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Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants

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General Comment

Table 7 of DG-1389 includes dose criteria for all postulated DBAs, except design basis LOCA. Why is there no design basis LOCA dose assessment and separate criteria?

1. The MHA LOCA is a beyond design basis event and is a useful metric for reactor siting. However, given likely differences in event probabilities, the MHA LOCA should not be used to "bound" the design basis LOCA. DBAs are not allowed to digress to severe accidents involving core melt.

2. The design basis LOCA is the "design basis" for many Technical Specification Limiting Conditions of Operation (LCO) and Limiting Safety System Settings (LSSS) parameters. How can changes to these TSs be assessed with no consideration of radiological consequences?

3. The design basis LOCA is the "design basis" for many safety-related SSCs (e.g., reactor fuel, ECCS, containment, containment isolation, containment sprays). Many of these SSCs are designed to limit core damage during and following a LOCA. Yet, the degree of core damage (i.e., number of fuel rod failures, releases from failed fuel rods) is not predicted nor judged against regulatory criteria. For all other DBAs, the performance of safety-related SSC is judged against core damage and unique radiological limits. How can changes to these SSCs be assessed with no consideration of radiological consequences?

4. Other countries (e.g., Germany) have established dose criteria for design basis SBLOCA and LBLOCA, recognizing differences in event probability (break size) and differences from MHA LOCA. Why has the NRC not adopted this same regulatory position?

5. Below are three examples illustrating how the lack of a design basis LOCA dose criteria has resulted in unintentional consequences.
a. Several of the advanced reactor designs (e.g., AP1000) rely upon passive systems to provide emergency coolant and reflood/refill the reactor following a major pipe break. For smaller pipe breaks, the plants are designed to quickly depressurize the RCS which allows the flow of emergency coolant into

the core. However, rapid RCS depressurization increases the stress on the cladding due to differences between internal rod pressure and RCS pressure. This increased stress will likely result in a larger number of fuel rod failures (balloon/burst). Hence, the act of depressurizing the RCS increases the radiological source term and the dose to the public. Had a design basis LOCA limit been established, this problem would have been recognized.

b. Fuel designs and fuel utilization has evolved significantly over the past few decades. Several of these changes would promote a larger population fuel rod failures (balloon/burst) during a LOCA. For example, fuel rod cladding wall thickness has been reduced and cladding materials are weaker (e.g., SRA Zry-4 has a higher yield strength than RXA M5). Rod internal pressure is allowed to exceed system pressure (e.g., PWR operating system pressure of 2250 psi). Power uprates and higher discharge fuel burnup. Had a design basis LOCA limit been established, these changes would have been properly assessed against their impact to public dose. Since MHA LOCA dose assessment involves a substantial core melt, it is much less sensitive to fuel design.

c. As described in Section 3.2 of DG-1389, the potential impact of FFRD on the MHA LOCA source term was evaluations and found to be insignificant. However, this same conclusion would not be possible for a design basis LOCA. Research Information Letter (RIL) 2021-13, Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup, describes significant quantities of transient fission gas release produces as a result of FFRD. This additional source of radionuclides would likely increase public dose under design basis conditions. Furthermore, future regulatory decisions regarding FFRD would also likely be steered in a different direction in the presence of a design basis LOCA dose assessment.

6. Design basis LOCA would appear to have the same probability of occurrence as a PWR CRE since both involve failure of robust, passive structures. Yet, PWR CRE is only allowed “well within” (i.e., 25%) of the upper dose limit. Please explain these differences.