text{-}85)_{Upper\ Bound} * (0.5)]$
Power 1 = [(0.1275 * (0.5)] = 0.0638
Power 2 = [(0.1135 * (0.5)] = 0.0568

Short-Lived Volatile (R/B) Ratio:

$Kr\text{-}85m_{Upper\ Bound}$ = $[(Maximum\ Kr\text{-}85m\ R/B)_{FRAPCON} * 5.0]$
Power 1 = [(0.0008) * 5.0] = 0.0040
Power 2 = [(0.0013) * 5.0] = 0.0065

$Kr\text{-}87_{Upper\ Bound}$ = $[(Maximum\ Kr\text{-}87\ R/B)_{FRAPCON} * 5.0]$
Power 1 = [(0.0004) * 5.0] = 0.0020
Power 2 = [(0.0006) * 5.0] = 0.0030

$Kr\text{-}88_{Upper\ Bound}$ = $[(Maximum\ Kr\text{-}88\ R/B)_{FRAPCON} * 5.0]$
Power 1 = [(0.0006) * 5.0] = 0.0030
Power 2 = [(0.0009) * 5.0] = 0.0045

$Xe\text{-}133_{Upper\ Bound}$ = $[(Maximum\ Kr\text{-}85m\ R/B)_{FRAPCON} * 5.0 * (Fractal\ Scaling)_{Xe\text{-}133}]$
Power 1 = [(0.0008) * 5.0 * 2.276] = 0.0091
Power 2 = [(0.0013) * 5.0 * 2.276] = 0.0148

$Xe\text{-}135_{Upper\ Bound}$ = $[(Maximum\ Kr\text{-}85m\ R/B)_{FRAPCON} * 5.0 * (Fractal\ Scaling)_{Xe\text{-}135}]$
Power 1 = [(0.0008) * 5.0 * 1.301] = 0.0052
Power 2 = [(0.0013) * 5.0 * 1.301] = 0.0085

$I\text{-}131_{Upper\ Bound}$ = $[(Maximum\ Kr\text{-}85m\ R/B)_{FRAPCON} * 5.0 * (Fractal\ Scaling)_{I\text{-}131}]$
Power 1 = [(0.0008) * 5.0 * 2.395] = 0.0096
Power 2 = [(0.0013) * 5.0 * 2.395] = 0.0156

$I\text{-}132_{\text{Upper Bound}}$ = [(Maximum Kr-85m R/B)$_{\text{FRAPCON}}$ * 5.0 * (Fractal Scaling)$_{\text{I-132}}$]
Power 1 = [(0.0008) * 5.0 * 2.702] = 0.0108
Power 2 = [(0.0013) * 5.0 * 2.702] = 0.0176

$I\text{-}133_{\text{Upper Bound}}$ = [(Maximum Kr-85m R/B)$_{\text{FRAPCON}}$ * 5.0 * (Fractal Scaling)$_{\text{I-133}}$]
Power 1 = [(0.0008) * 5.0 * 1.439] = 0.0058
Power 2 = [(0.0013) * 5.0 * 1.439] = 0.0094

$I\text{-}134_{\text{Upper Bound}}$ = [(Maximum Kr-85m R/B)$_{\text{FRAPCON}}$ * 5.0 * (Fractal Scaling)$_{\text{I-134}}$]
Power 1 = [(0.0008) * 5.0 * 0.900] = 0.0036
Power 2 = [(0.0013) * 5.0 * 0.900] = 0.0059

$I\text{-}135_{\text{Upper Bound}}$ = [(Maximum Kr-85m R/B)$_{\text{FRAPCON}}$ * 5.0 * (Fractal Scaling)$_{\text{I-135}}$]
Power 1 = [(0.0008) * 5.0 * 1.029] = 0.0041
Power 2 = [(0.0013) * 5.0 * 1.029] = 0.0067

The licensee's cycle-specific steady-state release fraction calculations confirm the bounding plant-specific values assumed in the UFSAR.

GROUP	**GAP INVENTORY**	
	UFSAR	**Cycle-Specific**
I-131	0.04	0.02
I-132	0.04	0.02
Kr-85	0.20	0.13
Other Noble Gases	0.04	0.02
Other Halogens	0.04	0.01
Alkali Metals	0.10	0.07

Attachment 4:
Reconstruction of Technical Bases and Recommendations
for RG 1.183 Revision 00 Non-LOCA Release Fractions

Purpose

The purpose of this attachment is to reconstruct the technical bases for the steady-state release fractions in RG 1.183, Revision 00, Table 3 (2000). A draft technical bases document for the values presented in Table 3 was initiated but does not appear to have been finalized. This investigation will identify areas no longer supported by the current state-of-knowledge, along with recommendations for the future use of these release fractions.

Technical Bases (2000)

Table 3 of RG 1.183 Revision 00 provides the following non-LOCA steady-state radionuclide release fractions, along with limits on its applicability in Footnote 11.

Table 3[11]
Non-LOCA Fraction of Fission Product Inventory in Gap

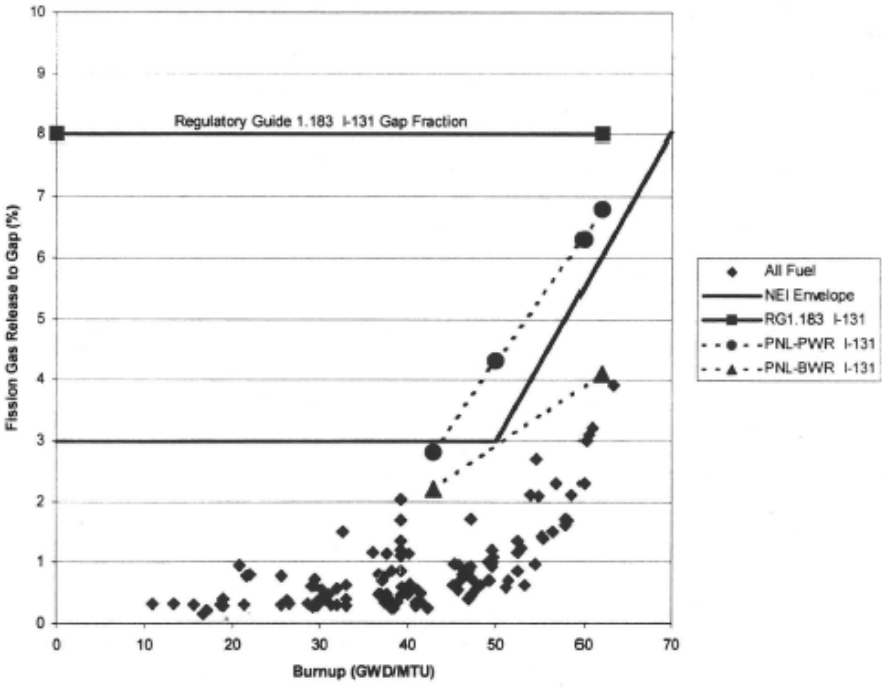
Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

[11] The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.

Based upon review of the draft technical bases document, the non-LOCA steady-state release fractions were based on engineering judgement, backed by some preliminary, best-estimate FRAPCON calculations, and NEI-supplied rod puncture data. The discussion of the bases for the Table 3 release fractions in the draft document is summarized below.

The plot below illustrates the information available at the time the technical bases was drafted.

- Diamond shaped symbols are the NEI-supplied rod puncture data. The NEI comment stated that the data were based on sampling the xenon and krypton gases captured from punctured fuel rods and comparing this activity with the quantity of xenon and krypton generated in a particular fuel rod. This FGR represents stable and long-lived isotopes. NEI proposed that these FGRs were appropriate for the unmeasured short-lived isotopes based on the assumption that the diffusion rates were similar.
- Based upon their data, NEI proposed the solid black line, which ramps up from 3% starting at 50 GWd/MTU rod average burnup.
- The two dotted lines represent preliminary, best-estimate FRAPCON-3 predictions (circle – PWR, triangle – BWR) for I-131 release fractions.



According to the draft memo, the staff reviewed the data supplied by the industry and determined that the data, by themselves, did not provide a basis to accept the industry proposal. This determination was based on: (1) the data are from fuel used in current reactors which likely was not exposed to heating rates associated with non-accident power transients; (2) the historical data could have limited applicability to fuel used in today's more aggressive fuel management programs; (3) the proposed envelope curve is based on a slope of 2.3 percent per 10 GWd/MTU above 50 GWd/MTU. However, a visual comparison of the slope of the data between 50 and 60 GWd/MTU and that for the data above 60 GWd/MTU, shows an increasing slope that is not reflected in the proposed envelope curve; (4) the industry data for rods with burnup greater than 45 GWd/MTU is based on LHGRs less than 5 kW/ft, whereas the high temperature release from the fuel is very temperature related and becomes significant at LHGRs greater than 6 Kw/ft, and (5) given uncertainties in FGR assessments, the extrapolation beyond the last measured data point can not be supported. The staff did note that the data provided represent longer-lived noble gases and that the gap fraction for shorter-lived nuclides could be less.

The staff initially considered using a 1-131 gap fraction of 5 percent for burnups from 0 to 40 GWd/MTU and a value of 8 percent for burnups between 40 and 62 GWd/MTU. However, this approach was removed from consideration due to uncertainties regarding the generic applicability of the assumed power histories. Based on the review of these data, the staff determined that the follow gap fractions would be acceptable for use in DBA analyses.

Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
1-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

These values bound the data obtained for both PWRs and BWRs. The staff believes that the data in Table 3 of the final guide bound the likely radioactivity releases during DBA accidents (other than LOCAs) involving clad failures, with reasonable certainty. However, the staff determined that it would be inappropriate to distinguish between PWR and BWR gap fractions given the limited fuel types considered in the preliminary, best-estimate FRAPCON-3 calculations supporting draft NUREG/CR-6703, "Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU."

Although the traditional approach to fuel management has placed higher burnup fuel in lower power (and lower linear heat generation rate) positions in the core, the staff became aware that in some long-life core load designs, an expended high burnup assembly is placed at the core center to minimize the likelihood of exceeding fuel temperature criteria that might occur if a fresh assembly were to be in that position. Since this arrangement is not consistent with the power histories assumed in the preliminary, best-estimate FRAPCON-3 calculations supporting draft NUREG/CR-6703 (i.e., as burnup increased, linear heat generation rate (LGHR) decreased), the staff elected to impose a restriction on the LHGR for fuel with burnup more than 54 GWd/MTU. This limitation is expressed in Footnote 11 on page 1.183-14. It is important to recognize that this limitation applies only to those fuel rods having a peak rod average burnup greater than 54 GWd/MTU. Other fuel in the core load would not be subject to this limitation.

As noted in the draft technical bases memo, draft NUREG/CR-6703 provides a disclaimer that although conservative assumptions were made, the assumed power histories may be less conservative than those that would have been used in performing a licensing analysis, since they do not include normal operating power transients. The draft reports do note, however, that the likelihood of a normal operating power transient severe enough to significantly change the release fractions for the peak rod has a low probability. It is important to note that the peak-rod average analysis for gap fraction assumes that the same rod is the peak operating rod in each of the irradiation cycles. A rod that is in the peak power position in the first cycle, will not be in the peak power position during the second cycle. This is conservative. It is important to consider how these data are used in radiological accident analyses when evaluating the level of conservatism. The fission product inventory of rods projected to be breached is based on the core-average inventory conservatively multiplied by the maximum radial peaking factor for any rod in the core. The gap fraction as discussed above is conservatively based on the peak rod average burnup. While it is entirely credible that the rod with the peak operating power could be the damaged rod; and while it is also credible that a rod with the peak rod average burnup could be the damaged rod; it is increasingly conservative to assume that the damaged rod is both the peak burnup and peak power rod, since the rods with these characteristics are generally in different regions of the core. The staff has determined that its use of the preliminary, best-estimate FRAPCON-3 calculations supporting draft NUREG/CR-6703 is appropriate given the considerations above and the comparative use of the data in this paper.

As noted in the draft technical bases memo, the values in Table 3 of RG 1.183 are not applicable to reactivity excursion accidents such as the PWR rod ejection accident (REA) and the BWR control

rod drop accident (CRDA). For these accidents the noble gas and iodine gap fractions are assumed to be 10%. For these accidents, the staff has decided to retain the values stated in SRP Sections 15.4.8 and 15.4.9 respectively. The staff decision is based on the recent foreign testing results (e.g., CABRI, 1993) that suggest that the current NRC approach to modeling these events and the associated fuel criteria may not achieve their intended purpose for high burnup fuel. When the NRC became aware of these results, an assessment was made based on probability and power level, and the staff concluded at that time that there was no safety concern requiring immediate regulatory action. These events were further considered in the Agency Program Plan for High-Burnup Fuel (dated July 6, 1998). This report included the conclusion, based on the NRC's current interpretation of the data, generic safety assessments, and the low probability of BWR CRDA and PWR REA events, that no re-analysis would be required for existing approvals. The NRC is participating in international cooperative programs on this issue. There remains considerable disagreement in the international community as to appropriate acceptance criteria. As such, the staff does not believe that adopting any revised criteria is appropriate at this time. This issue will need to be re-visited in about 3-5 years as significant results of these programs become available.

Potential Issues with Technical Bases (2000)

The state-of-knowledge regarding fission product release during steady-state operations and during transients has increased dramatically. But this is not the only thing driving concerns with the original technical basis. As described in the list below, there were looming issues at the time the RG was issued.

1. By the time NUREG/CR-6703 was issued in 2001, the best-estimate FRAPCON-3 calculations supporting this environmental impact changed. The table below show the preliminary and final calculations. Examination of these comparisons show that the final calculations do not support the I-131 and Alkali metal release fractions in Table 3.

Group	RG 1.183 Table 3	Preliminary Maximum Best-Estimate Predictions	NUREG/CR-6703 Maximum Best-Estimate Predictions
I-131	**0.08**	**0.068**	**0.087**
Kr-85	**0.10**	**0.079**	**0.10**
Other Noble Gases	**0.05**	**0.028**	**0.036**
Other Halogens	**0.05**	**- -**	**- -**
Alkali Metals	**0.12**	**0.11**	**0.14**

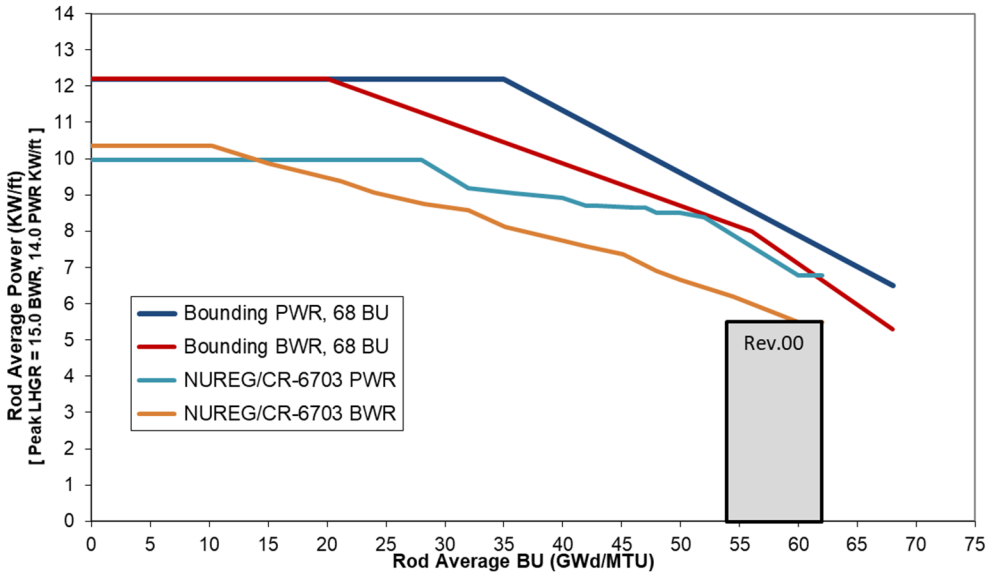
2. NUREG/CR-6703 is an environmental assessment and not intended for licensing applications. The following disclaimer is included in this report.

   This analysis is not intended to be a licensing analysis. A licensing analysis would have several conservatisms not included in this analysis; for example, in a traditional licensing analysis uncertainty associated with fuel fabrication, model predictions, and normal (power) operating transients are considered in estimating release fractions. Licensing analyses predict release fractions that are a factor of 1.8 to a factor of 2.5 greater than those predicted in a best-estimate analysis. Fabrication and model predictions generally account for a factor of 1.5 to a factor of 2.0 difference.

3. Footnote 11 limitation on applicability (6.3 KW/ft beyond 54 GWd/MTU) is inadequate. Limitations on rod power prior to 54 GWd/MTU are needed to ensure applicability. Short-lived volatile radionuclides (e.g., I-131) usually occur at a lower burnup, higher power operating point. Figure A4-1 illustrates the footnote 11 range of applicability (shown as grey box), the bounding power profiles used in the Rev.1 (2021) technical bases, and the

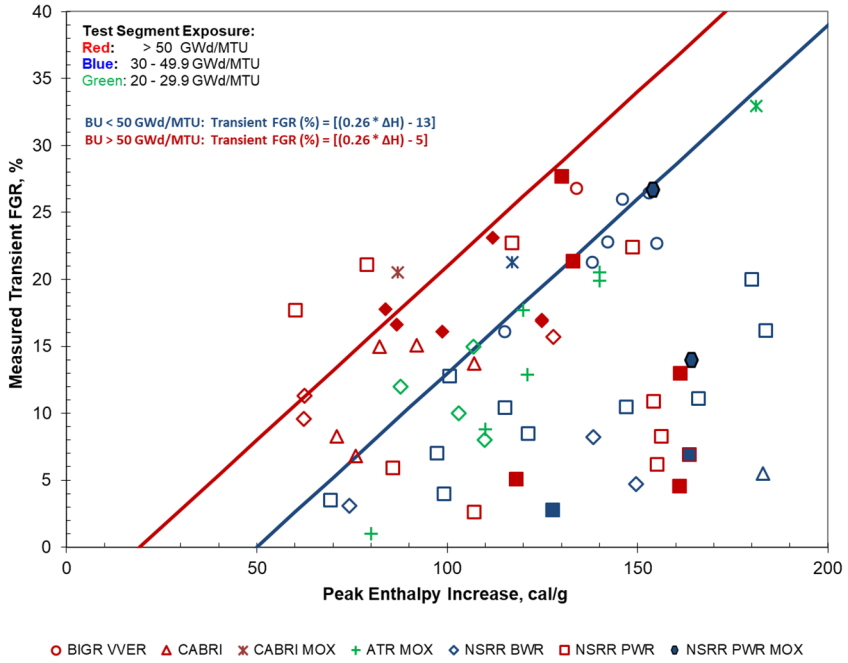
NUREG/CR-6703 rod power histories. Note that the staff is uncertain whether the NUREG/CR-6703 power histories are consistent with the preliminary FRAPCON calculations described earlier. As shown on Figure A4-1, the NUREG/CR-6703 rod power histories are much more restrictive than the bounding curves which were negotiated with industry to support current fuel utilizations.

Figure A4-1: Comparison of Rod Power Histories



4. The draft technical bases document acknowledges that further investigation is needed to defend the 10% iodine and noble releases under CRE and CRD accidents. Figure A4-2 illustrates the transient FGR empirical database under prompt-critical power excursion conditions. Examination of this figure reveals that 10% total release fraction is not supported by the data. Note that the burnup-dependent, deposited energy-dependent transient FGR would need to be added to the existing steady-state releases (i.e., 8% I-131, 10% Kr-85).

Figure A4-2: Transient Fission Gas Release Empirical Database



Recommendation for Future Applications

RG 1.183, Revision 00, Table 3 (2000) has flawed technical bases which have been further overtaken by both changes in fuel design, an expanded empirical database, and improved analytical models. These advancements are not all in the negative direction with respect to steady-state release fractions and there are often competing effects. Competing effects are summarized below.

- The legacy FRAPCON-3 model is known to underestimate fuel temperature and FGR. Application of a modern fuel rod thermal-mechanical code would tend to increase predicted release fractions.
- The preliminary, best-estimate FRAPCON-3 predictions did not consider modelling uncertainties. Application of modelling uncertainties is appropriate for licensing calculations and would tend to increase predicted release fractions.
- The preliminary, best-estimate FRAPCON-3 predictions were based on BWR 8x8 and PWR 17x17 fuel rod designs. Application of a BWR 10x10 fuel rod design (8x8 and 9x9 configuration no longer operating), which operate at lower LHGRs and fuel temperatures, would tend to decrease predicted release fractions. On the other hand, application of a PWR 14x14 fuel rod design, which operate at higher LHGRs and fuel temperatures, would tend to increase predicted release fractions (relative to the 17x17 design).
- The preliminary, best-estimate FRAPCON-3 predictions for the short-lived, volatile radionuclides (e.g., I-131) were based on ANS-5.4 standard release model (1982). Application of the ANS-5.4 standard release model (2011) would significantly reduce predicted release fractions.

The cumulative impact of these competing effects is difficult to quantify. Using engineering judgment, informed by sensitivity studies between legacy FRAPCON-3.4 and FAST-1.0 code predictions and comparisons between ANS-5.4(1982) and ANS-5.4(2011) standard release model predictions, the staff has determined that the likely combined effects are that the long-

lived, stable radionuclide (e.g., Kr-85, Cs-134) release fractions are underpredicted and the short-lived, volatile radionuclides (e.g., I-131) are overpredicted.

Radionuclide release fractions are very sensitive to power operating history. Trends in reactor operations have changed significantly since 2000 and there are competing effects with respect to predicted release fractions.

- BWRs have migrated to 10x10 and 11x11 fuel bundle configurations which increase the core linear feet of fuel, reducing LHGR. A reduction in LGHR would tend to decrease predicted release fractions.
- BWRs have performed extended power uprates (EPU). This change increases LHGR which, in turn, would tend to increase predicted release fractions.
- Combustion Engineering PWRs have replaced B_4C poison rods with fuel rods which increases the core linear feet of fuel, reducing LHGR. A reduction in LGHR would tend to decrease predicted release fractions.
- PWRs have performed power uprates. This change increases LHGR which, in turn, would tend to increase predicted release fractions.
- Batch average ^{235}U enrichment and discharge burnup has increased. This would tend to increase predicted release fractions.
- Since the removal of the design limit restricting fuel rod internal pressure below system pressure, predicted release fractions have increased.

Each plant has evolved their power operating regime over the past two decades. The extent of change is highly plant-specific and potential impacts on release fractions are difficult to quantify. Based upon increases in predicted rod internal pressure, it is likely that release fractions are trending up.

The likely cumulative effects of changes in fuel design, plant operations, and modern analytical models are that the RG 1.183, Revision 00, Table 3 long-lived, stable radionuclide (e.g., Kr-85, Cs-134) release fractions are underpredicted and the short-lived, volatile radionuclides (e.g., I-131) are overpredicted. Given the importance of I-131 on predicted radiological consequences, it is likely that individual plant dose calculations remain conservative. However, lacking a meaningful limit on applicability, it is difficult to ensure this assessment remains valid. It is recommended that future licensing applications involving changes in fuel design or fuel utilization should include justification for the continued use of RG 1.183, Revision 00, Table 3 steady-state non-LOCA release fractions. Alternatively, Revision 01 should be used.

With respect to radionuclide release fractions during CRE and CRD accidents, the guidance in Revision 00, Footnote 11 appears non-conservative based on the empirical database of measure transient FGR shown above. However, its application may remain conservative. It depends on the conservatism build into the core physics models, the predicted energy deposition, and quantity and burnup distribution of fuel rods predicted to experience cladding failure. Assuming 10% release fractions on an overly conservative prediction of failed, low burnup fuel rods may still produce a conservative estimate of radiological consequences. However, the degree of conservatism is difficult to quantify, highly plant-specific, and may be cycle-specific.

Given the industry movement toward more accurate (and less inherently conservative) 3D core neutronic analytical models, it is recommended that future licensing applications involving changes in fuel design, fuel utilization, or changes in plant configuration or analytical models impacting CRE and CRD accident analyses, should include justification for the continued use of RG 1.183, Revision 00 CRE and CRDA release fractions (i.e., 10% iodines and nobles). Alternatively, Revision 01 should be used.

Attachment 5:

Best-Estimate Release Fractions for Environmental Assessment

Purpose:

The purpose of this attachment is to provide best-estimate, steady-state radionuclide release fractions for limiting fuel rods operating up to a rod average burnup of 68 GWd/MTU. These release fractions represent the amount of fission gas released from fuel pellets, residing in the fuel rod void volume, and available for release upon cladding failure. The potential impacts of extended rod power operation from 62 to 68 GWd/MTU (rod average) on release fractions and gap inventories are discussed. This attachment does not attempt to assess the environmental impact.

NUREG/CR-6703, *Environmental Effects of Extending Fuel Burnup Above 60 Gwd/MTU*, published in 2001, provided a best-estimate assessment of the environmental and economic impacts of extending peak-rod burnup above 60 GWd/MTU. Fission product inventories were calculated for burnup up to 75 GWd/MTU, and gap release fractions were calculated up to 62 GWd/MTU. At that time, evaluation of release fractions was limited to 62 GWd/MTU due to existing analytical models (i.e., validation of fuel performance code). Changes in analytical models since 2001 (e.g., ANS-5.4 standard) will promote significant differences in predicted release fractions (even for the same fuel rod design and power history). Any future assessment of environmental impact of extended burnup should not mix or compare NUREG/CR-6703 release fractions at 62 GWd/MTU and the release fractions presented below for 68 GWd/MTU. Further discussion is provided at the end of this attachment.

Best-Estimate Release Fractions:

Section 2.2 of NUREG/CR-6703 describes the calculation of core average and peak rod steady-state radionuclide release fractions using the FRAPCON-3 fuel thermal-mechanical fuel rod performance code. NUREG/CR-6703 states that best estimate release fractions are more appropriate for environmental assessments than bounding values and that release fractions calculated for this study do not include components to account for uncertainties in fuel fabrication and in the prediction of release fractions.

The main body of this memorandum provides upper-bound, steady-state radionuclide release fractions residing in the fuel rod void volume and available for release upon cladding failure. These gap fractions were calculated for fuel rod operation up to 68 GWd/MTU (rod average) using the FAST thermal-mechanical fuel rod performance code (which superseded FRAPCON-3). Because steady-state release fractions are sensitive to both fuel rod design and power operating history, these release fractions were based on the limiting fuel rod designs (BWR 10x10 and PWR 14x14) and the bounding power profile shown in Figure 1 of the main body. Table 5-1 provides best-estimate versions of these steady-state release fractions. Best-estimate values were obtained by removing the FAST fission gas release modeling uncertainty term (additive 0.0614) from the long-lived radionuclides and the ANS-5.4 standard release model uncertainty term (5x multiplier) from the short-lived radionuclides.

Impact of Extended Fuel Rod Average Burnup to 68 GWd/MTU:

As shown in Figures 3 and 4 of this memo, the maximum short-lived radionuclide (e.g., I-131 release fractions occur earlier in life where fuel rod power (i.e., rate of production) and fuel temperatures (i.e., rate of diffusion) are at maximum values. For this calculation, these maximum values occur at 20 GWd/MTU for BWR fuel rods and 35 GWd/MTU for PWR fuel rods. While plant-specific and cycle-specific fuel rod power histories vary widely, maximum release fractions and gap inventories of these short-lived radionuclides will occur earlier in life. Hence, extending allowable fuel rod burnup beyond 62 GWd/MTU will not impact the gap

inventory of short-lived radionuclides. However, to achieve a higher fuel rod burnup, utilities will likely implement changes in fuel pellet composition (e.g., uranium density) or ^{235}U enrichment (e.g., higher assembly average enrichment, maximum fuel pellet enrichment beyond 5.0 wt%). These changes may promote higher fuel rod power or an extension in the rod power profile (e.g., higher power at higher burnup). This more aggressive fuel operation would result in higher releases fractions for the short-lived radionuclides.

As shown in Figures 3 and 4 of this memo, the maximum long-lived radionuclide (e.g., Kr-85, Cs-137) release fractions occur at approximately 50 GWd/MTU. However, due to the progressive accumulation of long-lived fission gas in the fuel rod void volume, the maximum gap inventory (i.e., moles of gas) occurs at end-of-life. Extending allowable fuel rod burnup beyond 62 GWd/MTU may not impact the release fractions but will impact the end-of-life gap inventory. Similar to the short-lived radionuclides, fuel changes introduced to achieve the higher burnup, which promote more aggressive fuel operation, would also result in higher releases fractions for the long-lived radionuclides.

Comparison to NUREG/CR-6703: (For Information Only)

Changes in analytical models since 2001 (e.g., ANS-5.4 standard) will promote significant differences in predicted release fractions (even for the same fuel rod design and power history). Any future assessment of environmental impact of extended burnup should not mix or compare NUREG/CR-6703 release fractions at 62 GWd/MTU and the Table 1 release fractions at 68 GWd/MTU. The above information, based upon the most up-to-date analytical models, and consistent set of assumptions and inputs, may be used to assess the impact of extended fuel rod burnup.

For information only, Tables 5-2 and 5-3 provide PWR and BWR comparisons of the radionuclide release fractions from this calculation against the earlier NUREG/CR-6703 results. Examination of these tables reveals a dramatic increase in Kr-85 release fraction. This change is mostly due to the expanded fuel rod power operating envelope shown in Figure 1 which contains both higher rod power and extended burnup. While the FAST prediction for Cs-134 and Cs-137 are larger, changes in the treatment of cesium discussed in Attachment 1 reduce the impact of the expanded power operating domain. The significant reduction in short-lived radionuclides (e.g., I-131) is driven by the 2011 ANS-5.4 standard release model (which replaced the 1982 standard used in NUREG/CR-6703). These I-131 reductions are even more dramatic if you consider the expanded power operating domain.

One notable observation regarding NUREG/CR-6703 results shown in Tables 5-2 and 5-3 is that the trend in release fractions with burnup is different between the PWR and BWR calculations. For the BWR cases, the release fractions are larger at 43 GWd/MTU than those at 62 GWd/MTU. This trend is similar to the recent FAST calculations which showed peak values earlier in life. Whereas, the PWR cases show an increasing trend in release fraction with burnup. Even with consideration of the different fuel rod power histories (as shown in Figure A4-1), this PWR trend is difficult to explain.

Table 5-1: Best-Estimate Steady-State Radionuclide Release Fractions

Nuclide	Half-Life	PWR	BWR
Xe-133	5.243d	0.0115	0.0045
Xe-135	9.10h	0.0066	0.0026
Xe-135m	15.3m	- -	- -
Xe-137	3.82m	- -	- -
Xe-138	14.1m	- -	- -
Xe-139	39.7s	- -	- -
Kr-85	10.76y	0.3380	0.2560
Kr-85m	4.48h	0.0051	0.0020
Kr-87	1.27h	0.0026	0.0010
Kr-88	2.84h	0.0036	0.0014
Kr-89	3.15m	- -	- -
Kr-90	32.3s	- -	- -
I-131	8.04d	0.0121	0.0047
I-132	2.28h	0.0137	0.0053
I-133	20.8h	0.0073	0.0028
I-134	52.6m	- -	- -
I-135	6.57h	0.0052	0.0020
Cs-134	2.07y	0.1690	0.1280
Cs-137	30.1y	0.1690	0.1280

Table 5-2: Comparison of PWR Best-Estimate Steady-State Release Fractions

Radionuclide	NUREG/CR-6703 at 43 GWd/MTU	NUREG/CR-6703 at 62 GWd/MTU	FAST Maximum up to 68 GWd/MTU
Kr-85	0.043	0.079	0.3380
Kr-87	0.00088	0.004	0.0026
Kr-88	0.0013	0.0056	0.0036
I-131	0.028	0.068	0.0121
Xe-133	0.00869	0.028	0.0115
Xe-135	0.0023	0.011	0.0066
Cs-134	0.061	0.11	0.1690
Cs-137	0.061	0.11	0.1690

Table 5-3: Comparison of BWR Best-Estimate Steady-State Release Fractions

Radionuclide	NUREG/CR-6703 at 43 GWd/MTU	NUREG/CR-6703 at 62 GWd/MTU	FAST Maximum up to 68 GWd/MTU
Kr-85	0.10	0.079	0.2560
Kr-87	0.0051	0.004	0.0010
Kr-88	0.0072	0.0056	0.0014
I-131	0.087	0.068	0.0047
Xe-133	0.036	0.028	0.0045
Xe-135	0.014	0.011	0.0026
Cs-134	0.14	0.11	0.1280
Cs-137	0.14	0.11	0.1280