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Prepared for the U.S. Department of Energy
under Contract DE-AC05-76RL01830

Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard

CE Beyer
PM Clifford

June 2011

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Summary

This report provides the technical basis for a revision to the non-loss-of-coolant accident (LOCA) fission product gap inventories in NRC Regulatory Guide 1.183 (issued for public comment as DG-1199). Specifically, Table 3 **(**from Regulatory Guide 1.183) gap fractions were revised based upon an extended rod power operating history at a 95/95 probability/confidence level using the ANSI/ANS-5.4-2011 standard, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" (revision of withdrawn standard ANSI/ANS-5.4-1982, approved May 19, 2011). In addition, a transient gas release component was developed based on empirical data for the reactivity-initiated accident. Revision 1 of this report updates the non-LOCA fission product gap inventories and adds a new appendix (Appendix C) detailing an acceptable analytical procedure for calculating gap fractions on a design and operational specific basis. Changes in revision 1 of this report reflect public comments received on DG-1199.

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Acronyms and Abbreviations

ANS	American Nuclear Society
ANSI	American National Standards Institute
BIGR	Fast-pulse graphite reactor
BWR	boiling-water reactor
cal/gm	calorie/gram
EOL	end of life
FGR	Fission gas release
FRAPCON	fuel rod performance code
GWd/MTU	gigawatt day per metric ton uranium
IFA	instrumented fuel assembly
JAERI	Japan Atomic Energy Research Institute
kW/ft	kilowatt per foot
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LWR	light-water reactor
MOX	mixed oxide fuel
ms	millisecond
NRC	U.S. Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
PWR	pressurized-water reactor
R/B	release to birth
RIA	reactivity initiated accident
UTL	upper tolerance level
VVER	Soviet-designed pressurized water reactor

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1.0 Introduction

The U.S. Nuclear Regulatory Commission (NRC) has established guidance on fission product release fractions from the fuel rod void volume (e.g., fuel-to-cladding gap and plenum) of fuel rods that have been breached during postulated accidents for use in determining dose consequences from these events. The guidance for fission product gap release fractions for non-loss-of-coolant accident (LOCA) events has been defined in the following regulatory guides:

- Regulatory Guide 1.5, *Assumption Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors* (NRC 1971)
- Regulatory Guide 1.25, *Assumption Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling Water and Pressurized Water Reactors* (NRC 1972)
- Regulatory Guide 1.77, *Assumption Used for Evaluating the Potential Radiological Consequences of a Control Rod Ejection Accident for Pressurized Water Reactors* (NRC 1974)
- Regulatory Guide 1.195, *Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors* (NRC 2003).

In addition, Regulatory Guide 1.183 (NRC 2000) was issued to provide guidance for alternative source terms and acceptable radiological analysis assumptions for both LOCA and non-LOCA events. The objective of the work reported here is to provide a recommended replacement for Section 3.2 of Regulatory Guide 1.183 based on an extended rod power operating history. When approved, the release fractions recommended in this report will supersede those in the above regulatory guides for non-LOCA accidents.

During normal operation (leading up to the initiation of a postulated accident), gas atom diffusion controls the time-dependent inventory of fission products residing in the fuel-to-cladding gap that are available for release upon cladding breach. In addition to this steady-state component, fission products may be released from the fuel during certain postulated accidents that experience a rapid power (and associated fuel temperature) excursion or fuel fragmentation. Thus, for these types of events, this transient fission gas release component needs to be combined with existing steady-state gap inventories.

The reactivity-initiated accidents (RIAs), boiling-water reactor (BWR) control rod drop or pressurized-water reactor (PWR) control rod ejection, experience a prompt critical power excursion and associated fuel temperature spike. Regulatory guides 1.183 and 1.77 recommend a gap release fraction of 0.10 for both noble gases and iodines for these RIAs. Stable fission gas release data from test fuel rods subjected to simulated reactivity initiated transients have been measured at the CABRI, Nuclear Safety Research Reactor (NSRR), and Bystry Impulsny Grphitovy Reaktor (BIGR) test reactors. These measurements reveal that the RIA power excursion prompts transient gas release and that the previous guidance (gap fraction of 0.10) is non-conservative. Based on these empirical data, new gap release fractions will be recommended for RIA events.

No new data are available that suggest an additional transient fission gas release component is required for non-LOCA events that experience a less significant power excursion relative to RIAs. Similarly, no fission gas release measurements exist on the potential transient release associated with fragmentation of high-burnup fuel pellets resulting from an assembly drop or fuel rod balloon/burst.

Future testing may be necessary to address the potential transient releases resulting from grain boundary separation and fuel fragmentation.

This report is divided into two types of non-LOCA scenarios: 1) those that do not involve large fuel temperature increases and only a small number of assemblies are impacted such as fuel handling accident, single reactor coolant pump locked rotor, or steam line break accidents, and 2) those that experience a large fuel temperature increase such as reactivity initiated accidents.

A discussion of gap release fractions for accidents without large fuel temperature increases is provided in Section 2.0, gas release fractions for RIAs are provided in Section 3.0, and conclusions are provided in Section 4.0. The FRAPCON-3.3 fuel rod performance code (Lanning et al. 2005) input values for calculations in Section 2.0 are provided in Appendix A, and RIA data are provided in Appendix B. Appendix C provides an acceptable analytical procedure and guidance for developing gap inventories based upon unique fuel rod designs or power profiles.

2.0 Gap Release Fractions for Accidents without Large Fuel Temperature Increases

Gap release fractions in accidents that do not involve large fuel temperature increases (such that the gap fractions are due to gas atom diffusion during normal operation) were determined using the ANSI/ANS-5.4-2011 standard, “Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel” (revision of withdrawn standard ANSI/ANS-5.4-1982, approved May 19, 2011). As with the earlier version, the 2011 ANS 5.4 standard does not cover accidents that achieve high fuel temperatures as experienced during a RIA; therefore, this event needs to be addressed separately.

In Regulatory Guide 1.183, the Table 3 gap fractions were determined from analyses using FRAPCON-3.0 (Lanning et al. 1997) and the previous 1982 ANS 5.4 standard. The FRAPCON-3 code provides predicted fuel temperatures needed by the ANS 5.4 model to predict gap release fractions. The reason for updating the ANS 5.4 standard and Table 3 values in Regulatory Guide 1.183 is that the 1982 ANS 5.4 standard significantly over predicts the iodine release fractions. In addition, current operation in plants is at or close to the limit on rod average powers of 6.3 kilowatt per foot (kW/ft) at burnups above 54 gigawatt day per metric ton uranium (GWd/MTU) as recommended for the gap fractions in Regulatory Guide 1.183. It is not possible to eliminate a power and burnup limit on release fractions because fission product release is highly dependent on both of these parameters. Therefore, this proposed revision to Regulatory Guide 1.183 will increase the rod power and burnup envelope from the 6.3 kW/ft (rod average) at 54 GWd/MTU burnup. The revised power envelope in Figures A.1 (for PWRs) and A.3 (for BWRs) provided in Appendix A was created to be equal to or to bound maximum rod powers in current reactor cores. These peak rod power envelopes versus burnup were used to calculate the new release fractions using a fuel performance code and the 2011 ANS 5.4 standard.

The gap release values calculated in this report involve the use of the updated FRAPCON-3.3 fuel performance code (Lanning et al. 2005) with the 2011 ANS 5.4 standard equations implemented in the code to determine the release-to-birth (R/B) fractions of the volatile radioactive isotopes of noble gases (xenon and krypton), iodines, and cesiums. Those radioactive isotopes with half-lives of less than 60 days have utilized the release equations recommended by the 2011 ANS 5.4 standard; this includes all of the noble gases (except Kr-85) and the iodines. For those isotopes with very long half-lives such as Kr-85, Cs-134, and Cs-137 with half-lives of 10.76 years, 2.07 years, and 30.1 years, respectively, the steady-state fission gas release model in the FRAPCON-3.3 code was used as recommended by the 2011 ANS 5.4 standard. The calculated results include the 2011 ANS 5.4 recommended uncertainties for those short-lived isotopes (half-lives of less than 60 days) and the FRAPCON-3.3 release uncertainties, both at a 95/95 tolerance level.

2.1 Background of Revised ANS 5.4 Standard

The ANS 5.4 standard has been recently revised (approved by American National Standards Institute [ANSI] in May 2011) based on in-reactor release measurements of radioactive noble gas and iodine isotopes. The 1982 standard was based only on the release of stable noble gases with no in-reactor measurements of iodine release. As a result, the 1982 standard for iodine release is based on a very small amount of out-of-reactor (heating) release data that provided higher release values than observed from the recent in-reactor data. The in-reactor data come primarily from two Halden experimental rigs, instrumented fuel assembly (IFA)-504 and IFA-558 (Turnbull 2001, White and Turnbull 1998), while a few data from IFA-633 were also used for verification of the draft ANS 5.4 release model coefficients. These three IFAs involved helium gas flowing through each of the fuel rods in the assembly to sweep out the radioactive isotopes such that the quantity of radioactive gas could be measured by detectors outside of the reactor taking into account decay during transport from the rods to the detector.

The details of how the in-reactor isotopic activity are measured and how they are converted to release data, R/B, for the radioactive isotopes of Kr-85m, Kr- 87, Kr-88 Kr-89, Kr-90, Xe-133, Xe-135, Xe-135m, Xe-137, Xe-138, Xe-139, I-131, I-133, and I-135 are provided in a Halden document (White and Turnbull 1998) for the revised standard. The release data (R/B) for the Kr-85m isotope were the principal release data used to verify the revised ANS 5.4 model, which consisted of a total of 312 Kr-85m R/B data points from IFA-504, IFA-558, and IFA-633 (131, 175, and six data points, respectively). The Kr-85m release data were used because this isotope is relatively long-lived (4.48 hours) with a large activity peak that allowed it to be counted accurately with detectors, and a large quantity of these data existed. There were longer-lived isotopes measured, such as Xe-133 (5.24 days) and Xe-135 (9.1 hours), but the measurement accuracies for these isotopes were significantly less than for Kr-85m. The revised standard has also been compared and verified against I-131 R/B data from IFA-504 and IFA-558 but these data are of much lower quantity (only 17 R/B data points) due to difficulty in measuring this isotope. The I-131 release data are very important for verifying the revised standard because it dominates most dose consequence analyses due to its large contribution of dose to the thyroid. The I-131 data also verify that the use of the Kr-85m data for establishing coefficients of the model and determining predictive uncertainties are reasonable. Further background information on the development of the standard based on the sweep gas data from Halden has been published in a background document (Turnbull and Beyer 2010).

The range of the data in terms of rod average power and burnup is given in Figure 2.1. The rod power and burnup peaking factors for these rods are relatively low, between 1.06 to 1.09, compared to commercial fuel rod peaking factors of 1.1 to 1.3 for PWRs and 1.1 to 1.4 for BWRs.

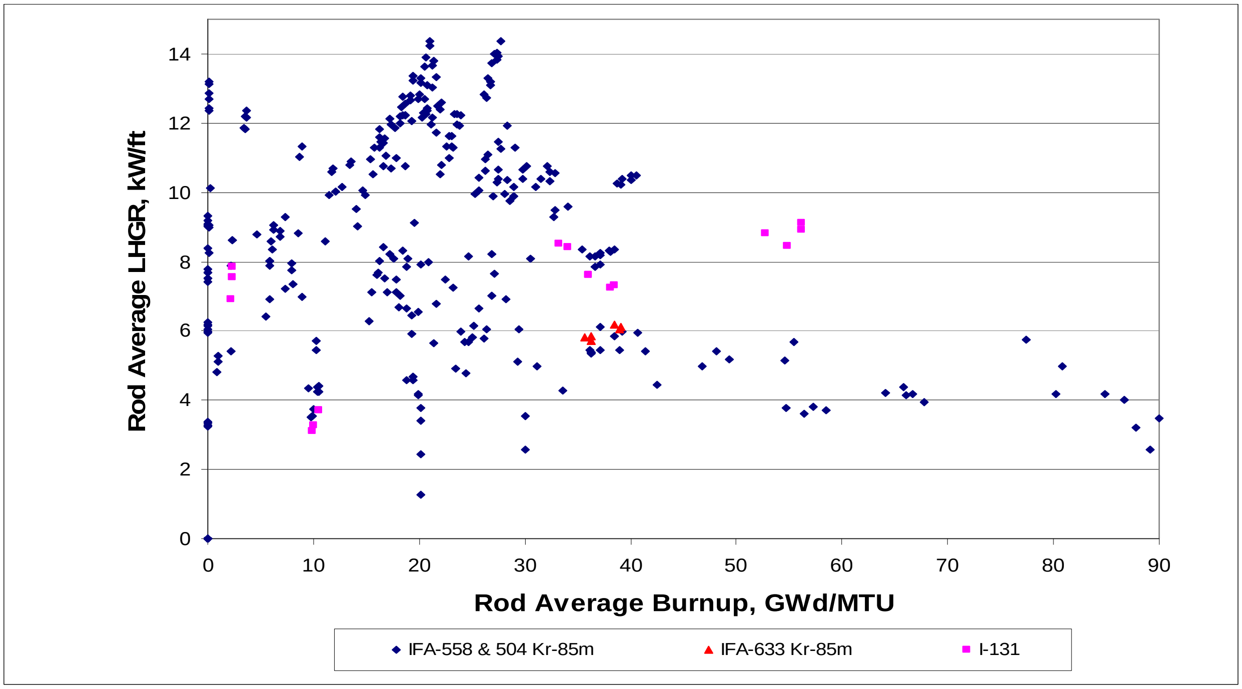

Figure 2.1. Operating Range of Data Used to Develop and Verify the ANS 5.4 Model (Data From Instrumented Sweep Gas Experiments in the Halden Reactor)

2.1.1 Uncertainties in Revised ANS 5.4 Standard and Comparisons to Release Data

The predicted release fractions provided in this report, using the 2011 ANS 5.4 standard, correspond to an upper bound of the Kr-85m data at a 95/95 upper tolerance level (UTL). This is achieved by multiplying the ANS 5.4 best estimate predictions by a factor of 5 as recommended by this standard. The factor of 5 multiplier is derived assuming a non-normal distribution of the Kr-85m data. This factor of 5 multiplier on the best estimate predictions results in a conservative prediction of the I-131 release data as demonstrated in Figure 2.2 at a level greater than 95/95 tolerance level for these limited data. This is due to the fact that the multiplier is based on the prediction of the Kr-85m release data and not the I-131 release data. The factor of 5 is approximately a factor of 2 higher than what would be computed from the revised ANS 5.4 best estimate model predictions of the I-131 data. The extra conservatism was maintained for I-131, relative to that calculated for Kr-85m data, because some members of the standards sub-committee felt that the I-131 release measurements might be biased low by up to a factor of 2; therefore, a factor of 5 uncertainty is assumed for the I-131 predictions. The best-estimate predictions of the I-131 data are provided in Figure 2.3 to demonstrate the accuracy of the model predictions.

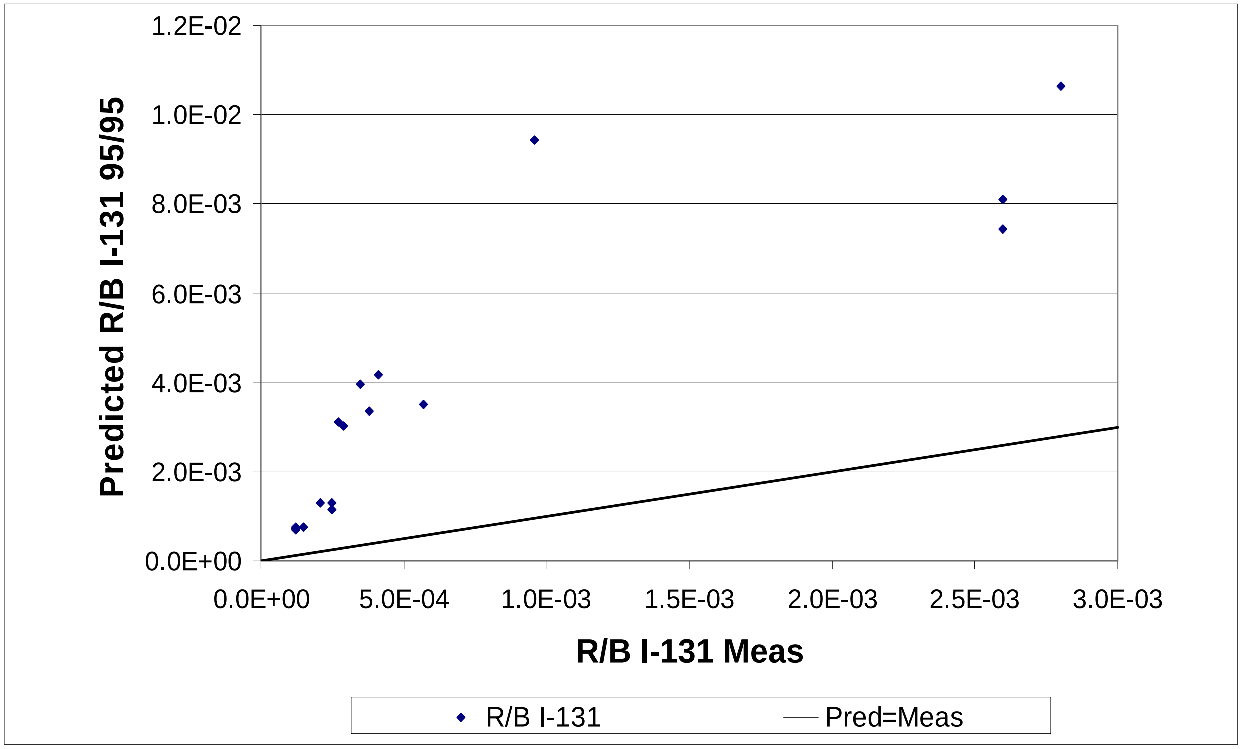

Figure 2.2. Draft ANS 5.4 Standard Model Comparison of Prediction to Measured I-131 Release Data Assuming a Factor of 5 Multiplier on Prediction Results in a 95/95 UTL

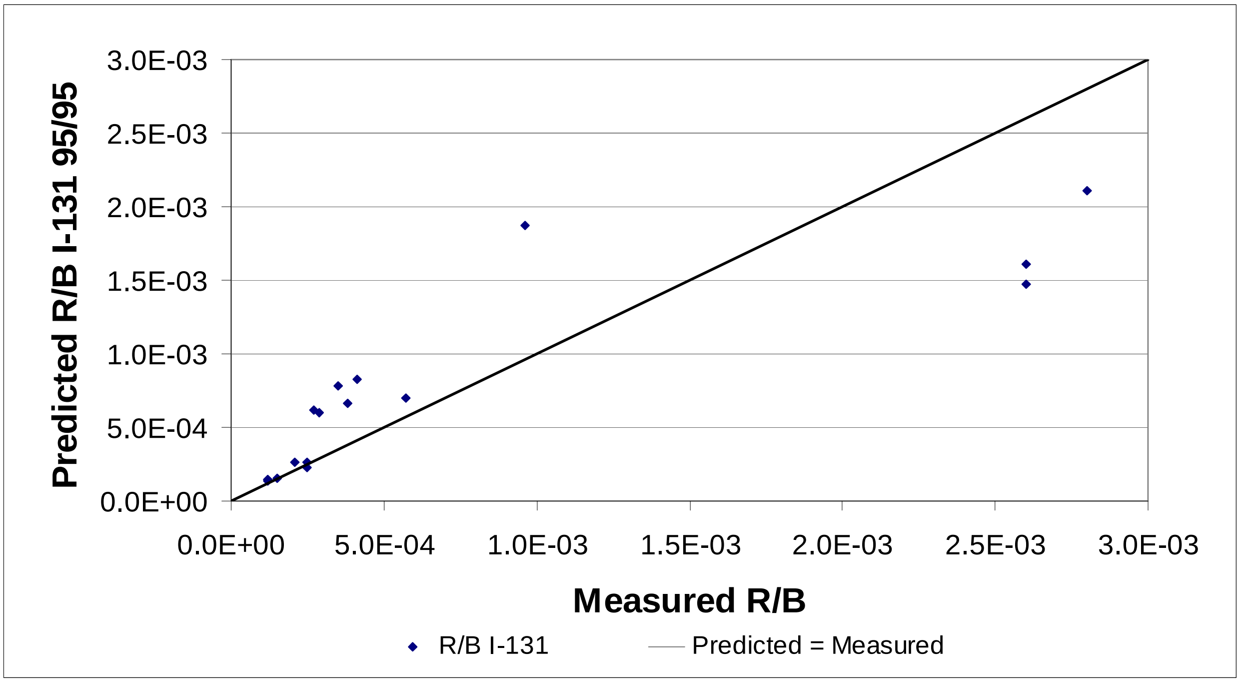

Figure 2.3. ANS 5.4 Model Best Estimate Prediction of R/B I-131 Data from Halden Experiments IFA-504 and IFA-558

2.1.2 Application of Revised ANS 5.4 Using a Fuel Performance Code

The application of the 2011 ANS 5.4 standard for predicting the release of radioactive fission products requires the use of a fuel performance code that predicts fuel temperatures accurately up to the burnup and peak rod power levels achieved in current light-water reactor (LWR) cores (e.g., a peak pellet burnup of 70 GWd/MTU).

The ANS 5.4 standard also requires the use of a fuel performance code fission gas release model for the stable noble gases to predict the release of the long-lived isotopes of Kr-85, Cs-134, and Cs-137 and their associated uncertainties. This fuel performance code is required to accurately predict the release of the stable noble gases up to the burnup and peak rod power levels achieved in current LWR cores (e.g., a peak pellet burnup of 70 GWd/MTU). The 95/95 upper bound gap release prediction of these long-lived isotopes is based on the standard deviation of the stable noble fission gas fractions and the number of data points used to determine the standard deviation. The release data used to verify the code need to be based on fuel rods that operated near the peak power in an LWR core. The gap release fractions for Cs-134 and Cs-137 take into account the diffusion coefficient differences for the cesium isotopes relative to noble gases (see Section 2.2.4 below).

For this particular application, the FRAPCON-3.3 fuel performance code was selected. This code has been verified up to 70 GWd/MTU and beyond for both fuel centerline temperature and stable noble gas predictions. As noted in the ANS 5.4 standard, other fuel performance codes may be used to estimate the fuel temperatures and gap release fractions for the long-lived isotopes of Kr-85, Cs-134, and Cs-137 if they have been verified against similar peak power rods up to peak pellet burnups of 70 GWd/MTU for LWR cores and have been approved by the NRC for licensing applications.

2.2 FRAPCON-3.3/ANS 5.4 Calculated Gap Release Fractions

Appendix C details the analytical procedure used in the development of the generic fission product gap inventories following the 2011 ANS 5.4 standard. For future applicants, this appendix provides an acceptable analytical technique for calculating non-LOCA fission product gap inventories based upon specific fuel rod designs or more realistic fuel rod power histories. Example calculations are included to clarify the analytical guidance and illustrate the potential gains in predicted gap fractions using a less aggressive fuel rod power history than used in Appendix A for the generic inventories.

The analyses using the ANS 5.4 standard are divided into bounding PWR and BWR results, and sensitivity analyses have been performed to determine which of the current fuel designs are the most limiting in terms of release of fission products. The bounding PWR design was found to be a 14 x 14 fuel design and the bounding BWR design was a 9 x 9 fuel design. It should be noted that an 8 x 8 fuel design is slightly more bounding than the 9 x 9 design but U.S. reactors currently have very little or no 8 x 8 fuel in their cores, and the 8 x 8 design will not be used in future cores within the United States.

Examination of the fuel temperature and fission gas release results from an earlier NRC report, NUREG-1754 (O'Donnell et al. 2001), of analyses for different fuel designs also demonstrated that the 14 x 14 fuel design was most limiting of the PWR designs (14 x 14, 15 x 15, 16 x 16, and 17 x 17). This report also demonstrated that the 8 x 8 BWR fuel design was slightly more limiting in terms of temperatures and fission gas release than the 9 x 9 fuel design, and that the 10 x 10 design was the least limiting of BWR fuel designs.

2.2.1 PWR Analyses

The input parameters assumed for the 14 x 14 fuel design are provided in Table A.1 of Appendix A. The maximum power history in terms of rod average power assumed for this analysis is very important. This maximum (bounding) rod average power history is based on bounding the thermal-mechanical design linear heat generation rate (LHGR) limits in terms of peak nodal power (often referred to as the F_Q limit) for current cores licensed within the United States divided by the assumed peak-to-average of the axial power shapes given in Figure A.2. Rod average power versus burnup is used for this analysis because fission product release for a fuel rod is more strongly dependent on the rod average power than peak nodal power. It is further noted that a flatter axial power shape is more conservative in terms of release than a more peaked shape for a given rod average power.

The power level in this analysis remains constant at the thermal-mechanical design limit up to a given burnup level and then decreases with burnup. The decrease with burnup occurs because burnup depletes the U-235 to a level at which the remaining fissile material cannot sustain power operation at the peak F_Q limit at higher burnups for current fuel core management schemes without other rods in the core exceeding this limit. Therefore, the assumed bounding power history, given in Figure A.1 of Appendix A, is conservative at any given burnup level for current cores. It should be noted that running at the bounding rod average power history in Figure A.1 for today's cycle lengths would exceed current burnup limits in the United States.

In order for the power histories to be more realistic, but still bounding in terms of possible power operation, seven different power histories were considered with each running at 90% of the peak node history shown in Figure A.1, with the exception of running at the bounding rod average LHGR for approximately 9 to 10 GWd/MTU burnup (rod average) for seven different burnup intervals up to 65 GWd/MTU. The PWR limiting rod average history for stables and radioactives is also noted in Figure A.1. The assumed PWR axial power shapes are given in Figure A.2. The upper bound PWR power histories in Figure A.1 are below the highest power Halden data in Figure A.1 to 56 GWd/MTU (rod average) burnup (or peak node burnup of ~ 60 GWd/MTU) with the data having a peak-to-average power of between 1.06 to 1.09. The data have a lower power rating than the PWR peak node history assumed in Figure A.1 above 56 GWd/MTU (rod average); therefore, there is a small extrapolation in burnup. This small extrapolation is acceptable due to the large conservatism placed on the ANS 5.4 release predictions, particularly the I-131 predictions that dominate the thyroid dose limits that are most limiting.

The R/B values calculated for the long-lived isotopes (half-life greater than 60 days), and the short-lived noble gas and halogen isotopes for each of the seven power histories are provided in Tables 2.1, 2.2, and 2.3, respectively. The maximum calculated release fractions, R/B, from the seven fuel rod power histories considered in Tables 2.1, 2.2, and 2.3 are provided in Table 2.4 for all of the radioactive isotopes of noble gases, iodines, and cesiums with half-lives greater than 1 hour.

Table 2.1. PWR Fuel Rod Gap Release Fractions, R/B, From Seven Different Power Histories Operating at LHGR Limits at Seven Different Burnup Intervals (Long-Lived (Based on Stable) Release Fractions)

Case	FRAPCON EOL FGR (fraction)	FRAPCON 95/95 UTL k * sigma	Kr-85 95/95 UTL	Cs-134 95/95 UTL	Cs-137 95/95 UTL
PNNL-18212 Rev.0	0.2770	0.0661	0.3431	0.4578	0.4578
PWR History 1	0.2550	0.0661	0.3211	0.4267	0.4267
PWR History 2	0.2560	0.0661	0.3221	0.4281	0.4281
PWR History 3	0.2580	0.0661	0.3241	0.4310	0.4310
PWR History 4	0.2670	0.0661	0.3331	0.4437	0.4437
PWR History 5	0.2890	0.0661	0.3551	0.4748	0.4748
PWR History 6	0.2910	0.0661	0.3571	0.4776	0.4776
PWR History 7	0.2630	0.0661	0.3291	0.4380	0.4380

Table 2.2. PWR Fuel Rod Gap Release Fractions, R/B, From Seven Different Power Histories Operating at LHGR Limits at Seven Different Burnup Intervals (Short-Lived Volatile R/B Ratios for Noble Gases)

Nobles (Kr, Xe) - Release/Birth								
	Kr-85m		Kr-87		Kr-88		Xe-133	Xe-135
Case	FRAPCON	UTL	FRAPCON	UTL	FRAPCON	UTL	UTL	UTL
PNNL-18212 Rev. 0	0.0061	0.0307	0.0032	0.0160	0.0044	0.0218	0.0697	0.0399
PWR History 1	0.0035	0.0175	0.0018	0.0091	0.0025	0.0124	0.0397	0.0227
PWRHistory 2	0.0035	0.0175	0.0018	0.0091	0.0025	0.0124	0.0398	0.0228
PWR History 3	0.0049	0.0244	0.0025	0.0127	0.0035	0.0173	0.0554	0.0317
PWR History 4	0.0064	0.0321	0.0033	0.0167	0.0046	0.0228	0.0729	0.0417
PWR History 5	0.0054	0.0270	0.0028	0.0141	0.0038	0.0192	0.0614	0.0351
PWR History 6	0.0035	0.0174	0.0018	0.0091	0.0025	0.0124	0.0396	0.0226
PWR History 7	0.0035	0.0174	0.0018	0.0091	0.0025	0.0124	0.0396	0.0226

Table 2.3. PWR Fuel Rod Gap Release Fractions, R/B, From Seven Different Power Histories Operating at LHGR Limits at Seven Different Burnup Intervals (Short-Lived Volatile R/B Ratios for Halogens)

Halogen (I) - Release/Birth				
	I-131	I-132	I-133	I-135
Case	UTL	UTL	UTL	UTL
PNNL-18212 Rev.0	0.0734	0.0828	0.0441	0.0315
PWR History 1	0.0418	0.0471	0.0251	0.0180
PWR History 2	0.0419	0.0473	0.0252	0.0180

Table 2.3. (contd)

Halogen (I) - Release/Birth				
	I-131	I-132	I-133	I-135
Case	UTL	UTL	UTL	UTL
PWR History 3	0.0583	0.0658	0.0350	0.0251
PWR History 4	0.0768	0.0866	0.0461	0.0330
PWR History 5	0.0647	0.0729	0.0389	0.0278
PWR History 6	0.0417	0.0470	0.0250	0.0179
PWR History 7	0.0417	0.0470	0.0250	0.0179

Table 2.4. PWR Generic Fuel Rod Gap Release Fractions, R/B, To Gap Maximum from Tables 2.1, 2.2, and 2.3 (Based on Bounding 14x14 UO_2 Power History Up to 65 GWd/MTU Rod Average Burnup)

Nuclide	Half-Life	Maximum (95/95)
Xe-133	5.243d	0.0729
Xe-135	9.10h	0.0417
Xe-135m	15.3m	- -
Xe-137	3.82m	- -
Xe-138	14.1m	- -
Xe-139	39.7s	- -
Kr-85	10.76y	0.3571
Kr-85m	4.48h	0.0321
Kr-87	1.27h	0.0167
Kr-88	2.84h	0.0228
Kr-89	3.15m	- -
Kr-90	32.3s	- -
I-131	8.04d	0.0768
I-132	2.28h	0.0866
I-133	20.8h	0.0461
I-134	52.6m	- -
I-135	6.57h	0.0330
Cs-134	2.07y	0.4776
Cs-137	30.1y	0.4776

2.2.2 BWR Analyses

The input parameters assumed for the 9 x 9 fuel design are identified in Table A.1 of Appendix A. The maximum power history in terms of rod average power assumed for this analysis is very important. This maximum (bounding) rod average power history in Figure A.3 is based on bounding the LHGR technical specification limits at the peak node for current BWR cores under normal operation licensed within the United States divided by the assumed peak-to-average of the axial power shapes given in Figure A.4. Rod average power versus burnup is used for this analysis because fission product release for a fuel rod is more strongly dependent on the rod average power than peak nodal power. It is further noted that a flatter axial power shape is more conservative than a more peaked shape for a given rod average power.

The rod average power level remains constant at the LHGR limit up to a given burnup level and then decreases with burnup. The decreasing power level beyond a given burnup level for BWRs is generally due to the need to meet rod internal gas pressure limits from the no cladding liftoff specified in Section 4.2 of the Standard Review Plan (NRC 2007), as well as due to fissile depletion with burnup. Therefore, the assumed bounding power history given in Figure A.3 of Appendix A is conservative at any given burnup level for current BWR cores. It should be noted that running at the bounding rod average power history in Figure A.3 for today's cycle lengths would exceed current burnup limits.

In order for the power histories to be more realistic, but still bounding in terms of possible power operation, seven different power histories were considered with each running at 90% of the history in Figure A.3 with the exception of running at the LHGR limit for approximately 9 to 10 GWd/MTU burnup (rod average) for different burnup intervals as demonstrated in Figure A.3 in Appendix A. The assumed BWR axial power shapes are given in Figure A.4. The upper bound BWR rod average power histories are near the highest power Halden data in Figure 2.1 up to 56 GWd/MTU (rod average) burnup (or peak node burnup of ~ 60 GWd/MTU) with the data having a peak-to-average power of between 1.06 to 1.09. Therefore, the data have a lower power peak rating than the BWR rod average history assumed in Figure A.3 above 56 GWd/MTU; therefore, there is a small extrapolation in burnup. This small extrapolation is acceptable due to the large conservatism placed on the ANS 5.4 release predictions, particularly the I-131 predictions that dominate the thyroid dose limits that are most limiting

The R/B values calculated for the long-lived isotopes (half-life greater than 60 days), and the short-lived noble gas and halogen isotopes for each of the seven BWR power histories are provided in Tables 2.5, 2.6, and 2.7, respectively. The maximum calculated release fractions, R/B, from the seven fuel rod power histories in Table 2.5, 2.6, and 2.7 are provided in Table 2.8 for all of the radioactive isotopes of noble gases, iodines, and cesiums with half-lives greater than 1 hour.

Table 2.5. BWR Fuel Rod Gap Release Fractions, R/B, From Seven Different Power Histories Operating at LHGR Limits at Seven Different Burnup Intervals (Long-Lived Stable Release Fractions)

Case	FRAPCON EOL FGR (fraction)	FRAPCON 95/95 UTL k * sigma	Kr-85 95/95 UTL	Cs-134 95/95 UTL	Cs-137 95/95 UTL
PNNL-18212 Rev.0	0.1790	0.0661	0.2451	0.3192	0.3192
BWR History 1	0.2510	0.0661	0.3171	0.4211	0.4211
BWR History 2	0.2540	0.0661	0.3201	0.4253	0.4253
BWR History 3	0.2570	0.0661	0.3231	0.4296	0.4296
BWR History 4	0.2660	0.0661	0.3321	0.4423	0.4423
BWR History 5	0.2930	0.0661	0.3591	0.4805	0.4805
BWR History 6	0.3060	0.0661	0.3721	0.4988	0.4988
BWR History 7	0.2600	0.0661	0.3261	0.4338	0.4338

Table 2.6. BWR Fuel Rod Gap Release Fractions, R/B, From Seven Different Power Histories Operating at LHGR Limits at Seven Different Burnup Intervals (Short-Lived Volatile R/B Ratios for Noble Gases)

Nobles (Kr, Xe) - Release/Birth								
	Kr-85m		Kr-87		Kr-88		Xe-133	Xe-135
Case	FRAPCON	UTL	FRAPCON	UTL	FRAPCON	UTL	UTL	UTL
PNNL-18212 Rev.0	0.0030	0.0151	0.0016	0.0079	0.0021	0.0107	0.0344	0.0196
BWR History 1	0.0023	0.0115	0.0012	0.0060	0.0016	0.0082	0.0262	0.0150
BWR History 2	0.0039	0.0196	0.0020	0.0102	0.0028	0.0139	0.0445	0.0254
BWR History 3	0.0043	0.0216	0.0023	0.0113	0.0031	0.0153	0.0490	0.0280
BWR History 4	0.0036	0.0179	0.0019	0.0093	0.0025	0.0127	0.0406	0.0232
BWR History 5	0.0030	0.0152	0.0016	0.0080	0.0022	0.0108	0.0346	0.0198
BWR History 6	0.0023	0.0115	0.0012	0.0060	0.0016	0.0081	0.0261	0.0149
BWR History 7	0.0023	0.0115	0.0012	0.0060	0.0016	0.0081	0.0261	0.0149

Table 2.7. BWR Fuel Rod Gap Release Fractions, R/B, From Seven Different Power Histories Operating at LHGR Limits at Seven Different Burnup Intervals (Short-Lived Volatile R/B Ratios for Halogens)

Halogen (I) - Release/Birth				
	I-131	I-132	I-133	I-135
Case	UTL	UTL	UTL	UTL
PNNL-18212 Rev.0	0.0362	0.0408	0.0217	0.0155
BWR History 1	0.0275	0.0311	0.0165	0.0118
BWR History 2	0.0468	0.0528	0.0281	0.0201
BWR History 3	0.0516	0.0582	0.0310	0.0222
BWR History 4	0.0427	0.0482	0.0257	0.0184
BWR History 5	0.0364	0.0411	0.0219	0.0156
BWR History 6	0.0274	0.0309	0.0165	0.0118
BWR History 7	0.0274	0.0309	0.0165	0.0118

Table 2.8. BWR Generic Fuel Rod Gap Release Fractions, R/B, To Gap Maximum from Tables 2.5, 2.6, and 2.7 (Based on Bounding 9x9 UO_2 Power History Up to 65 GWd/MTU Rod Average Burnup)

Nuclide	Half-Life	Maximum (95/95)
Xe-133	5.243d	0.0490
Xe-135	9.10h	0.0280
Xe-135m	15.3m	- -
Xe-137	3.82m	- -
Xe-138	14.1m	- -

Table 2.8. (contd)

Nuclide	Half-Life	Maximum (95/95)
Xe-139	39.7s	- -
Kr-85	10.76y	0.3721
Kr-85m	4.48h	0.0216
Kr-87	1.27h	0.0113
Kr-88	2.84h	0.0153
Kr-89	3.15m	- -
Kr-90	32.3s	- -
I-131	8.04d	0.0516
I-132	2.28h	0.0582
I-133	20.8h	0.0310
I-134	52.6m	- -
I-135	6.57h	0.0222
Cs-134	2.07y	0.4988
Cs-137	30.1y	0.4988

2.2.3 Assumed Analysis Uncertainties

The predicted release fractions calculated in this report are based on an upper bound tolerance of 95 percent probability with 95 percent confidence that the fractions are bounding. For the short-lived isotopes, the 95/95 release fractions calculated with the 2011 ANS 5.4 standard are determined by multiplying the best estimate predictions by a factor of 5 as recommended by the revised standard. The FRAPCON-3.3 95/95 predicted release fractions are based on a standard deviation of 0.028 (absolute) from the prediction of the release of stable noble gases from 23 fuel rods (Lanning et al. 2005) and assuming a normal distribution.

2.2.4 Discussion and Recommendations for Gap Release Fractions

Examining the maximum release values in Tables 2.4 and 2.8 for PWR 14 x 14 and BWR 9 x 9 fuel designs demonstrates that the PWR design is the more limiting (i.e., has the higher release fractions) for the short-lived isotopes but the BWR design is slightly more limiting for the long-lived isotopes. The short-lived is more limiting for a PWR because the rod powers remain higher up to higher burnups than for the BWR, which decreases in power at a lower burnup. The maximum release values from each of these two designs are compiled in Table 2.9, these values are rounded up to the nearest 0.01 fraction. The previous release fractions in Regulatory Guide 1.183 Table 3 are also given in Table 2.9 of this report for reference and show that the new release values for most of the isotopes are significantly higher. This is because today's fuel rod powers and burnups are significantly more aggressive than those of 20 years ago. The reason that release for I-131 isotope is not higher for the current analysis is because there was significant conservatism in the 1982 ANS 5.4 standard for this isotope that was used as the basis for Table 3 of Regulatory Guide 1.183; this conservatism for this isotope was reduced in the 2011 ANS 5.4 standard.

Table 2.9. PWR and BWR Fuel Rod Peak Gap Release Fractions (Based on Peak Values from Tables 2.4 and 2.8[(a)]

	Gap Release Fractions - 95/95 UTL			
Isotope	Calculated PWR 14x14 Design	Calculated BWR 9x9 Design	Maximum	Current RG 1.183 Table 3
Kr-85	0.357	0.372	0.38	0.10
I-131	0.077	0.052	0.08	0.08
I-132	0.087	0.058	0.09	0.05
Other Nobles	0.073	0.049	0.08	0.05
Other Halogens	0.046	0.031	0.05	0.05
Alkali Metals	0.478	0.499	0.50	0.12

a. Gap fractions for non-LOCA events with exception of RIA events

Further examination of Tables 2.1-2.3 and 2.5-2.7 demonstrates that the peak R/B release for the short-lived isotopes (less than 60-day half-life) always occurs at the highest burnup at which the highest rod power is achieved. This is because release increases with both increasing temperature and burnup, with temperature having the strongest impact. The peak release for all the short-lived isotopes is based on only the current temperature, production rate, and burnup. This is because the release of the short-lived isotopes reaches an equilibrium release quickly (approximately three half-lives), which is proportional to the current local fuel temperature, production rate (power), and burnup. Because the short-lived isotope release is more strongly dependent on fuel temperatures and power than burnup, the release decreases as power decreases such that the peak release in terms of number of moles or curies occurs early in life. The decrease in R/B when power decreases is illustrated in Figure 2.4 where the predicted R/B for the I-131 isotope is plotted versus burnup for the PWR 14 x 14 Power History #4 shown in Figure A.1 of Appendix A. The increase in R/B with constant power as burnup increases for burnups below 35 GWd/MTU for the PWR (Figure 2.4) and below 20 GWd/MTU for the BWR is caused by an increase in fuel temperatures due to the degradation in fuel thermal conductivity with burnup and an increase in diffusion coefficient with burnup. The increase of both temperature and burnup contributes to increased saturation of the grain boundaries resulting in a compounding increase in release with increasing burnup at constant power.

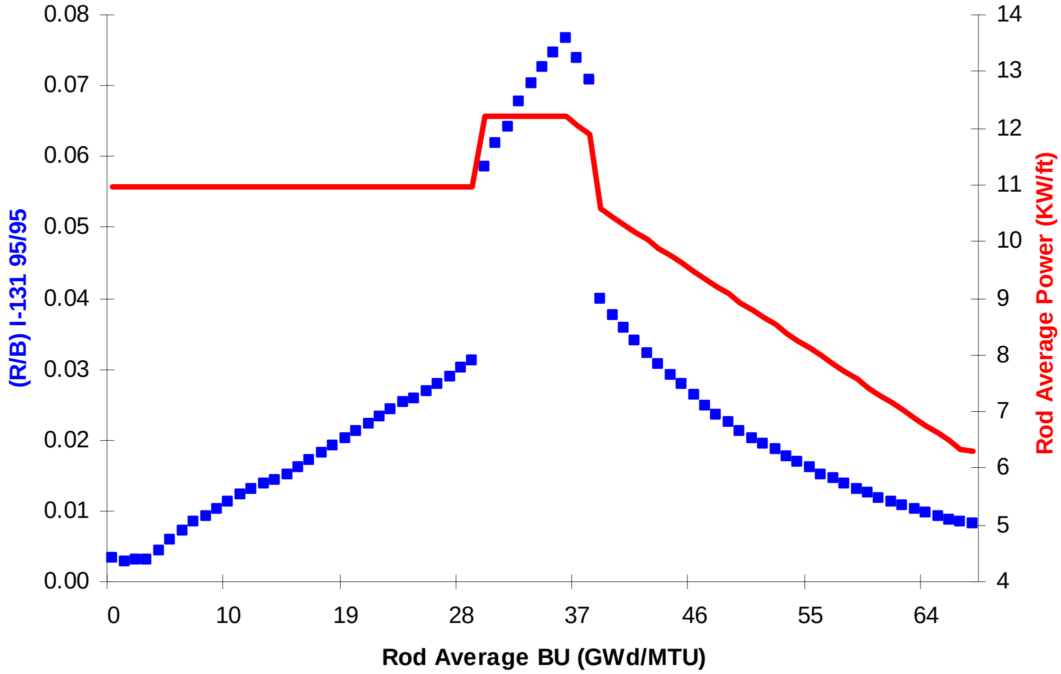

Figure 2.4. Predicted R/B Release for I-131 as a Function of Burnup Illustrating Peak Release Occurs at the Maximum Burnup at which Maximum Rod Power is Achieved (PWR History #4)

The peak release in terms of release fraction, for the long-lived isotopes of Kr-85, Cs-134, and Cs-137 occurs near the end-of-life (EOL) for Power History #6 for both the PWR and BWR power histories assumed for this analysis, as shown in Figure 2.5. The release fraction, F, for long-lived isotopes is defined as the ratio of total atoms released to the gap to the atoms produced.

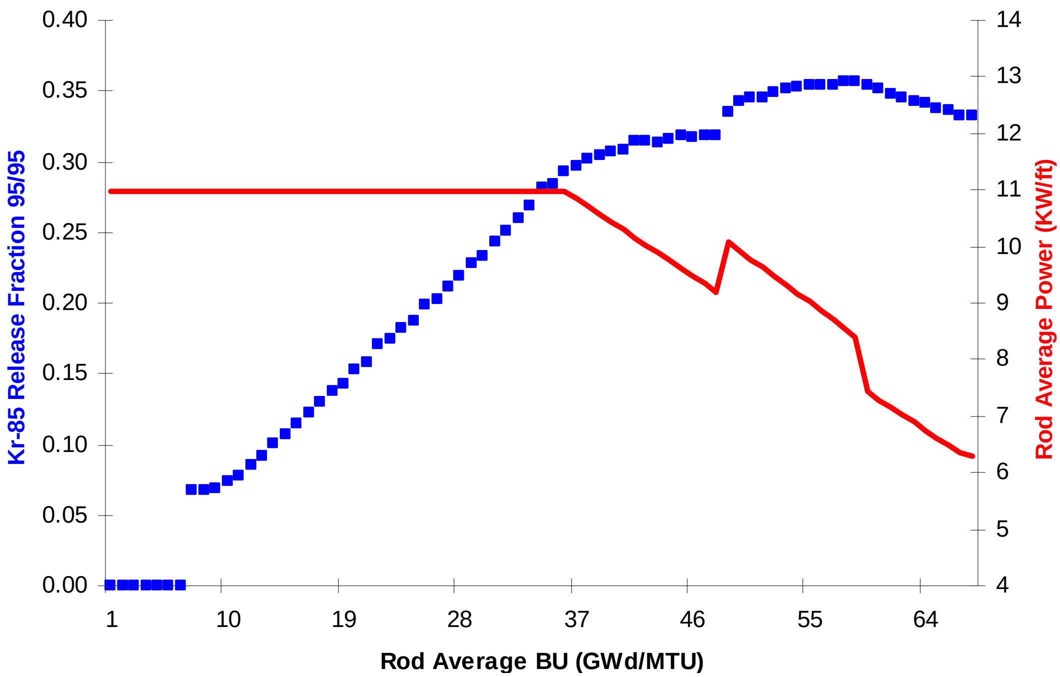

Figure 2.5. Predicted Release Fraction for Kr-85 (Long-Lived) Isotope as a Function of Burnup Illustrating Peak Release Occurs Near EOL (PWR History #6)

As recommended by the 2011 ANS 5.4 standard the cesium diffusion should be assumed to be a factor of 2 higher than for the noble gases (xenon and krypton), which has been carried over from the 1982 ANS 5.4 standard because the current ANS 5.4 Working Group was not aware of any new quantitative cesium release data under normal LWR operating conditions.

The maximum gap release values in Table 2.9 are intended to replace the non-LOCA release fractions in Table 3 of Regulatory Guide 1.183 and the recommended gap release fractions in Regulatory Guides 1.5 and 1.25. The methodology described in Appendix C can be used to determine gap release fractions for a specific fuel design that may have less limiting power operation. The assumed fuel rod average power history versus burnup is the major driving force for the calculated gap release fractions such that a fuel design with a lower power history than used in this analysis will have significantly lower release fractions. However, as noted in Appendix C, uncertainties in rod powers and possible deviations in plant maneuvering from those assumed in the depletion calculations should be accounted for in assumed power histories.

2.2.5 Fuel Rod Gap Activity

The release rate, R, and birth rate, B, both have units of atoms/cm^3-s with both including the decay of the specific isotope. Due to decay, an equilibrium R/B rate is established within three half-lives of an isotope (e.g., for I-131 with an 8-day half-life, equilibrium is achieved in 24 days) as long as the fuel temperature remains relatively constant during this time period. All of the isotopes with half-lives greater than 1 year are conservatively treated as stable isotopes for ease of analysis as recommended by both the 1982 and 2011 ANS 5.4 standard. The isotopes in Tables 2.1 through 2.8 with half-lives greater than 1 day are those that dominate the radiological consequences due to external and internal dose, while generally the iodine releases provide the smallest margins to dose limits due to their dose to the thyroid. Those isotopes with half-lives of less than 1 day generally do not contribute much to dose due to their quick decay; as a result, these isotopes are not presented in these tables.

Total curies of release to the gap for both the short-lived isotopes and the long-lived isotopes can be calculated by multiplying the total inventory of an isotope by either R/B (short-lived) or fractional release (long-lived). An NRC-approved code such as ORIGIN-S should be used to determine the maximum inventory for both the short-lived and long-lived isotopes. For determining the inventory of short-lived and long-lived isotopes, the following guidance is offered:

- The short-lived isotope inventory and release is always greatest at the highest burnup at which the highest rod power (fission rate) is achieved and should be assumed to have reached equilibrium (steady-state) for non-LOCA events. The highest rod power should be based on the technical specification or COLR integral radial peaking factor for non-LOCA events. Treating the R/B as a release fraction is acceptable because under equilibrium conditions (as assumed) neither R (release rate) nor B (birth rate) are changing with time. This means that the moles or curies in the gap and in the fuel are not changing for the time interval considered.
- The maximum inventory of the long-lived isotopes such as Kr-85, Cs-134, and Cs-137 is always greatest near or at end-of-life (i.e., highest exposure).

3.0 Gap Release Fractions for Reactivity Initiated Accident

The total fission-product gap fraction available for release following any RIA should include the steady-state gap inventory (present from normal operation before the RIA event) plus any fission gas released during the RIA event. Conservative steady-state gap fractions from normal operation are provided in Table 2.9 of this report and the equations provided in this section will provide the conservative gap release fraction during the RIA event that need to be added to the steady-state gap release fractions from Table 2.9.

Regulatory Guides 1.183 and 1.77 recommend a gap release fraction of 0.10 for both noble gases and iodines. The stable noble gas release data from simulated RIA tests on PWR, BWR, and VVER test rods (shortened rod segments from actual commercially irradiated rods) are compiled in Appendix B from tests in CABRI (Lemoine et al. 2000), NSRR (Fuketa et al. 1997, Fuketa et al. 2005, Nakamura et al. 2000, and Nakamura et al. 2002), and BIGR (Yegorova et al. 2006) test reactors. These release data for stable isotopes are plotted in Figure 3.1 as a function of enthalpy increase. The release of stable noble gases in this figure applies to the long-lived Kr-85 isotope and demonstrates that above an enthalpy increase of ~ 40 cal/gm the upper bound Kr-85 release exceeds the 0.10 recommended release fraction for RIA in the regulatory guides. In addition, the release in Figure 3.1 is only the fraction of gas released during the RIA transient and does not include the release during normal operation. The release fractions provided in Figure 3.1 and Appendix B are relative to the total gas produced in the fuel. Therefore, a new recommended release will be provided in this report to replace the value of 0.10 specified in Regulatory Guides 1.77 and 1.183.

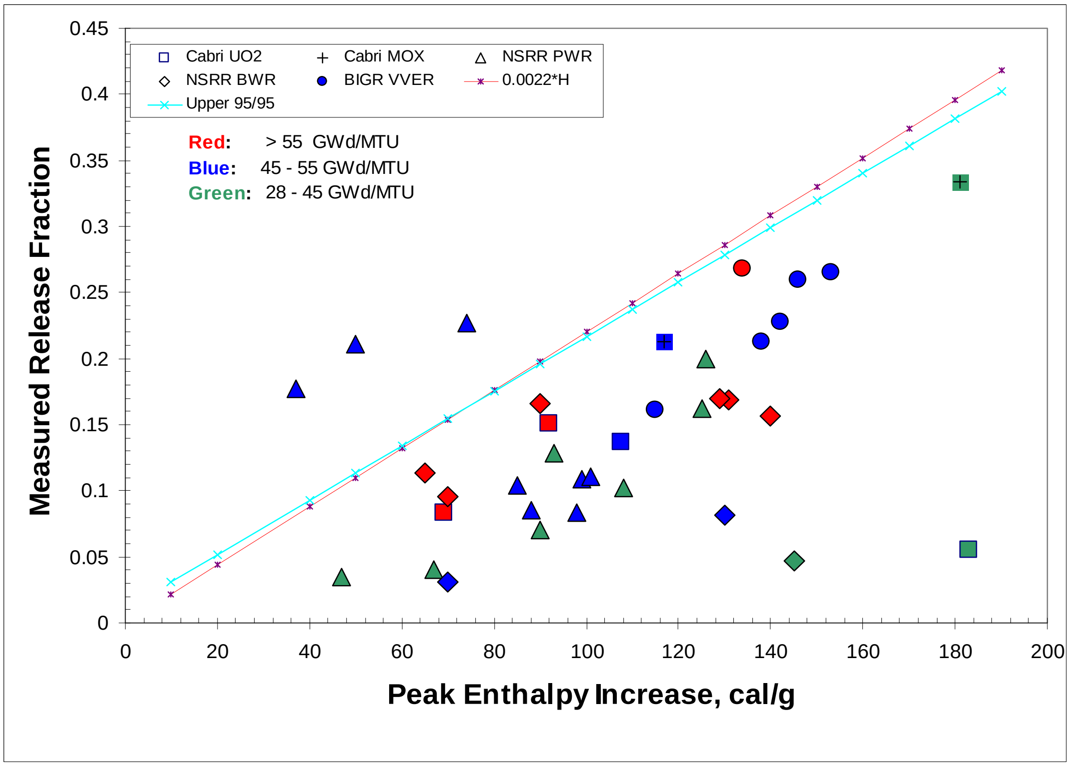

Figure 3.1. Stable Fission Gas Release Data as a Function of Peak Fuel Enthalpy Increase from Simulated RIA Tests in CABRI, NSRR, and BIGR Test Reactors

3.1 Discussion of RIA Release Data and Recommendations for Gap Fractions

From examination of the data in Appendix B, it is observed that the pulse widths from these different test reactors varied considerably with the CABRI tests having the widest pulse width between 9 to 76 millisecond (ms), the NSRR tests between 4 to 7 ms, and the BIGR tests having a pulse width of 2 to 3 ms. Examination of the data in Figure 3.1 and Table B.1 in Appendix B reveals that release increases with increasing enthalpy and that pulse width does not appear to have a large influence on FGR between ~ 2 to 76 ms. Also, fuel burnup may have an impact on release but the scatter in the data does not allow a definitive relationship to be established. The release fractions are from test rods with very short lengths such that the enthalpy increases and release values can be considered to be local rather than for a full-length LWR fuel rod.

An upper 95/95 tolerance level curve is presented in Figure 3.1 that bounds the RIA release data with the exception of three data points from NSRR tests of PWR segmented rods. The upper tolerance curve does not intercept the origin (a small positive release of 0.01 at zero enthalpy increase); consequently, a slightly different relationship than the 95/95 tolerance curve is recommended such that the release fraction for long-lived isotopes can bounded by the relationship:

$$F \text{ (stable)} = 0.0022 * \Delta H$$

where ΔH is the enthalpy increase in cal/gm.

This relationship provides a zero release at zero enthalpy increase.

The three short fuel test rods that are not bounded by the 95/95 curve are HBO-2, HBO-3, and HBO-4. These three test rods were refabricated from the same full-length PWR rod and then RIA tested in NSRR. The Japan Atomic Energy Research Institute (JAERI) papers and reports on these tests note that the release data from these three HBO specimens were anomalous compared to the rest of the release data from the NSRR tests. It should be noted that they are also anomalous to the remainder of the 32 other RIA FGR data in Figure 3.1. These JAERI reports noted that the fuel fabrication process for HBO-2, -3, and -4 rods was different (labeled as Type-A fuel) than the rest of the HBO test series (labeled as Type-B fuel), but was similar to some of the fuel in the TK series test rods. For example, the TK-4 rod had Type-A fuel with similar burnup of 50 GWd/MTU but peak enthalpy was over twice as high for TK-4 as for HBO-2 (98 cal/gm versus 37 cal/gm, respectively), thus suggesting that TK-4 should have significantly higher FGR. However, the FGR in HBO-2 was over twice as high FGR as TK-4 (17.7 versus 8.3 percent release). This suggests that some unknown phenomenon caused higher FGR in HBO-2.

It has also been hypothesized that the higher FGR of the HBO-2, -3, and -4 rods may be due to their base irradiation (commercial reactor) powers being different from the other fuel rods shown in Figure 3.1 at equivalent burnup levels. However, examination of both TK-4 and HBO-2 test specimens demonstrated that they had similar base irradiation power histories. Therefore, base irradiation power histories do not explain the high release in the HBO-2, -3, and -4 rods unless there are errors in the base irradiation powers. Therefore, there is no clear explanation for why the HBO Type A fuel experienced significantly higher FGR than any other RIA tests performed in CABRI, NSRR (including other NSRR tests with Type-A fuel), and BIGR at low fuel enthalpies.

Further examination of Figure 3.1 also shows that two NSRR BWR specimens and one CABRI PWR specimen provide significantly lower release than the majority of the other release data. The largest deviation was from a PWR CABRI test rod (REP Na-2) with the lowest burnup level (33 GWd/MTU) of the UO_2 test rods. The two NSRR BWR test rods (FK-1 and FK-3) were at relatively low burnups of 45 to 41 GWd/MTU, respectively. A qualitative theory of fission gas release can partially explain the lower release for these test rods based on the increase in interconnected fission gas bubbles on grain boundaries with increasing burnup. The fission gas release from the RIA test rods appears to be from the fracturing of the grain boundaries within the high burnup fuel rim and main body of the fuel and not due to diffusion during the RIA tests. These lower burnup fuel rods have little or no fuel rim and have less grain boundary gas in the main body of the fuel. Therefore, the lower burnup fuel will have less grain boundary gas than the higher burnup fuel with the latter having more inventory for release during the RIA. It is further noted that the low burnup (only 28 GWd/MTU) mixed oxide fuel (MOX) test rod from CABRI (REP Na-9) was within the release amounts of the higher burnup UO_2 test rods. This can be explained by the bubble interconnection process, which appears to occur in MOX (in the PuO_2 rich particles) at much lower burnups than for UO_2 (Lemoine et al. 2000).

A bounding (at a 95 percent probability with 95 percent confidence) relationship for Kr-85 release during a RIA has been developed that is only a function of peak fuel enthalpy increase. The bounding gap release fraction, F, for Kr-85 is defined as (assumes Kr-85 is stable):

$$F\ (\text{Kr-85}) = 0.0022 * \Delta H$$

where ΔH is the enthalpy increase in cal/gm.

There are no release data for the cesium, iodine, or short-lived noble gas isotopes from the RIA test rods. Therefore, their release fractions are estimated from the bounding relationship for stable noble gases and Kr-85 above. The release of the long-lived cesium isotopes (Cs-134 and Cs-137) can be estimated utilizing the ANS 5.4 standard recommendation in that cesium has a factor of 2 higher diffusion coefficient than the noble gases. Since release fraction is approximately proportional to the square root of the diffusion coefficient, the bounding release fraction for the long-lived cesium isotopes can be expressed as:

$$F\ (\text{Cs}) = 0.0022 * \Delta H * (2)^{0.5} = 0.0031 * \Delta H$$

where ΔH is the enthalpy increase in cal/gm.

There are no release data for the short-lived isotopes for a RIA event, only release data for the stable noble gases exist; therefore, the short-lived isotope releases must be estimated from the stable noble gas release data. For short-lived isotopes, the time the gas is created (born) to the time it takes to be released is critical for determining the release fraction, R/B, because the longer the holdup of gas in the fuel, the more R/B is reduced due to decay of the isotope.

The release of the short-lived isotopes of the noble gases and iodine during a RIA is similar to that for steady-state power operation once the grain boundary is saturated. This is because once the grain boundaries are saturated during steady-state operation, there is no holdup of the gas on the grain boundaries and there is no holdup on grain boundaries during a RIA. The actual physical mechanism for release from the boundary is different between a RIA and that during steady-state operation. The RIA release is due to the fact that the large temperature increase within the fuel during this event fractures the grain boundaries, releasing the gas on the boundaries immediately. The steady-state release is caused by an interconnection of the (when bubble saturation is achieved) gas-filled bubbles on the boundary due to gas diffusion to the boundary opening the boundary for release. Both mechanisms release all of the grain boundary gas (radioactive and stable). The grain boundary saturation level for release decreases with fuel temperature for normal power operation such that an increase in fuel temperature over a few hours from a power ramp will also release grain boundary gas similar to a RIA event where the grain boundaries are fractured. Therefore, there is a ratio between the radioactive R/B release and the stable release fractions, F, which is primarily dependent on the fuel temperature increase (delta power increase) and burnup for both a RIA event and a slow power increase during normal operation. This ratio can then be used to estimate the release for a given isotope, such as I-131, which is of primary importance for dose calculations for a RIA.

Calculations have been performed with the FRAPCON-3.3 code and the 2011 ANS 5.4 model to examine the ratio between stable noble gas release and the release of I-131 for the BWR 9 x 9 and PWR 14 x14 fuel design at power increases of 14, 26, 31, and 41 percent and at rod average burnups between 12 to 38 GWd/MTU. Only the release of the I-131 isotope was examined because it has the highest R/B release of the short-lived volatiles that has the largest impact on dose calculations. The ratio of the best estimate predicted release fractions between the stable noble gases and I-131 at a given time step when power is increased provides an indication of the delay time between when a I-131 atom is produced to when it is released during normal power operation with little holdup on the grain boundary. Examination of the calculated results demonstrates that the ratio between the stable isotopes and I-131 release (e.g., $F_{stable}/R/B_{I\text{-}131}$), is typically between 6 to 15 when the power is increased between 14 to 41 percent and rod average burnups are between 12 to 38 GWd/MTU. An increase in power of 41 percent for steady-state power operation results in a delta increase in stable release fraction of 0.15, which is the upper range of delta release of a RIA for an LWR. The ratio of $F_{stable}/R/B_{I\text{-}131}$ varies depending on power and burnup. This suggests that the release fraction from decay for I-131 is reduced by a factor of 6 to 15 due to the time for diffusion to the grain boundary and release. Therefore, it can be conservatively assumed that the diffusion from the fuel grain matrix to the grain boundary with no holdup on the grain boundary reduces the fractional release by a factor of 3 compared to the stable isotopes. As noted, the actual reduction in fractional release compared to the stable isotopes is most likely between a factor of 6 to 15, but without actual I-131 release data for test rods with simulated RIA power increases at various burnup levels, it is difficult to determine the exact factor of reduction in release.

The bounding gap release fraction, R/B, for I-131 and the other short-lived isotopes is defined as:

$$F \text{ (short life isotopes)} = (0.33)*0.0022*\Delta H = 0.00073*\Delta H$$

where ΔH is the enthalpy increase in cal/gm.

The combined total RIA gap release fractions equals the steady-state gap fraction (Section 2.0) plus the transient releases provided in this section, as summarized in Table 3.1. An example cycle specific calculation for a RIA event is provided in Appendix C.

Table 3.1. Local Gap Release Fractions for Reactivity Initiated Accidents

Isotope	Combined RIA Release Fraction[a,b]
Kr-85	((0.38) + (0.0022 * ΔH))
I-131	((0.08) + (0.00073 * ΔH))
I-132	((0.09) + (0.00073 * ΔH))
Other Nobles	((0.08) + (0.00073 * ΔH))
Other Halogens	((0.05) + (0.00073 * ΔH))
Alkali Metals	((0.50) + (0.0031 * ΔH))

a. ΔH= increased fuel enthalpy during RIA event
b. Assumes no fuel melting

4.0 Conclusions and Limitations

The recommended bounding release fractions for accidents without large temperature increases and that do not involve a full core of assemblies (only a small number of assemblies) are summarized in Table 2.9. The limitations of these release fractions are the power and burnup plots provided in Figure A.1 for PWRs and Figure A.3 for BWRs in Appendix A. If the peak rods in future cores exceed these power and burnup levels, the maximum release values in Tables 2.4, 2.8, 2.9, and 3.1 are no longer applicable.

The recommended bounding release fractions for RIA accidents are summarized in Table 3.1. These release fractions include both the steady-state gap inventory and the transient fission gas release.

5.0 References

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Appendix A

Code Input Including Power Histories and Axial Power Shapes Used for FRAPCON-3.3/ANS 5.4 Analysis

Appendix A

Code Input including Power Histories and Axial Power Shapes Used for FRAPCON-3.3/ANS 5.4 Analyses

Table A.1. FRAPCON-3.3 Input Parameters for PWR and BWR Analyses

Description of Design Parameter	PWR 14 x 14	BWR 9 x 9
Pitch (mm, *in*)	14.7, *0.58*	13.0, *0.510*
Cladding OD (mm, *in*)	11.2, *0.44*	10.8, *0.424*
Cladding Thickness (mm, in)	0.737, *0.029*	0.711, *0.028*
Cladding ID (mm, *in*)	9.70, *0.382*	9.35, *0.355*
Diametral Gap Thickness (mm, *in*)	0.191, *0.0075*	0.203, *0.008*
Fuel Pellet Diameter (mm, *in*)	9.51, *0.374*	9.14, *0.360*
Plenum Spring Diameter (mm, *in*)	9.51, *0.374*	9.14., *0.360*
Pellet Length (mm, *in*)	12.7, *0.5*	11.2, *0.44*
Dish Diameter (mm, *in*)	4.57, *0.18*	5.81, *0.2288*
Dish Depth (mm, *in*)	0.305, *0.012*	0.216, *0.0085*
Plenum Length (mm, *in*)	254, *10*	245, *9.64*
Turns in Plenum Spring	30	29
Plenum Spring Wire Diameter (mm, *in*)	1.32, *0.052*	1.4, *0.055*
Helium Fill Gas Pressure (MPa, *psi*)	2.07, *300*	1.03, *150*
Active Fuel Length (m, *in*)	3.66, *144*	3.81, *150*
System Coolant Pressure (MPa, *psi*)	15.5, *2250*	7.27, *1055*
Coolant Inlet Temperature (°C, °*F*)	288, *550*	277, *530*
Coolant Flow Rate (x10^3 kg/s-m^2, x10^6 lb/hr-ft^2)	3.60, *2.65*	1.67, *1.23*
Enrichment (atom %)	4.95	5
Pellet Density (% TD)	96	96
Limit on Pellet Density Increase (% TD)	0.7	0.7
Fuel Surface Roughness (μm, *in*)	1.6, $6.3x10^{-5}$	1.5, $5.9x10^{-5}$
Cladding Surface Roughness (μm, *in*)	1.0, $3.9x10^{-5}$	0.9, $3.6x10^{-5}$
Cladding Material	ZIRLO™	Zircaloy-2
Cold Work (%)	50	0

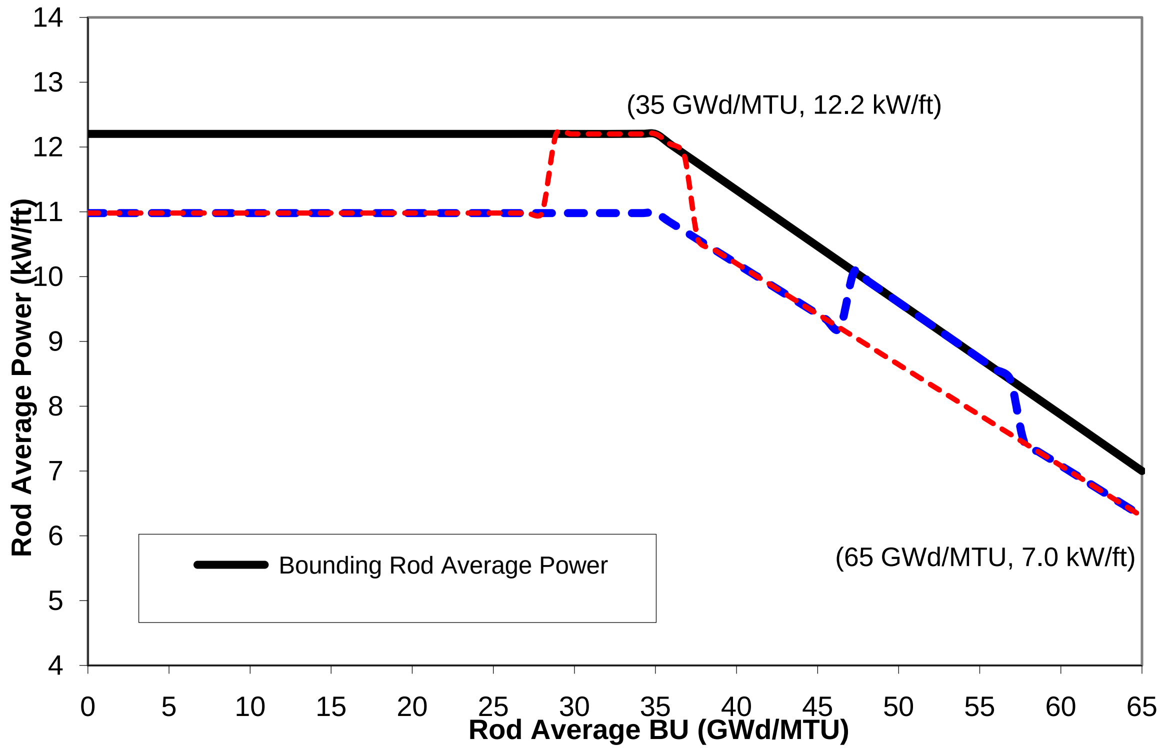


Figure A.1. Bounding PWR LHGR and Limiting Segmented Power Histories for Stable (PWR History #6) and Short-Lived Releases (PWR History #4)

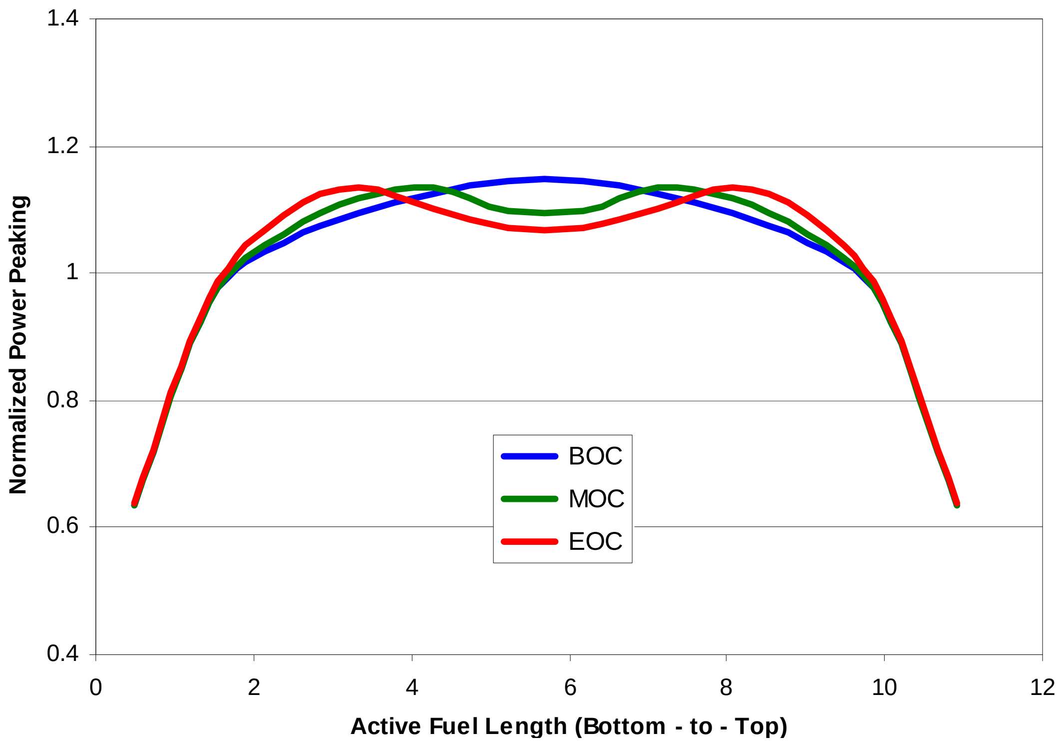


Figure A.2. PWR Normalized Axial Power Shapes (Peak F_z=1.144)

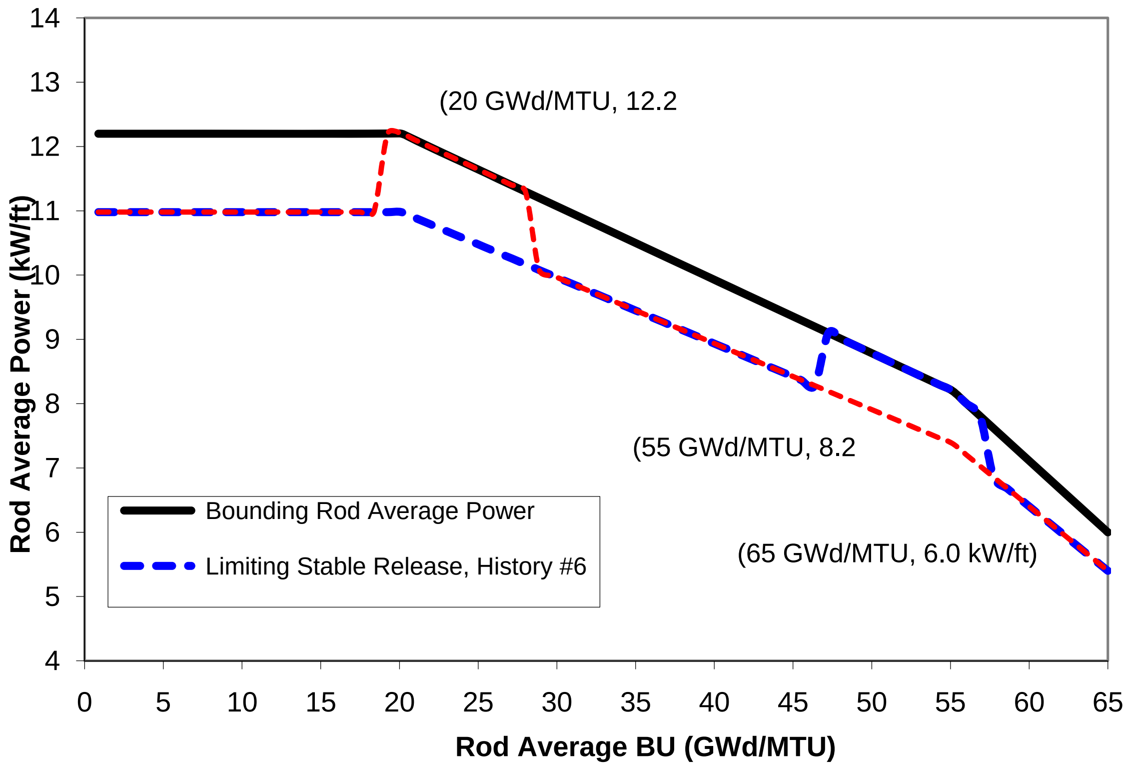


Figure A.3. Bounding BWR LHGR and Limiting Segmented Power Histories for Stable and Short-Lived Releases (BWR History #3)

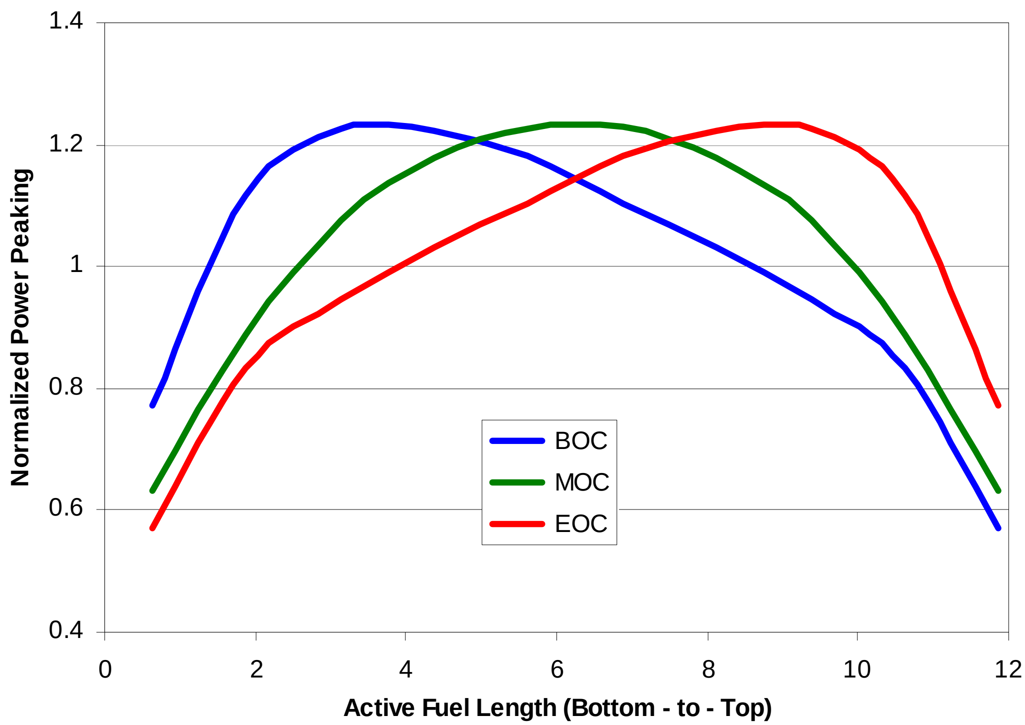


Figure A.4. BWR Normalized Axial Power Shapes (Peak F_z=1.228)

Appendix B

Compilation of Fission Gas Release Data From RIA Tests in CABRI, NSRR, and BIGR

Appendix B

Compilation of Fission Gas Release Data from RIA Tests in CABRI, NSRR and BIGR

Table B.1. Fission Gas Release Data from RIA Tests in CABRI, NSRR and BIGR

Test Rods	Burnup, GWd/MTU	Enthalpy Deposited, cal/gm	Pulse width, ms	Enthalpy Increase ΔH, cal/gm	FGR Fraction
CABRI PWR					
Na2	33	207	9.6	183	0.0554
Na3	53.8	122.2	9.5	107.5	0.137
Na4	62	95	76.4	69	0.083
Na5	64	104	8.8	92	0.151
Na6	47	156	32	117	0.213
Na9	28.1	233	33	181	0.334
NSRR PWR					
TK1	38	126	4.8	126	0.2
TK3	50	99	4.8	99	0.109
TK4	50	98	4.8	98	0.083
TK5	48	101	4.8	101	0.111
TK6	38	125	4.8	125	0.162
HBO2	50.4	37	4.8	37	0.177
HBO3	50.4	74	4.8	74	0.227
HBO4	50.4	50	4.8	50	0.211
HBO6	49	85	4.8	85	0.104
HBO7	49	88	4.8	88	0.085
OI-2	39.2	108	4.4	108	0.102
MH1	38.9	47	4.5	47	0.035
MH3	38.9	67	4.5	67	0.04
GK1	42.1	93	4.6	93	0.128
GK2	42.1	90	4.6	90	0.07
NSRR BWR					
FK-1	45.4	130	4.5	130	0.082
FK-2	45.4	70	7	70	0.031
FK-3	41	145	4.5	145	0.047
FK-4	56	140	4.3	140	0.157
FK-5	56	70	7.3	70	0.096
FK-6	61	131	4.3	131	0.169
FK-7	61	129	4.3	129	0.17
FK-8	61	65	7.3	65	0.113
FK-9	61	90	5.7	90	0.166
BIGR VVER					
RT2	47-49	115	2	115	0.161
RT3	47-49	138	2	138	0.213
RT1	47-49	142	2	142	0.228
RT5	47-49	146	2	146	0.26
RT6	47-49	153	2	153	0.265
RT7	60	134	2	134	0.268

Appendix C

Analytical Technique for Calculating Fuel Design Specific Non-LOCA Fission Product Gap Inventories

Appendix C

Analytical Technique for Calculating Fuel Design Specific Non-LOCA Fission Product Gap Inventories

This appendix provides an acceptable analytical technique for calculating non- loss-of-coolant accident (LOCA) fission product gap inventories based upon specific fuel rod designs or more realistic fuel rod power histories. Using the analytical technique detailed below, bounding gap inventories were developed based on bounding fuel rod power histories and limiting fuel rod designs. Table 3 (Section 3.2 of DG-1199) and Table 2.9 of this report lists those bounding (maximum) gap inventories. Lower gap fractions are achievable using less aggressive rod power histories and/or less limiting fuel rod designs (e.g., 17 x 17 versus 14 x 14 fuel rod design). Alternatively, applicants may use the bounding gap inventories provided in Table 3 of DG-1199 provided they satisfy the applicability requirements in footnote 11 of DG-1199 and peak radial average power envelope in Figure 1 of DG-1199.

There are two basic types of non-LOCA scenarios that produce fuel cladding failure and subsequent radiological source terms: 1) those that do not involve a significant fuel thermal transient such as the fuel handling accident and single reactor coolant pump locked rotor, and 2) those that involve a very large thermal (power) transient such as the reactivity insertion accident (RIA). These two types of accidents require different approaches for predicting release fractions of radioactive fission products. These two approaches are described in the following two sub-sections.

C.1 Gap Fractions for Accident Not Involving a Significant Thermal Transient

Non-LOCA gap inventories represent radioactive fission products generated during normal steady-state operation that have diffused within the fuel pellet, been released into the fuel rod void space (i.e., rod plenum, pellet-to-cladding gap), and are available for release upon fuel rod cladding failure. Non-LOCA events such as reactor coolant pump locked rotor predict departure from nucleate boiling cladding failure of the highest power fuel rods in the low flow region of the reactor core. Likewise, a fuel handling accident may involve mechanical failure of all fuel rods within a single assembly. As such, non-LOCA gap inventories should represent the limiting fuel rods in the reactor core at the most limiting time-in-life. This approach results in significantly larger gap inventories than core-average source terms used in LOCA dose calculations.

The U.S. Nuclear Regulatory Commission (NRC) maintains the FRAPCON (Lanning et al. 2005) fuel rod thermal-mechanical performance code to perform independent audit calculations for licensing activities. As such, FRAPCON may not be used by licensees to justify plant-specific, fuel-specific, or cycle-specific gap inventories.

The following attributes should be included in the analytical technique used to calculate non-LOCA gap inventories:

- For stable, long-lived radioactive isotopes such as Kr-85, Cs-134, and Cs-137, an NRC-approved fuel rod thermal-mechanical performance code with established modeling uncertainties shall be used to predict the integral fission gas release. The code should include the effects of thermal conductivity degradation with burnup and have been verified against measured fuel temperatures and stable fission gas release data up to the licensed burnup of the particular fuel rod design.

- Long-lived radioactive isotopes will continue to accumulate throughout exposure with insignificant amounts of decay because of their long half life. As such, maximum gap inventories for long-lived isotopes are likely to occur near or at the end-of-life of the fuel assembly.
- 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, the cesium release fraction equals:

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

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

- For volatile, short-lived radioactive isotopes such as iodine (i.e., I-131, I-132, I-133, I-135) and xenon and krypton noble gases except Kr-85 (i.e., Xe-133, Xe-135, Kr-85m, Kr-87, Kr-88), an NRC-approved release model or the NRC-endorsed 2011 ANS 5.4 release model shall 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 ANS 5.4 model calculations of R/B are local fuel temperature, fission rate, and axial node/pellet burnup. The code should include the effects of thermal conductivity degradation with burnup and have been verified against measured fuel temperatures and stable fission gas release data up to the licensed burnup of the particular fuel rod design.
  - Due to 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 power operating envelope). For current pressurized-water reactors (PWR)s, this is approximately 25 to 33 GWd/MTU. For current boiling-water reactors (BWR)s, this is approximately 15 to 20 GWd/MTU.
  - Guidance related to the calculation of short-lived (R/B) factors when using the ANS-5.4 release model is provided in NUREG/CR-7003 (Turnbull and Beyer 2010).
    - For nuclides with half-lives of less than 1 hour, no gap inventories are provided. Due to their rapid decay (relative to time for diffusion and transport), these nuclides will be bounded by the calculated gap fractions for longer-lived nuclides under the heading “Other Noble Gases” and “Other Halogens.”
    - For nuclides with half-lives of less than 6 hours, (R/B) predicted by NRC-approved fuel performance code, use Equation 12 of and terms defined in NUREG/CR-7003.

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

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

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

  The R/B 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 Te-132 has a half-life of 3.2 days, which controls the release of I-132.

- Fractal-scaling factors for each nuclide are listed in Table C.1 of this Appendix or calculated using the following equation from NUREG/CR-7003:

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

- Fission product gap inventories are calculated at a 95 percent probability and 95 percent confidence level (95/95).
  - For short-lived isotopes, the 95/95 upper tolerance gap inventory is based on the empirical database used in the development of the fission gas release model. For example, the 2011 ANS-5.4 release model standard recommends multiplying the best estimate predictions by a factor of 5.0 to obtain upper tolerance gap inventories.
  - For long-lived isotopes, the 95/95 upper tolerance (e.g., $\mu + k\sigma$ if normal distribution is valid) gap inventory is based on the verification and validation database of the fuel thermal-mechanical code. For example, FRAPCON-3.3 (Lanning et al. 2005) predicted release fractions for long-lived isotopes exhibit a standard deviation of 0.028 (absolute) based on its validation database of measured stable noble gases from 23 fuel rods. With a database of 23 fuel rod measurements, k = 2.36 assuming a normal distribution and 23-2 = 21 degrees of freedom. If FRAPCON-3.3 predicted a Kr-85 best-estimate integral release fraction of 0.228, then the Kr-85 95/95 upper tolerance gap inventory ($\mu + k\sigma$) would equal 0.294 (0.228 + 2.36*0.028).
- Nominal fuel design specifications (excluding tolerances) may be used.
- Actual in-reactor fuel rod power histories may diverge from reload core depletion calculations due to unplanned shutdowns or power maneuvering. As a result, the rod power history or histories used to predict gap inventories need to bound anticipated operation. Rod power histories used in the fuel rod design analysis based on core operating limits report (COLR) thermal-mechanical operating limits (TMOL) or radial falloff curves may be used. Any rod power history must be verifiable.
  - An example is provided for the power history used in calculating the gap inventories inTable 3 of DG-1199 (Table 2.9 of this report). This calculation employed a segmented power history for both the BWR and PWR limiting designs whereby seven different power histories were considered with each running at 90% of the bounding rod average power, with the exception of running at the LHGR limit for approximately 9 to 10 GWd/MTU burnup (rod average) at seven different burnup intervals.

- A flatter axial power distribution (e.g., low value of F_z) spreads the power and promotes a higher fission gas release along the fuel stack. A bounding axial power distribution should be used. Any rod axial power profile must be verifiable.
- Each fuel rod design (e.g., UO_2, UO_2-Gd_2O3, part-length, full-length) must be evaluated.
- The minimum acceptable number of radial and axial nodes is defined in the 2011 ANS-5.4 standard along with the methodology of summing the release for these nodes to determine the overall release from the fuel to the fuel-cladding gap.

The following example calculation illustrates the potential improvement in radiological source term achievable by calculating less bounding gap fractions. In this example, the licensee elects to calculate gap inventories based upon cycle-specific rod designs and power profiles. The resulting gap fractions are significantly lower than the generic, bounding values in Table 3 of DG-1199.

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

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

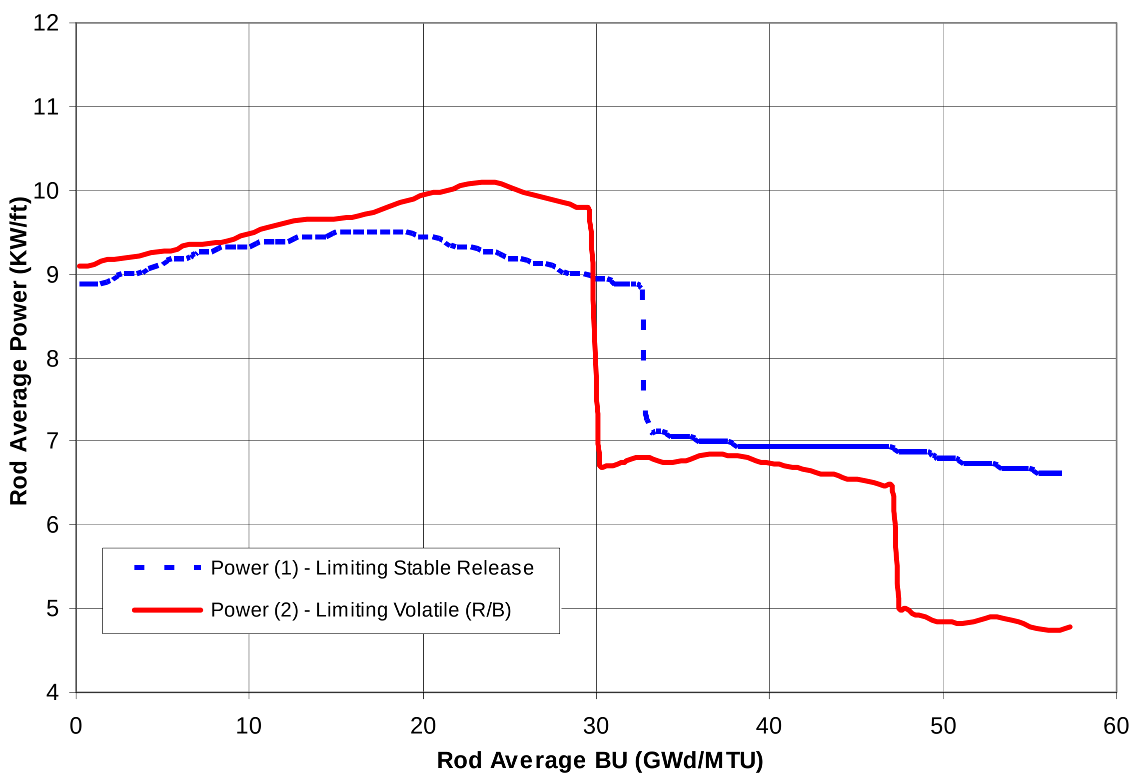
a. I-132 alpha term accounts for significant contribution from precursor Te-132 with its 4x diffusion coefficient.

EXAMPLE CALCULATION
PWR Gap Inventories Based on Realistic Power Histories

This example illustrates the potential improvement in radiological source term from calculating less bounding gap fractions. For this example, the licensee elects to calculate gap inventories based upon cycle-specific rod designs and power profiles.

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. Adjustments are then made 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 for the calculation of volatile (R/B) ratios. The licensee has verified that the full-length UO_2 fuel rod design is the most limiting.

As discussed in the analytical procedure, maximum stable releases occur at EOL 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 at low-to-middle burnup. 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 AOO fuel centerline melt analysis will coincide with that for maximum (R/B) ratios.



EXAMPLE CALCULATION
PWR Gap Inventories Based on Realistic Power Histories (Continued)

In this example, the FRAPCON-3.3 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. Following the analytical guidance, adjustments were made to calculate the remaining nuclides.

Long-Lived Stable Release:

$Kr\text{-}85_{95/95}$ = [$(\text{EOL FGR})_{FRAPCON} + (k\sigma)_{FRAPCON}$]

Power 1 = [(0.0614) + 2.36*0.028] = 0.1275

Power 2 = [(0.0474) + 2.36*0.028] = 0.1135

$Cs\text{-}134_{95/95}$ = [$(\text{EOL FGR})_{FRAPCON} + (k\sigma)_{FRAPCON}$]

Power 1 = [$(0.0614)(2.0)^{0.5} + 2.36*0.028$] = 0.1529

Power 2 = [$(0.0474)(2.0)^{0.5} + 2.36*0.028$] = 0.1331

Short-Lived Volatile (R/B) Ratio:

$Kr\text{-}85m_{95/95}$ = [$(\text{Maximum Kr-85m R/B})_{FRAPCON}$ * 5.0]

Power 1 = [(0.0008) * 5.0] = 0.0040

Power 2 = [(0.0013) * 5.0] = 0.0065

$Kr\text{-}87_{95/95}$ = [$(\text{Maximum Kr-87 R/B})_{FRAPCON}$ * 5.0]

Power 1 = [(0.0004) * 5.0] = 0.0020

Power 2 = [(0.0006) * 5.0] = 0.0030

$Kr\text{-}88_{95/95}$ = [$(\text{Maximum Kr-88 R/B})_{FRAPCON}$ * 5.0]

Power 1 = [(0.0006) * 5.0] = 0.0030

Power 2 = [(0.0009) * 5.0] = 0.0045

$Xe\text{-}133_{95/95}$ = [$(\text{Maximum Kr-85m R/B})_{FRAPCON}$ * 5.0 * $(\text{Fractal Scaling})_{Xe\text{-}133}$]

Power 1 = [(0.0008) * 5.0 * 2.276] = 0.0091

Power 2 = [(0.0013) * 5.0 * 2.276] = 0.0148

$Xe\text{-}135_{95/95}$ = [$(\text{Maximum Kr-85m R/B})_{FRAPCON}$ * 5.0 * $(\text{Fractal Scaling})_{Xe\text{-}135}$]

Power 1 = [(0.0008) * 5.0 * 1.301] = 0.0052

Power 2 = [(0.0013) * 5.0 * 1.301] = 0.0085

$I\text{-}131_{95/95}$ = [$(\text{Maximum Kr-85m R/B})_{FRAPCON}$ * 5.0 * $(\text{Fractal Scaling})_{I\text{-}131}$]

Power 1 = [(0.0008) * 5.0 * 2.395] = 0.0096

Power 2 = [(0.0013) * 5.0 * 2.395] = 0.0156

EXAMPLE CALCULATION:
PWR Gap Inventories Based on Realistic Power Histories (Continued)

$I\text{-}132_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}132}$]

Power 1 = [(0.0008) * 5.0 * 2.702] = 0.0108

Power 2 = [(0.0013) * 5.0 * 2.702] = 0.0176

$I\text{-}133_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}133}$]

Power 1 = [(0.0008) * 5.0 * 1.439] = 0.0058

Power 2 = [(0.0013) * 5.0 * 1.439] = 0.0094

$I\text{-}134_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}134}$]

Power 1 = [(0.0008) * 5.0 * 0.900] = 0.0036

Power 2 = [(0.0013) * 5.0 * 0.900] = 0.0059

$I\text{-}135_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}135}$]

Power 1 = [(0.0008) * 5.0 * 1.029] = 0.0041

Power 2 = [(0.0013) * 5.0 * 1.029] = 0.0067

The cycle-specific fuel rod design and power history gap inventories are listed below along with the generic, bounding values from Table 3 of this regulatory guide.

Group	Gap Inventory	
	Bounding	Cycle-Specific
I-131	0.08	0.02
I-132	0.09	0.02
Kr-85	0.38	0.13
Other Noble Gases	0.08	0.02
Other Halogens	0.05	0.01
Alkali Metals	0.50	0.16

C.2 Gap Fractions for a RIA

The total fission-product source term during an RIA includes both (1) the steady-state gap inventory present from normal operation prior to the RIA and (2) any transient fission gas released during the event. Stable fission gas release measurements taken during RIA test programs in the CABRI, NSRR, and BIGR test reactors were used to develop the transient fission gas release component of the combined source term. The empirical database for RIA transient fission gas release, which encompasses a wide array of fuel rod designs, exhibits a strong dependency on the increase in radial average fuel enthalpy. As a result, Table 4 Section 3.2 of DG-1199 and Table 3.1 of this report provide equations for calculating the transient fission gas release component and combined gap inventory as a function of change in radial average fuel enthalpy.

The following example illustrates the calculation of combined gap fractions for I-131 (volatile R/B) and Kr-85 (stable release). In this example, the licensee elects to divide the fuel rod into five axial regions, each predicted to experience a different deposited fuel enthalpy during the postulated RIA.

EXAMPLE CALCULATION
RIA Axial Dependent Transient Component of Gap Inventory

Due to the large variation in axial-dependent power peaking factors, an analyst might chose to divide the fuel rod into five equal length (equal volume) axial regions as shown in the example below. If non-equal volumes are used this needs to be accounted for in the calculation of release fractions and total atoms released. The number of axial regions is up to the analyst with the guidance that the fuel enthalpy within a given axial region not be underestimated. The peak radial average enthalpy change in each region is used to calculate the transient fission gas release component of the gap fraction within each axial region.

Axial Region	Peak Enthalpy Increase
Region #1 (Top)	150 cal/gm
Region #2	100 cal/gm
Region #3	75 cal/gm
Region #4	25 cal/gm
Region #5 (Bottom)	0 cal/gm

Combined Gap Inventory = [(Steady-State) + (Transient Release)]

Combined $R/B_{(I\text{-}131)}$ = [0.08[2] + ((0.00073*150/5) + (0.00073*100/5) + (0.00073*75/5) + (0.00073*25/5) + (0.00073*0/5))]
= 0.131

Combined $F_{(Kr\text{-}85)}$ = [0.38[3] + ((0.0022*150/5) + (0.0022*100/5) + (0.0022*75/5) + (0.0022*25/5)+(0.0022*0/5))]
= 0.504

C.3 References

Lanning DD, CE Beyer, and KJ Geelhood. 2005. *FRAPCON-3 Updates, Including Mixed Oxide Properties*, NUREG/CR-6534 (PNNL-11513) Vol. 4., U.S. Nuclear Regulatory Commission, Washington D.C.

Turnbull JA, and CE Beyer. 2010. *Background and Derivation of ANS-5.4 Standard Fission Product Release Model*, NUREG/CR-7003 (PNNL-18490) January 2010, U.S. NRC, Office of Nuclear Regulatory Research.

[2] For example, the values for normal operation steady-state gap inventory were taken from Table 2.9 of this report, which are also given in DG-1199 Table 3, Section 3.2. These may be calculated as per Section C.1 above.

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