Note: this document contains draft/predecisional guidance language to support discussions during the NRC's High Burnup Workshop and other public meetings, as needed. The NRC staff are not officially accepting comments during the public meetings and will not provide any responses to feedback provided during public meetings.

The contents of this document are being considered for inclusion in staff guidance in DG-1425. The purpose of this document is to engage stakeholders and receive informal feedback. This document has not been subject to NRC management or legal reviews or approvals, and its contents are subject to change and should not be interpreted as official agency positions.

**Figure 1. Maximum allowable power operating envelope for PWR steady-state release fractions**

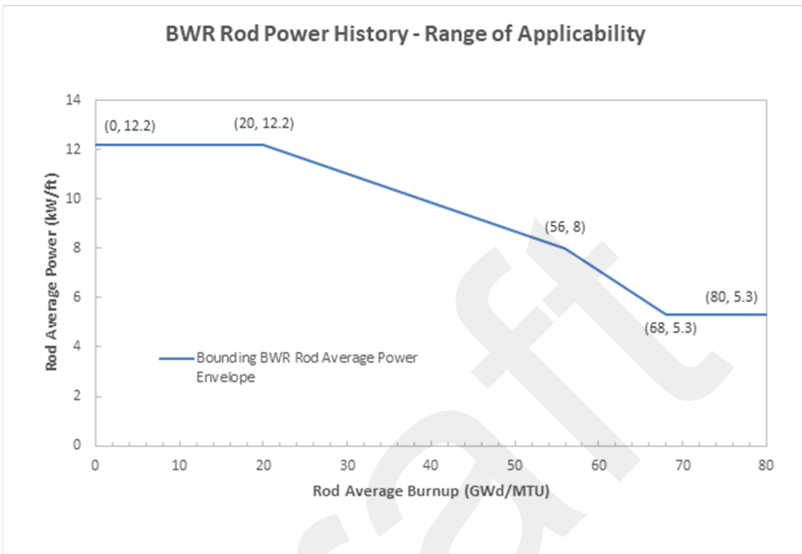


**Figure 2. Maximum allowable power operating envelope for BWR steady-state release fractions**

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 ~~as a result~~because of pellet fracturing and grain boundary separation. This transient fission gas release ($T_{FGR}$) increases the amount of activity available for release into the reactor coolant system for fuel rods that experience cladding breach. The empirical database suggests that $T_{FGR}$ is sensitive to both local fuel burnup and peak radial average fuel enthalpy rise. As a result, separate low-burnup and high-burnup $T_{FGR}$ correlations for stable, long-lived radionuclides (e.g., krypton (Kr)-85 and cesium-137) are provided, as follows:

pellet burnup < 50 GWd/MTU,

$T_{FGR}$ (long-lived isotopes) = maximum [ (0.26 * ΔH) – 13) / 100, 0 ], (Equation 1)

pellet burnup ≥ 50 GWd/MTU,

$T_{FGR}$ (long-lived isotopes) = maximum [ (0.26 * ΔH) – 5) / 100, 0 ], (Equation 2)

~~where~~
Where:

$T_{FGR}$ = transient fission gas release fraction, and
ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

An investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release concluded that different radionuclides require adjustments to the above empirically based correlations (Ref. ~~28~~44). For stable, long-lived noble gases (e.g., Kr-85) and alkali metals (e.g., cesium-137), 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) and xenon (Xe) and Kr noble gases except Kr-85 (e.g., Xe-133, Kr-85m), the transient fission product release correlations should be multiplied by a factor of 0.333. The low-burnup and high-burnup $T_{FGR}$ correlations for volatile, short-lived radioisotopes are as follows:

pellet burnup < 50 GWd/MTU,

$T_{FGR}$ (short-lived isotopes) = 0.333 * maximum [ (0.26 * ΔH) – 13) / 100, 0 ], (Equation 3)

pellet burnup ≥ 50 GWd/MTU,

$T_{FGR}$ (short-lived isotopes) = 0.333 * maximum [ (0.26 * ΔH) – 5) / 100, 0 ], (Equation 4)

~~where~~
Where:

$T_{FGR}$ = transient fission gas release fraction, and
ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

For the remaining high- temperature non-LOCA DBAs that predict fuel rod cladding failure, such as the PWR reactor coolant pump locked rotor and ~~fuel handling accident~~main steamline break, additional fission product releases may occur ~~as a result~~because of fuel pellet fragmentation (e.g., fracturing of high-burnup rim region) due to loss of pellet-to-cladding mechanical constraint or impact loads. $T_{FGR}$ has been experimentally observed under a variety of accident conditions. At the time of issuance of Revision ~~1~~ 2 of this RG, no consensus exists on the mechanism or the computation of $T_{FGR}$ for these events; therefore, ~~future applicants should address this using engineering judgment or experimental data.~~an acceptable method to address $T_{FGR}$ for non-LOCA DBAs other than reactivity initiated accidents would be to prevent balloon and burst failures though design and analysis. Though not fully applicable to non-LOCA and non-reactivity-initiated DBAs, NRC Research Information Letter 2021-13, “Interpretation of Research on Fuel Fragmentation Relocation, and Dispersal at High Burnup,” issued December 2021 (Ref. ~~29~~45), provides data that can be used to provide a bounding estimate of $T_{FGR}$ for high-temperature DBAs.

The total fraction of fission products available for release equals the steady-state fission product release fractions in tables 3 and 4 plus any $T_{FGR}$ prompted by the accident conditions. $T_{FGR}$ may be calculated separately for each axial node based on local accident conditions (e.g., fuel enthalpy rise) and then combined to yield the total $T_{FGR}$ for a particular damaged fuel rod. An NRC internal memorandum (Ref. ~~24~~40) documents the technical bases of the steady-state fission product release fractions and $T_{FGR}$ correlations.

The non-LOCA fission product release fractions and $T_{FGR}$ correlations do not include the additional contribution associated with fuel melting. The event-specific appendices to this RG provide guidance for adjusting these gap inventories for fuel rods that are predicted to experience limited fuel centerline melting.

### 3.3 Timing of Release Phases

Table 5 provides the onset and end time of each sequential release phase for LOCA DBAs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel release phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[18] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

The applicability of table 5 is consistent with the applicability of tables 1 and 2.

**Table 5. MHA~~-~~LOCA Release Phases**

| Phase | PWRs | | BWRs | |
|---|---|---|---|---|
| | Onset | End Time | Onset | End Time |
| Gap Release | 0.5 minutes | ~~0.23~~1.3 hours | 2 minutes | 0.~~19~~7 hours |
| Early In-Vessel | ~~0.23~~1.3 hours | ~~4.~~5.3 hours | 0.~~19~~7 hours | ~~8.0~~7.4 hours |

For facilities licensed with a leak-before-break methodology, the licensee may assume the onset of the gap release phase to be 10 minutes. The licensee may propose an alternative time for the onset of the gap release phase based on facility-specific calculations using suitable analysis codes~~,~~ or based on an accepted topical report shown to apply to the specific facility. In the absence of approved alternatives, the licensee should use the gap release phase onsets in table 5.

### 3.4 Radionuclide Composition

Table 6 lists the elements in each radionuclide group that should be considered in design~~-~~basis analyses.

**Table 6. Radionuclide Groups**

| Group | Elements |
|---|---|
| Noble Gases | Xe, Kr |
| Halogens | I, Br |
| Alkali Metals | Cs, Rb |
| Tellurium Group | Te, Sb, Se |
| Barium, Strontium | Ba, Sr |
| Noble Metals | Ru, Rh, Pd, Co |
| Lanthanides | La, Nd, Eu, Pm, Pr, Sm, Y, Cm, Am |
| Cerium<br>Molybdenum | Ce, Pu, Np, Zr<br>Mo, Tc, Nb |

[18] This statement excludes the effects of radioactive decay in the core inventory on the linear release modeled. In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase (i.e., in step increases).

### 3.5 Chemical Form

Of the radioiodine released from the reactor coolant system to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. ~~With the exception of~~Except for elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this RG contain additional details.

### 3.6 Fuel Damage in Non-~~Loss-of-Coolant-Accident Design Basis Accidents~~LOCA DBAs

The amount of fuel damage caused by non-LOCA DBAs should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel cladding is breached. Cladding failure mechanisms include high-temperature failure modes (e.g., critical heat flux, local oxidation, and ballooning) and pellet-to-cladding mechanical interaction.

Appendix B to this guide addresses the modeling of the amount of fuel damage caused by a fuel handling accident.

### 3.7 Assessment of Radiological Consequences due to Fuel Fragmentation Relocation and Dispersal for Analysis of a 10 CFR 50.46 Large-Break LOCA

Recent experimental findings indicate that under certain transient conditions, portions of a reactor containing high-burnup fuel operating at sufficiently high power can fragment and escape from fuel cladding that has burst[KM-06] (Ref. 45). The escaped fuel fragments could subsequently be distributed throughout a reactor coolant system. FFRD is an important concern because of the potential dose impacts on members of the public and workers and because of the potential challenges to the coolability of the reactor core.

The staff's understanding of FFRD phenomena has continued to advance. In 2012, the NRC issued NUREG-2121, "Fuel Fragmentation, Relocation, and Dispersal during the Loss-of-Coolant Accident" (Ref. 46), which captured the results of over 90 LOCA tests performed in eight different programs over 35 years. The NRC concluded from this review that FFRD is an important consideration during 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," LOCA analyses for a reactor using higher burnup or increased enrichment fuels and that additional research into this phenomenon is required. In 2015, the NRC published SECY-15-0148, "Evaluation of Fuel Fragmentation, Relocation and Dispersal under Loss-of-Coolant Accident (LOCA) Conditions Relative to the Draft Final Rule on Emergency Core Cooling System Performance during a LOCA (50.46c)" (Ref. 47). SECY-15-0148 concluded that immediate regulatory action was not needed to address FFRD phenomena at that time based on existing fuel design limits and assumptions on how high-burnup fuel would be used.

To assess the radiological consequences of 10 CFR 50.46 LOCA analyses that predict FFRD, a new acceptance criterion has been established in table 7. This acceptance criterion is consistent with the criteria applied to other non-MHA-LOCA DBAs, such that the radiological consequences associated with an FFRD event would be like those expected from other accidents.

In 2021, the NRC assessed FFRD impacts on the MHA-LOCA source term (also referred to as the "in-containment" source term) (Ref. 48). The FFRD-induced source term comes from the fission product gases generated within the reactor fuel. At a microscopic level, gas bubbles can form within grains of fuel pellets and at grain boundaries. The pressure of this gas increases with fuel burnup. The higher pressure in the grain boundaries drives fission gases to the gap/plena of fuel rods (i.e., the small space between the outer surface of a fuel pellet and the inner surface of the cladding). Under accident conditions, the average fuel temperature rises, which increases the gas pressure in the fuel, grain boundaries, and fuel-clad gap. Some species that are solid at operating temperatures may vaporize during accident conditions, further increasing pressure both within the intergranular gas bubbles and within the fuel-cladding gap. Large pressure differences between the fuel rod and the coolant can lead to clad ballooning, which removes the mechanical restraint provided by the cladding on the fuel pellets. This loss of restraint results in the formation of stresses in the fuel and can lead to fuel fragmentation. Under accident conditions, pressures can burst the fuel clad. This results in a sudden reduction of the gas pressure in the fuel-clad gap to that of the surroundings, resulting in a sudden large pressure differential between the gases in the grain boundaries and the surrounding gas. These changes in mechanical forces can cause the pellets to fragment and release into the reactor coolant system, creating a radiological source term above the normal operational source term, as described in Regulatory Position 1.1.4.

Without a best-estimate FFRD-induced source term, licensees would use a fraction of the applicable MHA-LOCA release fractions presented in Regulatory Position 3.2. The release fraction would be determined based on the total mass of FFRD predicted. This is appropriate since the scenarios considered in the development of the MHA-LOCA source term used to demonstrate compliance with requirements in 10 CFR 50.34(a)(1)(ii)(D), 10 CFR 50.67, and 10 CFR 100.11 exclude the effects of emergency core cooling. This MHA-LOCA source term is the result of a postulated substantial meltdown of the core. As such, the MHA-LOCA source term involves far greater radiological releases from the fuel than from FFRD. However, best-estimate FFRD-induced source terms may be considered on a case-by-case basis.

> This regulatory guide does not provide guidance on demonstrating compliance with the 10 CFR 50.46 requirements or how to estimate the total mass of fuel released because of FFRD.



**Table 7.1[a] Accident Dose Criteria for EAB, LPZ, and Control Room Locations**

| **Accident or Case** | **EAB and LPZ Dose Criteria (TEDE)** | **Control Room Dose Criteria[b] (TEDE)** | **Analysis Release Duration[c]** |
|---|---|---|---|
| MHA LOCA | 0.25 sievert (Sv) (25 rem) | See table 8[d] | 30 days for containment, emergency core cooling systems (ECCS), and MSIV (BWR) leakage |
| 10 CFR 50.46 LOCA with FFRD | 0.063 Sv (6.3 rem) | 0.10 Sv (10.0 rem) | 30 days for containment, ECCS, MSIV (BWR) leakage |
| BWR Main Steamline Break | | | Instantaneous puff |
| Fuel Damage or Pre-Accident Spike | 0.25 Sv (25 rem) | 0.10 Sv (10.0 rem) | |
| Equilibrium Iodine Activity | 0.025 Sv (2.5 rem) | 0.05 Sv (5.0 rem) | |

| **Accident or Case** | **EAB and LPZ Dose Criteria (TEDE)** | **Control Room Dose Criteria[b] (TEDE)** | **Analysis Release Duration[c]** |
|---|---|---|---|
| BWR Rod Drop Accident | 0.063 Sv (6.3 rem) | 0.10 Sv (10.0 rem) | 24 hours |
| PWR Steam Generator Tube Rupture | | | Affected steam generator: time to isolate[e] |
| Fuel Damage or Pre-Accident Spike | 0.25 Sv (25 rem) | 0.10 Sv (10.0 rem) | Unaffected steam generator(s): until shutdown cooling is in operation and releases from the steam generator have been terminated |
| Concurrent Iodine Spike | 0.025 Sv (2.5 rem) | 0.05 Sv (5.0 rem) | |
| PWR Main Steamline Break | | | Until shutdown cooling is in operation and releases from the steam generators have been terminated |
| Fuel Damage or Pre-Accident Spike | 0.25 Sv (25 rem) | 0.10 Sv (10.0 rem) | |
| Concurrent Iodine Spike | 0.025 Sv (2.5 rem) | 0.05 Sv (5.0 rem) | |
| PWR Locked Rotor Accident | 0.025 Sv (2.5 rem) | 0.05 Sv (5.0 rem) | Until shutdown cooling is in operation and releases from the steam generators have been terminated |
| PWR Control Rod Ejection Accident | 0.063 Sv (6.3 rem) | 0.10 Sv (10.0 rem) | Containment pathway: 30 days; Secondary system: until shutdown cooling is in operation and releases from the steam generators have been terminated |
| Fuel Handling Accident | 0.063 Sv (6.3 rem) | 0.10 Sv (10 rem) | 30 days |

[a] For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steamline break analyses.

[b] The control room exposure period is 30 days for all accidents.

[c] The column labeled "Analysis Release Duration" summarizes the assumed radioactivity release durations identified in the individual appendices to this guide. These appendices contain complete descriptions of the release pathways and durations.

[d] A graded, risk-informed, and performance-based framework has been established for the control room dose criteria. The framework is applicable to the MHA LOCA.

[e] Tube rupture in the affected steam generator may result in the need to control the steam generator water level using steam dumps. These releases may extend the duration of the release from the affected steam generator beyond the initial isolation.

**4. Dose Calculation Methodology**

The NRC staff has determined (e.g., in "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants: Final Rule" (61 FR 65157; December 11, 1996)) that there is an implied synergy between the ASTs and TEDE criteria and between the TID-14844 source terms and the whole-body and thyroid dose criteria~~.~~ (Ref. 49). The TEDE criteria will not be used with results calculated from TID-14844. The guidance in this ~~regulatory position~~Regulatory Position applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67 and 10 CFR Part 52. The regulatory position also provides guidance for determining control room and offsite doses and the control room and offsite dose acceptance criteria. Certain selective implementations may not require dose calculations, as described in Regulatory Position 1.3 of this guide.