Summary Report for the June 5, 2019 Research Assistant Request
Regulatory and Technical Basis for An
Acceptable Fuel Handling Accident Transport Model

The purpose of this report is to document the regulatory and technical basis for an alternative design-basis accident (DBA) fuel handling accident (FHA) analysis methodology to Appendix B of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."[1]

Section 2 provides a general description of the alternative method to the current Regulatory Guide 1.183 FHA. This section also provides a brief history of the use of "regulatory source terms" and a description of the environmental conditions in which fuel handling operations are taking place. Section 3 provides case study results which analyzed the radiological consequences of the alternative method and discusses the impacts on licensees' DBA FHA analyses of record.

1.0 Background

FHAs are analyzed to assess the risk to public health and safety resulting from the operation of the facility and to demonstrate compliance with the various numerical radiological criteria set forth in regulation and subsequent guidance. The primary purpose is to evaluate the design basis of systems, structures and components that mitigate radiological releases to the environment. As of late, the FHA has become one of the most common DBA radiological dose analyses evaluated in licensing actions. Due to the conservative modeling assumptions currently acceptable to the staff, as described in RG 1.183, evaluation of the FHA can require significant staff resources, and in some cases has become a limiting accident, despite its low safety and risk significance.

The technical basis for the fission product transport model is largely contained in, and unchanged from, studies reported from the early 1970s. Staff has confirmed the degree of conservatism within the methods employed in Regulatory Guide 1.183 after a recent staff re-evaluation of these studies.

By Informal Assistance Request (IAR) dated August, 17, 2018, the Office of Nuclear Regulatory Regulation (NRR), Division of Risk Assessment (DRA), Radiation Protection and Consequence Branch (ARCB), requested the Office of Nuclear Regulatory Research (RES), Division of Systems Analysis (DSA), Fuels and Source Term Code Development Branch (FSCB), to perform an independent review of the staff re-evaluation of the fission product release and transport model for the FHA DBA described in Regulatory Guide 1.183, Appendix B.[2] The main objective of this re-evaluation was to revisit the original studies forming the technical basis for the FHA dose analysis methodology and seek input in updating the model with current information and practices consistent with other DBAs.

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[1] Agencywide Documents Access and Management System (ADAMS) Accession Number ML003716792
[2] ADAMS Accession Number ML19274C045

RES's IAR response confirmed the staff's re-evaluation and recommended improvements to the fission product transport model.[3] These improvements would be established from our current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place. The staff then developed the updated FHA fission product transport model under these conditions. Several staff-level meetings and presentations were held to discuss the re-evaluation findings and potential impact that the updated FHA transport model would have on staff and licensees. A follow-on Research Assistance Request (RAR) dated June 5, 2019, NRR requested RES to finalize the FHA transport model, have it peer-reviewed, and formally submit it to NRR through an inter-office memo as an acceptable method for staff use.[4] Note that Sandia National Laboratory (SNL) and other experts in the field were contacted to aid the staff in the development of a technical basis for supporting this alternative method to the FHA.

The alternative FHA in-pool fission product transport model modifies the existing FHA boundary conditions. It is based on the environmental conditions in which fuel handling operations are taking place; incorporating several improvements of our current understanding of reactor fuel pin physics, iodine chemistry and re-evolution while maintaining conservatism.

2.0 Alternative Method for DBA Fuel Handling Accident Dose Analyses

2.1 Regulatory History of the Fuel Handling Accident

Use of regulatory source terms in DBA assessment is deeply embedded in the regulatory policy and practices of the NRC, even as the licensing process has evolved over the past 50 years. The source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release. It is based upon the concept of defense-in-depth in which power plant design, operation, siting, and emergency planning comprise independent layers of nuclear safety.

This approach encourages nuclear plant designers to incorporate several lines of defense in order to maintain the effectiveness of physical barriers between radiation sources and materials from workers, members of the public and environment in operational states and, for some barriers, in accident conditions. It centers on the concept of DBAs, assessment of which aims to determine the effectiveness of each line of defense. The DBAs establish and confirm the design basis of the nuclear facility, including its safety-related structures, systems and components, and items important to safety; ensuring that the plant design meets the safety and numerical radiological criteria set forth in regulation and subsequent guidance. From this foundation, specific safety requirements have evolved through a number of criteria, procedures and evaluations, as reflected in regulations, Regulatory Guides, standard review plans,

[3] ADAMS Accession Number ML19270E016
[4] ADAMS Accession Number ML19113A152

technical specifications, license conditions, and various regulatory technical information documents.

FHAs are analyzed to assess the risk to public health and safety resulting from the operation of the facility and to demonstrate compliance with various regulatory requirements. The primary purpose is to evaluate the design basis of systems, structures and components that mitigate radiological releases to the environment, to include such items as ventilation system design, filter efficiencies, and primary or secondary containment penetration closure times, for example. An illustrative accident sequence consists of the dropping of a nuclear fuel assembly during refueling operations, resulting in the non-mechanistic assumptions of complete and instantaneous shearing off of all fuel pins on the dropped assembly, release of a portion of the volatile fission gases from the damaged fuel rods, transport of soluble and insoluble gases through the water of the spent fuel or reactor pool, absorption of soluble gases in the spent fuel or reactor pool water, and release to and transport through the environment.

The technical basis for the current FHA fission product transport model is largely contained in two studies dating back to the 1970s. The first is a proprietary topical report by Westinghouse, WCAP-7518-L, "Topical Report Radiological Consequences of a Fuel Handling Accident," (Proprietary) (1970) (hereinafter referred to as the WCAP), reporting on a series of large and small scale experiments designed to be prototypic of an accident during refueling operations.[5] The non-proprietary version of WCAP-7518-L is WCAP-7828 (1971), "Radiological Consequences of a Fuel Handling Accident."[6] The primary purpose of this study was to measure the iodine scrubbing effect from gas bubbles to the water, commonly referred to as the iodine decontaminant factor (DF). At that time, the Atomic Energy Commission (AEC) staff reviewed and ultimately chose to not adopt the WCAP recommendations. Instead, the staff reported (Burley, 1971) (hereafter referred to as the Burley study) on the development of a theoretical iodine DF model which incorporated select measurement inputs from the Westinghouse study.[7]

In November 1999, the NRR staff performed an evaluation in response to a request from NRC Region II in Task Interface Agreement, TIA 99-03, "Potential Non-Conservative Assumptions for Fuel-Handling Accidents," of whether the assumptions used in the Burley analysis were conservative for high burnup fuel.[8] The evaluation concluded that adequate conservatism was provided by the analysis assumptions. In particular, the evaluation found that an effective spent fuel or reactor pool iodine DF of 100 is likely overly conservative; therefore, the staff subsequently qualitatively increased it to a value to 200 in Regulatory Guide 1.183. The staff also updated the previous iodine speciation gap fraction break down of 99.75% elemental and 0.25% organic from the original Safety Guide 25 which is now superseded by Regulatory Guide

[5] The non-proprietary version of WCAP-7518-L is WCAP-7828, "Radiological Consequences of a Fuel Handling Accident," and can be found under ADAMS Accession Number ML18025B589
[6] ADAMS Accession Number ML18025B589
[7] ADAMS Accession Number ML16357A003
[8] ADAMS Accession Number ML993340497

1.25 “Assumptions Use for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.”[9]

The staff elected to specify the iodine gap fraction released to the spent fuel or reactor pool as effectively 99.85% elemental and 0.15% organic without re-computing the effective iodine DF based on the Burley method. The recomputed effective iodine DF equates to 667. For a spent fuel or reactor pool depth of 23 feet or greater, to give an overall iodine effective DF of 200, the assumed spent fuel or reactor pool elemental iodine DF is 500 and organic DF is 1, with resulting iodine speciation of the release from the spent fuel or reactor pool as 57% elemental and 43% organic. These speciation values are currently recommended in Regulatory Guide 1.183. Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” which is based on methods prior to the publication of 10 CFR 50.67, retained the Regulatory Guide 1.25 assumption of 99.75% elemental iodine and 0.25% organic iodine species in the fuel gap, but revised the overall effective iodine DF to 200 if the depth of water above the damaged fuel is 23 feet or greater, consistent with the spent fuel or reactor pool DF guidance in Regulatory Guide 1.183.

During the 473rd ACRS Full-committee meeting, held June 7th, 2000, the committee members expressed concern with the staff’s treatment of iodine speciation and the continuation of modeling the FHA as a puff release. This is because iodine released from the fuel pin gap was primarily being considered to be in the form of gaseous iodine as I_2 and CH_3I and released to the environment instantaneously. In reality, most of the iodine is likely a solid, as CsI, at the time of the postulated FHA and therefore not available for instantaneous release. Rather, it is readily absorbed in the spent fuel or reactor pool water and slowly re-evolved over a long period of time. The staff explained that the chosen DF of 200 represented not only the iodine available for immediate release following the postulated fuel rupture but also represented re-evolution of iodine. This re-evolution release was assigned to the immediate release for the purposes of modeling simplicity. However, current research has shown that iodine re-evolution can potentially be significant and thus should be explicitly modeled.

2.2 General Fuel Handling Conditions

The fission product release from the breached fuel is based on the environmental conditions in which fuel handling operations are taking place. Under these conditions, the bulk spent fuel pool water temperature is normally maintained between 100 to 125°F, depending on the facility design. When a full-core offload of fuel is required, this temperature will rise due to the higher decay heat generation of the freshly discharged fuel. Depending on the amount of fuel discharged and the time since shutdown, the spent fuel or reactor pool temperature may rise to as much as 150°F. The spent fuel or reactor pool acidity is also controlled for the purposes of criticality safety where pH values of around 5 are generally held; values less than 7 need to

[9] ADAMS Accession Number ML083300022

consider the re-evolution of iodine along with the concentration of suspended organics. As such, the initial conditions consider, among others; a time period between power operation and the movement of recently irradiate fuel to account for both radioactive decay, less decay power; the use of spent fuel or reactor pool water temperature to determine internal gas temperature and pressure; and, the availability of iodine to re-evolve.

The iodine gap activity in the damaged rods is assumed to be released in two stages. The chemical form of radioiodine released from the fuel to the spent fuel or reactor pool water should be assumed to be 95 percent CsI, 4.85 percent I_2, and 0.15 percent organic iodide. The first stage is the instantaneous gaseous release from the fuel gap in rising bubbles where I_2 and organic iodine are conservatively assumed to be in vapor form and subsequently decontaminated by passage through the overlying spent fuel or reactor pool of water into the building atmosphere. This activity is then vented to the environment over a 2-hour period. The second stage is the protracted release initiated 2 hours (following the initial gaseous release) following the fuel bundle drop. The CsI in the fuel gap of the damaged assembly is conservatively assumed to completely dissociate into the spent fuel or reactor pool water then slowly re-evolve into the building atmosphere as I_2 due to the low spent fuel or reactor pool water pH. This activity is vented directly to the environment for a period of thirty days following 2 hours after the initial instantaneous gaseous release.

A reduction in the amount of radioactive material released from the spent fuel or reactor pool by engineered safety feature filter systems or spent fuel or reactor pool cooling and cleanup systems may be taken into account, provided these systems meet the guidance of Regulatory Guide 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” and Generic Letter 99-02, “Laboratory Testing of Nuclear-Grade Activated Charcoal,” or RG 1.13, “Spent Fuel Storage Facility Design Basis.” [10,11,12]

2.3 Non-Loss of Coolant Accident (LOCA) Fuel Pin Gap Inventories

Non-LOCA gap inventories represent radioactive fission products generated during normal steady-state operation that have diffused within the fuel pellet, have been released into the fuel rod void space, and are available for release upon fuel rod cladding failure. This inventory should represent the limiting fuel rods in the reactor core at the most limiting time in life. Regulatory Guide 1.183, Regulatory Position 3, Table 3, provides acceptable assumptions regarding core inventory and the release of radionuclides from the fuel. Like-wise, proposed Regulatory Guide 1.183[13], Revision 1, Appendix J, would provide an acceptable analytical technique to compute cycle-specific non-LOCA fuel pin gap inventories.[14] Appendix J provides an example calculation to illustrate the potential improvement in radiological source term

[10] ADAMS Accession Number ML12159A013
[11] https://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1999/gl99002.html
[12] ADAMS Accession Number ML070310035
[13] Also referred to as Draft Guide 1199, ADAMS Accession Number ML090960464
[14] ADAMS Accession Number ML090960464

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 Regulatory Guide 1.183, Table 3.

2.4 First Stage – Instantaneous Release

An overall iodine DF is a function of bubble size and rise time through the water column, both of which are functions of fuel pin pressure. If the water depth is 19 feet or greater, an overall effective iodine DF for I_2 and organic iodine can be computed based on a best-estimate rod pin pressure for the limiting fuel rods in the reactor core at the most limiting time in life. The time period between reactor shutdown and the movement of fuel may be used to compute radioactive decay and less decay power. The use of spent fuel or reactor pool water temperature based on a full-core offload may be used to determine internal gas temperature and thus pin pressure.

To compute rod bounding internal pressures, the staff used the NRC maintained FRAPCON fuel rod thermal-mechanical fuel performance code by modeling an assembly with an aggressive power history that maximizes at-power rod internal pressure. The initial conditions of the assembly are based on a 24-hour time period following and assuming a spent fuel or reactor pool water temperature of 150°F, simulations predicted a bounding end-of-life rod internal pressure of 760 psig.

The first stage assumes I_2 and organic iodine are in vapor form and are decontaminated by passage through the overlaying spent fuel or reactor pool of water. An overall iodine DF based on a pin pressure is computed as follows:

$$DF_I = 81.046e^{0.305(t/d)} \quad \text{(Equation 1)}$$

where:

t = bubble rise time (sec), computed as a function of pin pressure, x (psig), as:

$$t(sec) = 9.2261e^{-6E-4*x} \quad \text{(Equation 2)}$$

d = bubble diameter (cm), computed as a function of pin pressure, x (psig), as:

$$d(cm) = -0.0002 * \mathrm{x} + 1.0009 \quad \text{(Equation 3)}$$

A bounding rod internal pressure of 760 psig computes an effective iodine DF of 662.

2.5 Second Stage – Protected Release

Iodine present in the gap is expected to end up in the spent fuel or reactor pool. Although it is assumed to occur instantaneously, there is likely some delay. Cesium iodide, the predominant expected form of iodine in the gap, is not only hygroscopic (readily absorbs water even from atmosphere) but also deliquescent (absorbs water vapor from the atmosphere to the extent that it forms a solution). Most other forms of iodine also readily dissolve in water and many are hygroscopic and deliquescent. It would be expected that any iodine-containing particles that get trapped in the spent fuel or reactor pool would readily be dissolved. If liquid water penetrates the fuel-clad gap after rupture, which could be expected, solid CsI and I_2 remaining on clad/fuel surfaces in the gap would be likewise readily dissolved. Even if only water vapor penetrates into some parts of the fuel-clad gap, a solution will form, the solid CsI and I_2 would dissolve, and the iodine still would be expected to eventually end up in the spent fuel or reactor pool if it does not decay first. The amount of iodine originally in the fuel pin gap in solid form that transfers to the spent fuel or reactor pool before decay or before the end of the calculation period depends on the rate of release. Near spent fuel or reactor pool temperatures, conservatively assumed for this analysis to be 150°F, CsI would be solid, I_2 is solid, and CH_3I is a liquid. For the purposes of this analysis the iodine in the fuel pin gap deterministically transfers to the spent fuel or reactor pool water over a two hour period.

Two sets of iodine re-evolution calculations were performed. The first set consists of numerical calculations that account for the change in the volatile iodine fraction based on the change in concentration in the spent fuel or reactor pool. It considers the speciation of iodine that reduces the fraction available for release to the environment, and radioactive decay. It does not consider potential surface deposition of iodine, organic iodide formation with organics present in water, or filtration of the spent fuel or reactor pool water. The second, simplified set of calculations, neglects this change so the evolution removal coefficient remains constant which makes it easy to solve analytically and is therefore recommended for estimating re-evolution from spent fuel or reactor pools for the purposes of DBA FHA analyses.

The mass-transfer coefficient developed for iodine in a spent fuel or reactor pool of water applies the surface-film-renewal model based on the THAI experiments.[15] The THAI-23 experiment was conducted for the very purpose of evaluating the mass-transfer coefficient of iodine from spent fuel or reactor pools. The experiments are intended to bound possible iodine evolution scenarios by considering both stagnant and recirculating conditions. It involves a model-based mass-transfer coefficient, considers the partition coefficient, and refuel floor volume in the determination of net transfer rate of iodine to the gas space. The calculation considered the speciation of iodine that reduces the fraction available for release to the environment, and radioactive decay.

The liquid-phase mass transfer coefficients for the THAI test using a surface renewal model as follows:

[15] Fischer, K. W. (2012). *Experimental Determination and Analysis of Iodine Mass Transfer Coefficients from THAI Test Iod-23.* Cologne (Germany): 5th European Review meeting on Severe Accident Research (ERMSAR-2012).

$$k_L = \sqrt{\frac{D_{IL}}{\pi t_c}}, \quad \text{(Equation 4)}$$

where k_L is the liquid-phase mass transfer coefficient, D_{IL} the diffusivity of I_2 in the liquid and, t_c, is the contact time between the liquid and the gas surface.

The overall mass transfer coefficient based on the liquid phase is then:

$$K_L = \left[\frac{1}{k_L} + \frac{P_c}{k_G}\right]^{-1}, \quad \text{(Equation 5)}$$

where K_L (upper case K) is the overall mass transfer coefficient based on the liquid phase, k_G is the gas-phase mass transfer coefficient, and P_c is the partition coefficient which describes the ratio of the I_2 concentration in the liquid to the I_2 concentration in the gas at equilibrium conditions.

The evolution rate depends on the total iodine (radioactive + non-radioactive) in the spent fuel or reactor pool. Since the evolution rate is relatively low and most of the iodine in the spent fuel or reactor pool is non-radioactive and thus does not decay, the overall change in evolution rate does not change significantly over the course of the scenario.

With the overall mass transfer coefficient known, in conjunction with facility-specific parameters describing the spent fuel or reactor pool, the simplified approach to compute the iodine re-evolution coefficient from the spent fuel or reactor pool, λ_e, can be evaluated as follows:

$$\lambda_e = K_L \, X_e \left(S_{pool} \, / \, V_{pool}\right) . \quad \text{(Equation 6)}$$

Where X_e is the volatile fraction of iodine atoms in I_2 form, S_{pool} is the spent fuel or reactor pool surface area and V_{pool} is the spent fuel or reactor pool volume.

Finally, the re-evolution from the spent fuel or reactor pool to the surrounding building can be modeled as a volumetric flow rate Q_e by assuming a simple control volume representing the spent fuel or reactor pool dimensions and flow rate as follows:

$$Q_e = \lambda_e V_{pool}. \quad \text{(Equation 7)}$$

3.0 Impacts on Licensees' DBA FHA Doses

A case study was performed to analyze the radiological consequences of the proposed DBA FHA using the alternative analysis methodology. The purpose was to determine the impact of the proposed alternative methodology for the DBA FHA by comparing the computed radiological consequences to the licensing basis FHA analyses and whether the revised results would

exceed the radiological accident dose criteria of 10 CFR 50.67 and the FHA-specific dose acceptance criteria listed in Regulatory Guide 1.183 and Standard Review Plant Chapter 15.0.1.

A survey of operating plant Updated Final Safety Analysis Reports and recent Alternative Source Terms license amendments was conducted to review the various facility licensing- and design-bases and identify important modeling parameters. Models were developed based on the updated FHA transport model using the NRC maintained Symbolic Nuclear Analysis Package/RADionuclide Transport, Removal And Dose Estimation (SNAP/RADTRAD) dose analysis computer code.[16] The code is used to estimate transport and removal of radionuclides and determine DBA radiological doses at the exclusion area boundary, the low population zone, in the control room, and other locations of interest. Estimated doses at all three receptors are 91-98% lower than the licensee's current licensing bases. The impact of credit for various spent fuel or reactor pool cooling and cleanup systems was not explicitly modeled since such systems were not credited in the surveyed licensee's DBA FHA licensing basis analyses of record.

4.0 Alternative FHA Methodology

Enclosure 2 provides a general description of the alternative DBA FHA analysis methodology similar in format to Appendix B of Regulatory Guide 1.183; it does not replace it. This alternative captures the proposed revised FHA transport model and limitations for its applicability.

5.0 Example Calculation for the Revised FHA Fission Product Release and Transport Model

Enclosure 3 provides a calculational tool to compute various input parameters for the alternative DBA FHA model.

6.0 Technical Basis for the Revised FHA Fission Product Release and Transport Model

Enclosure 4 provides a staff report providing the technical basis for the alternative DBA FHA model. The report describes the re-analysis of the original studies using modern data analysis tools to confirm results and conclusions. Errors in computation were corrected, and certain parameters were updated to be consistent with current NRC staff assumptions and practices with other DBAs. A new re-evolution model was developed based on experimental data.

The study confirmed and elaborated on the available experimental data, various reports, and staff reviews while recognizing identified limitations. The study concluded there is considerable design margin regarding the scrubbing effects of iodine in the spent fuel or reactor pool and the current staff DBA FHA model is overly conservative because iodine is primarily being considered to be in the form of gaseous iodine as I_2 and CH_3I and released to the environment instantaneously. As such, the staff developed and documented in this report the revised FHA

[16] ADAMS Accession Number ML16160AA019

transport model based on the environmental conditions in which fuel handling operations are taking place.