Treatment of Reactor Systems within Draft Regulatory Guide 1.183 DG-1199

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In this note, a review of concerns relevant to Draft Regulatory Guide 1.183 rev.1 (DG-1199) is presented. These comments pertain to the treatment of the main steam line isolation valve (MSIV), emergency core cooling system (ECCS) and engineered safety features (ESF) during postulated accident scenarios contained within the regulatory guide. These comments are particularly salient to the mitigation and decontamination of the full core source created during a loss of coolant accident (LOCA) for boiling water reactors (BWRs).

Summary of Methodology and Recommendations in "BWR Steam Line Radionuclide Distribution following a DBA LOCA" (Metcalf and Perez, 2010)

Comment 103 of the public comments on Regulatory Guide 1.183 DG-1199 criticizes the choice of the steam dome aerosol concentration in lieu of the drywell concentration, which was proposed in Draft Regulatory Guide 1.183 DG-1081. The paper "BWR Steam Line Radionuclide Distribution following a DBA LOCA," by Metcalf and Perez (2010), provides a rationale for the method contained within DG-1081.

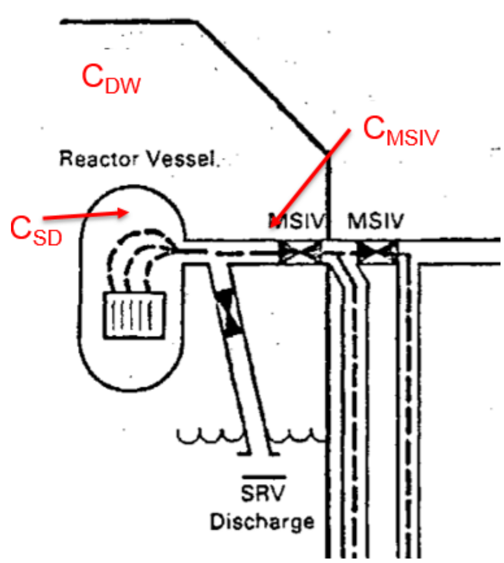

Figure 1: Conceptual view of steam lines and the MSIV in a BWR (SAND-2008-6601)

Metcalf argues that aerosol deposition and the absence of mixing in the steam line are sufficient, such that using the drywell aerosol concentration as a surrogate for the concentration at the MSIV is both bounding and conservative. Accordingly, he states that the concentration of aerosols in the drywell is higher than that just ahead of the inboard MSIV within the steam lines. In the paper, it is acknowledged that this treatment is not in fact the most physical treatment but claimed as a conservative surrogate for the actual situation. (Metcalf and Perez, 2010)

Metcalf then examines the total deposition that can occur in the main steam line before the MSIV in order to provide further justification. Using a two volume calculation, Metcalf calculates a decontamination of ~80% in the first control volume and a decontamination of ~60% in the second control volume. The leads to an overall removal efficiency of ~92% for the system. This, Metcalf claims, is higher than the reduction in leakage as a result of drywell sprays, which is claimed to be a factor of 5 to 6. Per this result he argues that it would be conservative to credit either deposition in the main steam lines or drywell sprays, but not both. (Metcalf and Perez, 2010)

Expected Decontamination Factor in a Horizontal Main Steam Line

Decontamination factors for aerosol deposition in a main steam line under conditions similar to those expected in an accident scenario were calculated using the same methodology found within the severe accident analysis code MELCOR. Two separate cases were examined and are included within Gelbard (2017). The first is a single volume which assumes a well-mixed pipe. This is shown in Figure 2. The second is a plug flow, which provides the limit as the number of control volumes in the main steam lines. See Figure 3. Both calculations were for 5.0 m long main steam line sections. (Gelbard, 2017)

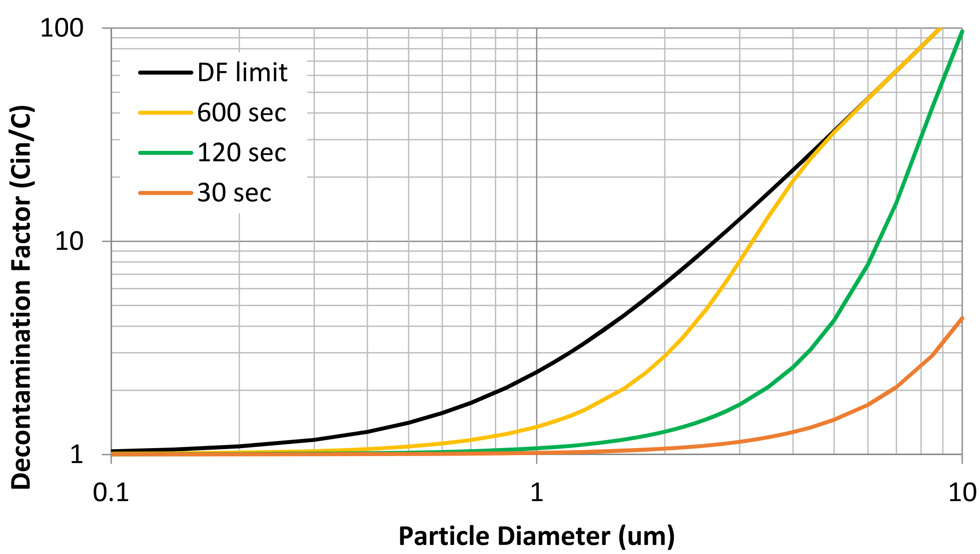

Figure 2: Decontamination factor for a well-mixed pipe with a constant flow rate and aerosol undergoing gravitational settling. (Gelbard, 2017)

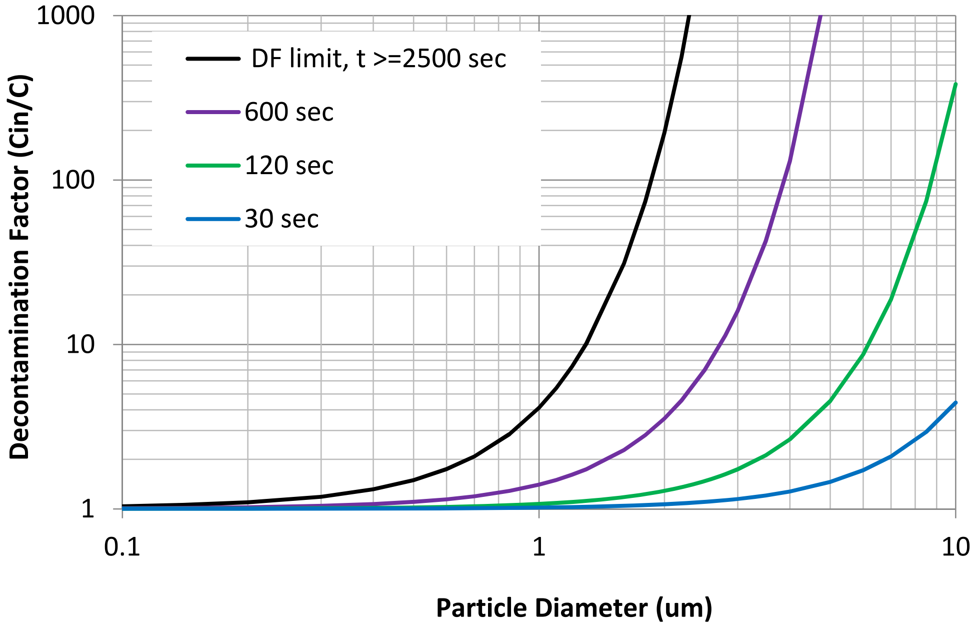

Figure 3: Decontamination factor for plug flow aerosol settling. (Gelbard, 2017)

It can be seen that both of these decontamination factors are highly dependent on the size of the aerosol particles in question. Recommended distributions from the OECD/NEA "State-Of-The-Art Report on Nuclear Aerosols" and the distribution used by Metcalf and Perez (2010) are presented in Figure 4. (Allelein et al, 2009) (Kissane, 2008)

The "State-of-the-Art Report on Nuclear Aerosols", recommends a lognormal distribution with a range of Aerosol Mass Median Diameter (AMMD) from 1.0 to 2.0 μm with a geometric standard deviation (GSD) of 2.0. (Allelein et al, 2009) Metcalf and Perez (2010) used the aerosol concentration presented in "Assessment of radiological consequences for the Perry pilot plant application using the revised (NUREG-1465) source term," issued as technical report AEB-98-03. It is uniform between equivalent aerosol diameters between 1.5 and 5.5 μm; the conversion of this to aerodynamic diameter is covered in Gelbard (2017).

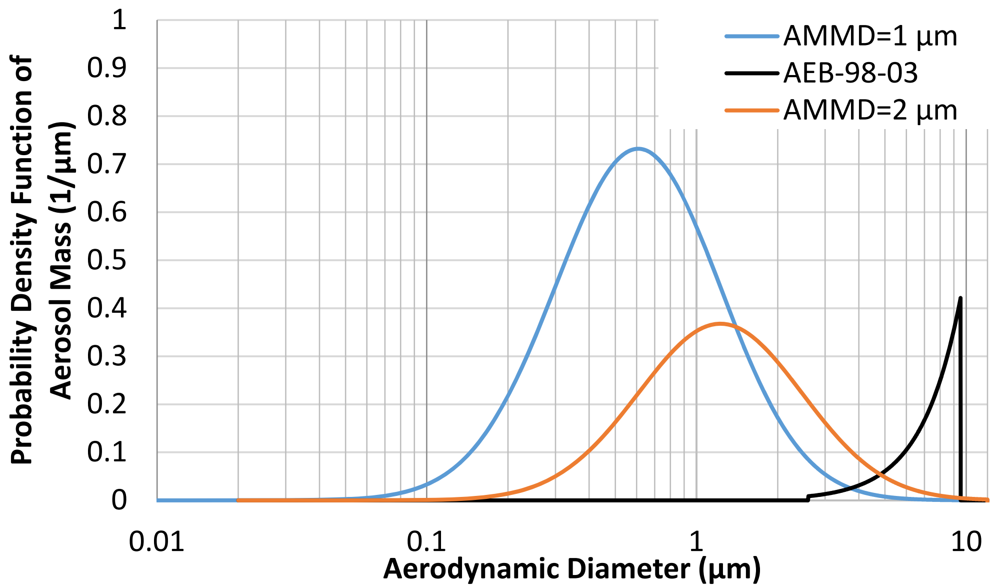

Figure 4: PDF of initial aerosol concentration, showing two separate concentrations based on the State-of-the-Art Report on Nuclear Aerosols (Allelein et al, 2009), and one based on the AEB-98-03 distribution (Schaperow, 1998).

When applying the decontamination factors calculated using the methodology contained within MELCOR, it is found that the overall decontamination factors for the 1 μm AMMD and the 2 μm AMMD distributions are significantly less than decontamination factor claimed in Metcalf and Perez (2010). However, if the aerosol size distribution from AEB-98-03 (Schaperow et al, 1998), which was used by Metcalf and Perez (2010), is applied, then a higher decontamination factor is calculated.

The decontamination factors in a well-mixed pipe for particle of diameters 1 μm and 9 μm are near 2.4 and 100 respectively. Accordingly, the choice of aerosol size distribution is key in determining the overall decontamination factor that can be credited in main steam line pipes. If the limit of plug flow is examined, a much higher decontamination factor can be obtained for larger diameter particles. However, smaller diameter (below 1 um) aerosol particles cannot achieve a DF above 4.1. (Gelbard, 2017) The DF limits for six separate cases using "State-of-the-Art" size distributions and AEB-98-03 distribution can be seen in Table 1.

Table 1: Determination of Maximum DF from various aerosol size distributions and decontamination factor distributions, DF is based on aerosol mass

Distribution	AMMD = 1.0 μm GSD = 2.0		AMMD = 2.0 μm GSD = 2.0		Uniform AEB-98-03	
Pipe Parameter	Well Mixed	Plug Flow	Well Mixed	Plug Flow	Well Mixed	Plug Flow
Maximum DF by Aerosol Mass	2.3	3.2	4.6	9.7	63.2	3.9×10^6

Recommendations:

- It is recommended that the aerosol size distributions detailed by State-Of-The-Art-Report on Nuclear Aerosols be used for nuclear accident applications, since this report contains the most up-to-date information. Therefore, the lower decontamination factors in SAND2008-6601 and RG1.183 DG-1199, compared to Metcalf and Perez (2010), are a better estimate for the postulated LOCA and the conservative case.
- It is maintained that the best representation for the concentration of aerosols in the steam lines before the inboard MSIVs is the concentration of the steam dome multiplied by the appropriate removal coefficient, which was calculated to be at a maximum a factor of 2.2 for the first two hours of the postulated accident and 0.0 afterwards. Within SAND2008-6601 it was recommended that removal coefficient for the first two hours be conservatively 0.0, since revaporization and resuspension needed to be accounted for.
- In order to provide the most accurate prediction of aerosol concentration before inboard MSIVs, a standalone steam line model with representative geometry would need to be implemented.

Main Steam Line Isolation Valve (MSIV) Closure Timing

Within comment M-5 of the public comments, it is stated that:

> Although this has nothing to do with this section/accident, the Sandia MELCOR report indicated that the inboard MSIV closes in 3 seconds and the outboard MSIV closes 3 seconds later. This puts the outboard valve closure at 6 seconds, which is LONGER than the Tech Spec allowed value.

The offset in the closure timings of the inboard and outboard MSIVs was done within SAND2008-6601 in order to create and evaluate a conservative, limiting case for MSIV leakage condition. If it is assumed that both the inboard and outboard MSIVs were closed simultaneously then a “plug” of high pressure aerosols and gas is trapped between the MSIVs. The high pressure “plug” then pushes aerosols out of both the inboard and outboard MSIVs. However, if the inboard MSIV closes before the outboard MSIV then this high pressure “plug” is not created between the MSIVs. This leads aerosols trapped between the MSIV to only leak out of the outboard MSIVs, not both of the valves. Given that the aerosols are leaking only out of the outboard MSIV; this creates the limiting condition.

Recommendation:

- The current treatment of MSIV closure timing is sufficient and conservative and should not be revised.

Emergency Core Cooling System (ECCS) and Engineered Safety Feature (ESF) Credit for Decontamination

Multiple public comments, including but not limited to comments 99 and M-3, provided criticism on SAND2008-6601 for not using (crediting) qualified ECCS and other ESFs for the mitigation of the specified source term. The comments argued that since ECCS and other ESFs were qualified to operate during accident conditions the utility should be allowed to credit them for the mitigation of the specified source term.

Operating ECCS in a plant would lead to a decrease in the total amount of core damage and thus result in a decreased source term. Additionally, ECCS operation would decrease the aerosol source that moves through the main steam lines to the MSIVs and result in a lower dose within the control room. However, unless ECCS were operated after the entire core were degraded, it is non-physical to credit ECCS for decreasing the aerosol source and not credit it for preventing core damage.

It is believed that the decision on whether or not this system can be credited falls clearly within the purview of the regulator. In particular, this issue was addressed in 1963 when San Onofre requested credit for ECCS and was subsequently denied by the Advisory Committee on Reactor Safeguards (ACRS).

Fukushima Daiichi Units Measured Control Room Dose

Since the aerosol leakage through the MSIV is the primary contributor to operator dose within the control room during a postulated accident, the Tokyo Electric Power Company (TEPCO) was contacted for information on the dose that was measured during the control room during the accidents at Fukushima Daiichi. Information from TEPCO was provided on the "shine" dose in the joint control rooms for Units 1 & 2 and for Units 3 & 4.

- In the Unit 1 & 2 joint control room, the dose was between 1 mrem/hr and 350 mrem/hr. Within roughly 25 hours (or less), a dose of 5 rem would be reached. (Mizokami, 2016)
- In the Unit 3 & 4 joint control room, the dose was between 350 mrem/hr and 1500 mrem/hr. Within roughly 10 hours (or less), a dose of 5 rem would be reached. (Mizokami, 2016)

The accident scenarios at Fukushima Daiichi showed a similar amount of core damage to the postulated LOCA with a significant amount of core damage that is used in Regulatory Guide 1.183.

References:

Hans-Josef Allelein, Ari Auvinen, Joanne Ball, Salih Gu¨ntay, Luis Herranz, Akihide Hidaka, Alain Jones, Martin Kissane, Dana Powers, and Gunter Weber, “State-Of-The-Art Report on Nuclear Aerosols,” Technical Report NEA/CSNI/R(2009)5, Nuclear Energy Agency/Committee on the Safety of Nuclear Installations (NEA/CSNI), December 2009.

Gauntt, R. O., Radel, T., Salay, M. A., and Kalinich, D. A., “Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD,” Issued as SAND2008-6601, October 2008.

Gelbard, F. and Andrews, N., “Aerosol Removal by Gravitational Settling in a Horizontal Steam Pipe,” November, 2016.

Kissane, M. P., “On the nature of aerosols produced during a severe accident of a water-cooled nuclear reactor,” Nuclear Engineering and Design, 238:2792-2800, 2008.

Metcalf, J. E. and Perez, P. B., “BWR Steam Line Radionuclide Concentration Distribution following a DBA LOCA,” 31st Nuclear Air Cleaning Conference, Charlotte, July 2010.

Mizokami, Shinya and Yamauchi, Daisuke; Technical communication with Tokyo Electric Power Company, November 2016.

Salay, Michael, “Aerosol Size in Main Steam Lines”, U.S. NRC Internal Presentation, December 2014.

Schaperow, J., Gingrich, C. and Ridgely, J., “Assessment of radiological consequences for the Perry pilot plant application using the revised (NUREG-1465) source term,” Issued as Technical Report AEB-98-03, NRC/RES/DST/AEB, 1998.

U.S. Nuclear Regulatory Commission, Public Comments on Regulatory Guide 1.183 rev. 1, Draft issued as DG-1199, December 2011.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, Draft issued as DG-1081, July 2000.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183 rev. 1, Draft issued as DG-1199, December 2011.