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A STUDY ON METHODOLOGY FOR IDENTIFYING CORRELATIONS BETWEEN LERF AND EARLY FATALITY

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The correlations between Large Early Release Frequency (LERF) and Early Fatality need to be investigated for risk-informed application and regulation. In Regulatory Guide (RG) -1.174, while there are decision-making criteria using the measures of Core Damage Frequency (CDF) and LERF, there are no specific criteria on LERF. Since there are both huge uncertainties and large costs needed in off-site consequence calculation, a LERF assessment methodology needs to be developed, and its correlation factor needs to be identified, for risk-informed decision-making. A new method for estimating off-site consequence has been presented and performed for assessing health effects caused by radioisotopes released from severe accidents of nuclear power plants in this study. The MACCS2 code is used for validating the source term quantitatively regarding health effects, depending on the release characteristics of radioisotopes during severe accidents.

This study developed a method for identifying correlations between LERF and Early Fatality and validates the results of the model using the MACCS2 code. The results of this study may contribute to defining LERF and finding a measure for risk-informed regulations and risk-informed decision-making.

KEYWORDS : Large Early Release Frequency, Early Fatality, off-site Consequence, Risk-informed Regulations

1. INTRODUCTION

The outcome of a Probabilistic Safety Assessment (PSA) for a nuclear power plant is a combination of qualitative and quantitative results. Quantitative results are shown as Core Damage Frequency (CDF) and frequency of radioactive release, e.g., as Large Early Release Frequency (LERF). In some countries, the safety authorities define these values or higher-level safety goals. In other countries, they have been defined by the nuclear utilities. Ultimately, the goals are intended to define an acceptable level of risk from the operation of a nuclear facility. It provides a tool for identifying and ranking issues with safety impact, and it includes both procedural and design related issues. Thus, CDF and LERF (safety goals) not only define an acceptable safety level, but they also have a wider and more general use as decision criteria.

The objective of using CDF and the LERF is to assess, from a risk perspective, the impact on the current licensing basis of any changes that may be proposed by various licensees. The use of CDF and LERF has been judged appropriate since the regulatory utilities have performed level-1 and level-2 PSA. Therefore, information on core damage frequency, containment failure likelihood, containment failure mode, and radiological releases, are readily available from these studies. Difficulties exist in defining "large early" release frequencies, since there are spectrums of fission product releases that are estimated as likely to occur following severe accidents, with their expected release quantities ranging over several orders of magnitude, and occurring at various times following accident initiation.

The concept of CDF and LERF has been considered as a suitable metric for making risk-informed regulatory decisions. However, the definition of "large early release" and the associated time needs to be evaluated from the regulatory perspective, and the potential implication of severe accidents. The definition of what constitutes a large early release differs greatly, and there are many parameters involved in the definition, the most important ones being the time, the amount, and the composition of the release. The underlying reason for the complexity of the release definition is largely due to the fact that the release definition constitutes the link between the level-2 PSA results, and an indirect attempt to assess health effects from the release. However, such consequential issues are basically addressed in level-3 PSA, and can only be fully covered in level-3 analysis. In order to judge the acceptability of results, various criteria for interpretation of the results, and assessment of their acceptability need to be defined. In this study, MACCS2 codes are used for evaluating the source terms quantitatively regarding health effects, depending on release characteristics of radioisotopes during severe accidents.

Section 2 describes the background of surrogate safety goals and LERF. This section contains the characteristics and the advantage of surrogate safety goals. It also includes the explanation of how to define large early release for estimation of the large release frequency. Section 3 discusses the approach to assessing the health effects. It contains a description of the robust method of calculating the dose received by an organ over a given period of inhalation and total external dose. Also, it analyses the various parameters used in carrying out the sensitivity analyses. Section 4 describes the results, which were obtained from this robust method of the calculation of the dose. It presents the threshold for source term magnitudes, which can result in lethal doses, and also shows the impact of these various evacuation assumptions on the early fatality risk.

2. DEFINITION OF LERF

2.1 The Background of Safety Goals and Surrogate Safety Goals

The Nuclear Regulatory Commission (NRC) established two goals that are stated in terms of public health risk, one addressing individual risk and the other addressing societal risk. The risk to an individual is based on the potential for death resulting directly from a reactor accident, that is, a prompt fatality. The societal risk is stated in terms of nuclear power plant operations, as opposed to accidents alone, and addresses the long-term impact on those living near the plant. In both cases, the NRC based its acceptable level of risk on a comparison with other types of risk encountered by individuals and by society from other causes, applying the rule that the consequences of nuclear power plant operation should not result in significant additional risks to life and health. The goals were expressed in qualitative terms, perhaps so the philosophy could be understood by all. In both cases, however, the NRC also expressed the qualitative goals for the safety of nuclear power plants in terms of individual and societal "quantitative health objectives" or "QHOs." These were established at one one-thousandth of the risk arising from other causes presenting the same type of risk. It is important to note that the QHOs have never been directly reflected in the NRC's regulations. QHOs were published to provide guidance as to the level of "public protection which nuclear plant designers and operators should strive to achieve". They were also meant to provide guidance to the NRC staff to use in the regulatory decision-making process. While the safety goals provided a metric to address the question of "how safe is safe enough?", practical implementation of the NRC's guidance proved to be difficult. This was the result of the large uncertainties involved in calculation of risk in the mathematical sense of Probability times Consequences. As a result, the NRC staff began looking for other metrics to use as surrogates for the QHOs in regulatory decision-making.

The Commission provided additional guidance to the staff regarding the safety goals, approving surrogate objectives concerning the frequency of core damage accidents and large releases of radioactivity. In addition, a conditional containment failure probability of one-tenth was approved for application to evolutionary light water reactor designs. This resulted in a large release frequency of one in one-hundred-thousand, since containment failure is necessary for a large release to occur. These values have evolved into the "benchmark" values of 10^{-4} for CDF and 10-5 for LERF, as discussed in Regulatory Guide 1.174 for use in risk-informed regulatory decision-making.

But, the definition of the LERF (risk metrics) can be a more complex activity, as it should be possible to relate the risk metrics to the degree of harm experienced by the population exposed to the risk. As an example, there is no simple connection of this kind between the core damage frequency for a nuclear power plant and the degree of risk experienced by the public. For level 2 PSA criteria (radioactive release), the connection is more evident, but not necessarily straightforward or easily interpreted. In contrast, safety goals for other man-made risks are often expressed in terms of frequency and number of fatalities, which usually provides safety goals, which are easier both to interpret and to apply. The F-N (frequency-fatalities of various numbers) curve approach may also be chosen for criteria related to the results of a level 3 PSA. While the F-N curve represents a high-level safety goal, the CDF and LERF criteria used for interpreting PSA results can be regarded as surrogates of the high-level safety goals. By using surrogate safety goals, which are easier to address, the role and importance of individual safety barriers can be assessed.

Once a surrogate safety goal has been defined, there is a need for an accepted procedure for carrying out the quantitative risk assessment, for applying the CDF and LERF goals to the relevant risk measure, and for acting on the outcome of the application. In case of exceeding the CDF and LERF goals, there is a need for a procedure for handling the deviation and for assessing the severity of the deviation. Thus, there is a need for defining how to decide that the CDF and LERF goals have been met, i.e., criteria for accepting a calculated risk.

2.2 The Advantage of Surrogate Safety Goals (LERF)

The QHOs are technology-neutral, risk-informed and performance-based. They do not need any reactor technology-specific parameters for implementation. They do not prescribe any particular implementation approach to how the QHOs are to be met. Therefore, defining the QHOs as a minimum level of safety is in keeping with the desirable objective of a technology-neutral, risk-informed, and performance-based metric. Use of the QHOs as the minimum level of safety is consistent with the level of safety adopted by the industry in their own design and regulatory initiatives. Use of the QHOs as the minimum level of safety would also allow the use of other risk measures to demonstrate that the QHOs have been met. But, it has a disadvantage of increasing the cost to carry out probabilistic assessments all the way to the health effect level. Two approaches are available to the licensees to demonstrate that the QHOs have been met: (1) a Level 3 PRA, i.e., a probabilistic consequence assessment in terms of health effects, or (2) other risk measures which are defined on a technology-specific basis. Assuming that appropriate subsidiary risk objectives can be established, it would eliminate the need for carrying out probabilistic assessments all the way to the health effect level, thus saving resources. Presumably, the calculation of the subsidiary risk objectives (LERF) could be accomplished with less effort than the calculation of the health effects, much like the calculation of CDF and LERF in risk assessments for current reactors requires less effort than a Level 3 consequence assessment for these reactors.

The principal advantages of defining the LERF are to clearly establish the need to balance prevention and mitigation and to provide practical guidance for implementing the safety goal quantitative health objectives. The use of Large Early Release Frequency eliminates the inherent uncertainties in Level 3 PRA calculations and represents a calculated parameter based on activities under licensees' control. Different methods were used to estimate the LERFs, based on the information provided in the IPE submittals. The methods are discussed below:

2.3 Alternative Definitions for Large Early Release Frequency

The result of a literature survey of the available alternatives for the definition of large early release, for estimation of the large release frequency following postulated severe accidents, has resulted in the evaluation of the following alternatives:

- •Alternative 1: Any release that occurs because of severe accidents that would entail early containment failure (including containment isolation failure) and containment bypass conditions.
- •Alternative 2: Any release that would exceed specific thresholds in terms of fractional releases and timing of release.
- •Alternative 3: Any release that occurs because of severe accidents that would entail 10% or more of the initial core inventory of iodine-131.
- •Alternative 4: A collection of all releases that would result in one or more early fatality consequences.

The first three alternatives are solely based on the release thresholds, and can be readily calculated using the results of the level-2 PSAs. For Alternative 1, the bypass and early containment failure frequencies reports in the Level-2 PSAs were summed to provide an estimate of the LERF. This approach simply considers early failure and bypass events without considering the magnitude of the environmental source terms. However, not all early containment failures result in large source terms and there is a threshold below which early fatalities will not occur. The threshold for early fatalities depends on several factors. In many calculations performed to determine the conditions under which an off-site early fatality could occur as a function of the fraction of the core inventory released of different radionuclide groups. A threshold of I, Cs, Te≥0.03 was determined from the spectrum of these calculations. This threshold is used as the basis for Alternative 2 below. For Alternative 2,the frequency of release categories reported in the IPEs that resulted in at least 3 percent release of I, Cs, and Te, were summed to provide an alternate estimate of LERF, based on the calculations reported above. The Alternative 3 approach is similar to Alternative 2, but uses larger release fractions to define LERF. The frequency of release categories reported in the IPEs that resulted in at least 10 percent release of I and Cs, were summed to provide another estimate of LERF. The 10 percent release fractions were used because several utilities used this threshold to define a large release in their IPE submittals. Estimation of release frequency using Alternative 4 requires either a level-3 PSA (i.e., offsite consequence calculations) or other approximations for calculation of early fatalities in the immediate vicinity of the power plants (i.e., in the area outside the exclusion zone, extending to about 2 km from the plant).

Within the current NRC regulatory framework, large early release is typically being used as a surrogate for the early fatality quantitative health objective of the NRC's safety goal policy. It is defined as "the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation."

This study has examined the implications of defining a "large early release" source term that has the potential to result in one early fatality within 1.6 kmof the plant boundary. This paper defines a simple relationship between the Safety Goal QHO for Individual Early Fatality and a Large Early Release Frequency (LERF) that can be used to estimate the Safety Goal QHO for a specific plant. This paper also provides a quantitative definition of the LERF. The relationship utilizes simple site-specific characteristics and results from a Level 2 plant-specific probabilistic risk assessment (PRA) (release category frequencies and source term characteristics).

3. MODELING METHODS

3.1 Risk-Based Regulatory Acceptance Criteria

The purpose of this section is to explore the concept of using the Safety Goal quantitative health effects (QHO) on early fatalities to derive lower tier risk acceptance criteria for application on a plant-specific basis. A starting point for expressing the early fatality QHO in a form that can be used to derive different tier criteria is the following definition for the risk of early fatalities for any specific plant in terms of the normal determinations of probabilistic risk assessments (PRAs).

$$\text{Mean number of early fatalities} \equiv \sum_i (\mathrm{STCF})_i (\mathrm{C_{ef}})_i \qquad (1)$$

where i refers to the spectrum of accident sequences

$(STCF)_i$ is the source term release category frequency for sequence i,

$(C_{ef})_i$ is the early fatality consequences given to sequence i, which has associated with it a source term, that may be defined in terms of the equivalent release of iodine-131 to the outside environment.

The QHO objective for early fatalities is expressed in terms of individual risk. The Safety Goal Policy Statement specifically states that the early fatality QHO is to be determined by calculating the cumulative individual fatalities within 1.6 km of the site boundary, C_{ef}, and dividing that by the population within that same 1.6 km region, P_1. Therefore, Equation 1, for purposes of comparing with the early fatality QHO, should be rewritten as

$$\text{Individual risk(IR)} \equiv \sum_i (\mathrm{STCF})_i (\mathrm{C_{ef}})_i / \mathrm{P_1} \qquad (2)$$

In order to proceed further, we first note that, in general, the individual early fatality (IEF) within 1.6 km of the site boundary can be related to the total effective dose expressed in terms of the equivalent release of iodine-131 by the relationship

$$\mathrm{IEF_{ef1}} \equiv \frac{(\mathrm{C_{ef}})_i}{\mathrm{P_1}} = 1 - \exp[-\mathrm{K}] \qquad (3)$$

$$\mathrm{K} \equiv 0.693 \left(\frac{\mathrm{H}}{\mathrm{D_{50}}}\right)^{\beta} \qquad (4)$$

where H total effective acute dose to the target organ

D_{50} dose required for producing an effect in 50 percent of the exposed individuals

β beta or exponential parameter in the hazard function that defines the steepness of the dose response function.

Consequently, if a calculation were available that gave the individual early fatality risk within 1.6 km of the site boundary, for any effective acute dose, Dose, then Equation 2 can be rewritten as

$$\text{Individual risk} \equiv \sum_i (\mathrm{STCF})_i (\mathrm{IEF})_i \qquad (5)$$

For our present purposes, Equation 3 can be rewritten as

$$\sum_i (\mathrm{STCF})_i \left\{1 - \mathrm{e}^{[-0.693(\mathrm{H}/\mathrm{D_{50}})^{\beta}]}\right\}_i \equiv \text{Individual risk} \qquad (6)$$

The items on the left of Equation 6 are those that are determined by a full-scope Level 2 PRA with source term capability and a site characterization parameter. The items on the right contain the result of a Level 3 consequence analysis for individual risk. This parameter can easily be determined using an appropriate computed output. The UCN 3&4 FSAR and PSA report was used to determine site-specific early fatality consequences for inventory, evacuation assumptions, actual site population and wind rose, best-estimate meteorology, and a variety of source terms.

Although the values for $\mathbf{C_{ef}}$ out to the 1.6 km boundary were not specifically reported in the UCN3&4 FSAR and PSA, the information may still exist in the archival printouts of the computed output. In case these data are not currently available, Section 3.2 presents a convenient and robust method for estimating the radiation doses from nuclear plants for any reference source term.

3.2 The Method for Estimating the Radiation Doses

This section presents a convenient and robust method for estimating the total effective acute doses from nuclear plants for any reference source term and provides an appropriate definition of LERF. In reactor accidents, radionuclides, especially fission product gases, are first released into the containment building and subsequently may leak into the atmosphere. Among other things, the magnitude of the resulting plume depends on the rate at which this leakage occurs. The leakage rate from a building is specified as a percentage of the gas in the building that leaks out per day. A reasonably tight reactor building, for example, will leak less than 0.1% per day of its contained gases. Now suppose that in an accident C_0 units of some stable fission product are released into the building, and let C be the amount of the nuclide remaining in the building at time t. Then, C is determined by the equation

$$\frac{\mathrm{dC}}{\mathrm{dt}} = -0.01\mathrm{pC} \qquad (7)$$

Where, p is the leakage rate in percent. Equation (7) is the same as the equation for radioactive decay with the decay constant

$$\lambda_l = 0.01\mathrm{p} \qquad (8)$$

The solution to Eq. (7) can then be written as

$$\mathrm{C} = \mathrm{C_0}\mathrm{e}^{-\lambda_l t} \qquad (9)$$

If, as is the usual case, the fission product is radioactive, then

$$C = C_0 e^{-(\lambda+\lambda_l)t} \tag{10}$$

Where λ is the radioactivity decay constant and C and C_0 are appropriate units of radioactivity.

Because λ_l is equivalent to a decay constant, the rate at which the fission product is released from the building is

$$\acute{Q} = \lambda_l C = \lambda_l C_0 e^{-\lambda_c t} \tag{11}$$

Where

$$\lambda_c = \lambda + \lambda_l \tag{12}$$

is the total decay constant of the fission product in the building.

Equation (11) is the source term in the dispersion relations, derived in dispersion of effluents from nuclear facilities. All nuclear plants emit small amounts of radioactivity, mostly fission product gases, during their normal operation. They may release considerably more radioactivity during the course of an accident. It is necessary to be able to calculate the doses to the public from such releases in order to evaluate the environmental impact of the normally operating plant, to ensure that this is within acceptable standards, and to ascertain the radiological consequences of reactor accidents. Such computations also play an important role in determining the acceptability of a proposed reactor site.

Before dose calculations can be carried out, however, it is necessary to determine how the concentration of the radioactive effluent varies from point to point following its emission into the atmosphere. This question is considered in this section. Let χ be the concentration of some effluent as a function of space and time. If the atmosphere is isotropic and at rest, then χ is determined by the usual diffusion equation

$$K\nabla^2\chi = \frac{d\chi}{dt}, \tag{13}$$

where K is the diffusion coefficient. For the more usual case of a nonisotropic atmosphere, the diffusion equation is

$$K_x \frac{d^2\chi}{dx^2} + K_y \frac{d^2\chi}{dy^2} + K_z \frac{d^2\chi}{dz^2} = \frac{d\chi}{dt}. \tag{14}$$

With a wind blowing at an average speed $\bar{v}$ in the x-direction, the diffusion equation must be altered to account for the fact that the entire medium in which the diffusion is taking place is in motion. The equation then becomes

$$K_x \frac{d^2\chi}{dx^2} + K_y \frac{d^2\chi}{dy^2} + K_z \frac{d^2\chi}{dz^2} = \frac{d\chi}{dt} + \frac{\bar{v}d\chi}{dx}. \tag{15}$$

Consider now a point source located at the origin of coordinates emitting effluent at the constant rate of Q′ units per unit of time. The concentration χ is then not a function of time. Further, it has been found experimentally that most of the movement of an effluent in the direction of the wind is due to the wind itself and not to diffusion. Thus, diffusion in the x-direction can be ignored, which can be accomplished by placing K_x equal to zero. Equation (15) then reduces to

$$K_y \frac{d^2\chi}{dy^2} + K_z \frac{d^2\chi}{dz^2} = \bar{v}\frac{d\chi}{dx} \tag{16}$$

The solution to Eq. (16) that satisfies all the usual boundary conditions can be shown to be

$$\chi = \frac{Q'}{4\pi x\sqrt{K_y K_z}} \exp\left[-\frac{\bar{v}}{4x}\left(\frac{y^2}{K_y} + \frac{z^2}{K_z}\right)\right] \tag{17}$$

According to Eq. (17), the effluent moving along the x-direction spreads out in Gaussian distributions in the y- and z-directions. The standard deviations of these distributions are given by

$$\sigma_y = \left(\frac{2xK_y}{\bar{v}}\right)^{1/2};\ \sigma_z = \left(\frac{2xK_z}{\bar{v}}\right)^{1/2} \tag{18}$$

For purposes of matching Eq. (17) with experimental data, it is convenient to write the equation in terms of σ_y and σ_z, which, it will be noted, are functions of x. Thus, Eq. (17) becomes

$$\chi = \frac{Q'}{2\pi\bar{v}\sigma_y\sigma_z} \exp\left[-\left(\frac{y^2}{2\sigma_y^2} + \frac{z^2}{2\sigma_z^2}\right)\right] \tag{19}$$

In the present context, σ_y and σ_z are called, respectively, the horizontal and vertical dispersion coefficients. Up to this point, it has been assumed that the effluents are emitted at the origin of coordinates into an infinite atmosphere. In fact, they are generally emitted at some altitude, h, into an atmosphere that exists only above the ground. The solution to the diffusion equation in this case can easily be found using the method of images, familiar from electrostatics. With z at the vertical coordinate, this solution is

$$\chi = \frac{Q'}{2\pi\bar{v}\sigma_y\sigma_z}\left\{\exp\left[-\left(\frac{y^2}{2\sigma_y^2} + \frac{(z+h)^2}{2\sigma_z^2}\right)\right] + \exp\left[-\left(\frac{y^2}{2\sigma_y^2} + \frac{(z-h)^2}{2\sigma_z^2}\right)\right]\right\} \tag{20}$$

From this result, the concentration at ground level, z=0, is

$$\chi = \frac{Q'}{\pi\bar{v}\sigma_y\sigma_z} \exp\left[-\left(\frac{y^2}{2\sigma_y^2} + \frac{h^2}{2\sigma_z^2}\right)\right] \tag{21}$$

For example, with ground level release, the value of χ is largest along the centerline of the plume (i.e., where y=0). The concentration there is

$$\chi = \frac{Q'}{\pi\bar{v}\sigma_y\sigma_z} \exp\left(-\frac{h^2}{2\sigma_z^2}\right) \tag{22}$$

Furthermore, because the exponential factor in Eq. (22) is never greater than unity, it follows that the effluent concentration at all points is always greater along the plume with a ground-level release (h=0) than when the effluents are released at some altitude. In this case, Eq. (22) reduces to

$$\chi = \frac{Q'}{\pi\bar{v}\sigma_y\sigma_z}. \tag{23}$$

Eq. (22) gives

$$\chi = \frac{\lambda_l C_0 e^{-\lambda_c t}}{\pi \bar{v} \sigma_y \sigma_z}, \tag{24}$$

where deposition from the plume and building wake effects have been ignored.

Once the atmospheric and ground level concentrations χ and χ_D for each radionuclide are determined, the radiation dose accumulated by individuals can be considered in two phases: the initial or acute phase, during and shortly after passage of the radioactive cloud, and the latent phase, sometimes after the cloud passage.

Radiation dose in the early phase can be received via the following pathways. First, direct external exposure to radiation emitted by radionuclide in the passing cloud (cloud shine) is considered. Second, early external exposure to radiation from radionuclide deposited on the ground (short-term ground shine) is considered. Third, internal exposure due to inhalation of radionuclide from cloud (inhalation) is considered.

The total external dose to a person who stands in the plume given by Eq. (24) for the time t_0 can be computed as in the derivation of Eq. (25).

$$H = \int_0^{t_0} \dot{H}(t)dt = \frac{\dot{H}_0}{\lambda}\left(1 - e^{-\lambda t_0}\right), \tag{25}$$

Where $\dot{H}_0$ is the initial dose rate in the concentration χ_0. When t_0 is long compared with the half-life of the nuclide, Eq. (25) reduces to

$$H = \frac{\dot{H}_0}{\lambda} = 1.44\dot{H}_0 T_{1/2} \tag{26}$$

For a puff of effluent, the total γ-ray and β-ray doses are given by

$$H = 0.262\chi_T \overline{E}_\gamma \text{ rem} \tag{27}$$

And,

$$H = 0.262\chi_T \overline{E}_\beta \text{rem}. \tag{28}$$

Where χ_T is defined in Eq. (29)

$$\chi_T = \int \chi(t)\, dt \tag{29}$$

For γ-rays,

$$H = 0.262\overline{E_\gamma} \int_0^{t_0} \chi\, dt \tag{30}$$

Introducing χ then gives

$$H = \frac{0.262\overline{E_\gamma}\lambda_l C_0}{\pi \bar{v} \sigma_y \sigma_z \lambda_c}\left(1 - e^{-\lambda_c t_0}\right)\text{rem}. \tag{31}$$

When $\lambda_c t_0 \ll 1$, the exponential in Eq. (31) can be expanded to give

$$H = \frac{0.262\overline{E_\gamma} d\lambda_l C_0 t}{\pi \bar{v} \sigma_y \sigma_z} \text{ rem}. \tag{32}$$

If the individual stands in the plume indefinitely, then placing $t_0 = \infty$ in Eq. (31) gives

$$H = \frac{0.262\overline{E_\gamma}\lambda_l C_0}{\pi \bar{v} \sigma_y \sigma_z \lambda_c} \text{ rem}. \tag{33}$$

The dose rate in an internal body organ after standing in the plume for the time t_0 is obtained by substituting Eq. (24) into internal dose from inhalation. Thus,

$$\dot{H} = \frac{592B\xi q \lambda_l C_0}{\pi \bar{v} \sigma_y \sigma_z M} \int_0^{t_0} e^{-\lambda_c \tau} \times e^{-\lambda_e (t_0 - \tau)}$$

$$d\tau = \frac{592B\xi q \lambda_l C_0}{\pi \bar{v} \sigma_y \sigma_z M}\left(e^{-\lambda_e t_0} - e^{-\lambda_c t_0}\right)\frac{\text{rem}}{\text{sec}}. \tag{34}$$

where q is the fraction of the radionuclide that goes to the particular organ, and B is the average breathing rate and M is the mass of the organ in grams and ξ is the effective energy equivalent.

If both $\lambda_e t_0 \ll 1$ and $\lambda_c t_0 \ll 1$, then Eq. (34) reduces to

$$\dot{H} = \frac{592B\xi q \lambda_l C_0 t_0}{\pi \bar{v} \sigma_y \sigma_z M}\frac{\text{rem}}{\text{sec}}. \tag{35}$$

Finally, the dose commitment for inhalation up to is found by introducing Eq. (24) into Eq. (25) and carrying out the integration. The result is

$$H = \frac{592B\xi q \lambda_l C_0}{\pi \bar{v} \sigma_y \sigma_z M \lambda_e \lambda_c}\left(1 - e^{-\lambda_c t_0}\right)\text{rem}. \tag{36}$$

When $\lambda_c t_0 \ll 1$, this becomes

$$H = \frac{592B\xi q \lambda_l C_0 t_0}{\pi \bar{v} \sigma_y \sigma_z M \lambda_e}\text{rem} \quad , \tag{37}$$

And, when either $\lambda_l t_0$ or $\lambda_c t_0 \ll 1$,

$$H = \frac{592B\xi q \lambda_l C_0}{\pi \bar{v} \sigma_y \sigma_z M \lambda_e \lambda_c}\text{rem}. \tag{38}$$

It should be noted that in the preceding discussion, the calculation of the dose received by an organ over a given period of inhalation, , was omitted. This is simply because such a calculation would of little value, as the organ is destined to received its full dose commitment in any case, beyond , barring death or surgery.

Equation 6 is an abbreviated form of the early fatality IR (individual risk) that can be used to derive an acceptance criterion for plants with full-scope, Level 2 PRAs with source term capability and site-specific data. For the QHO criterion to be met, the individual risk must be ≤ 5 x 10^{-7}/yr. Therefore, the relationship between the IEFF Safety Goal QHO and the LERF is defined as:

$$\text{IEFF} = \text{LERF} * \text{IEF} \tag{39}$$

where IEFF Individual Early Fatality Frequency
LERF Large, Early Release Frequency
ILF Individual Early Fatality

And:

$$\text{LERF} = \sum_{k=1}^{N_{STC}} (\text{STCF})_k$$ (:Individual Early Fatality ≥ 0.0625 and evacuation planning)

where N_{STC} number of source term release categories
$(\text{STCF})_k$ the frequency of source term release category k

The criterion, Equation 39, can also be cast in terms of a full-scope Level 2 PRA, with source term and a site characterization parameter.

$$\mathrm{IEF_i} = \left\{1 - e^{\left[-0.693(H/D_{50})^{\beta}\right]}\right\}_{\mathrm{i}} \quad (40)$$

Introducing then gives

$$\sum_i(\mathrm{STCF})_{\mathrm{i}}\left\{1-\exp\left[-0.693\left(\frac{592\mathrm{B}\varepsilon \mathrm{q}\lambda_l \mathrm{C_0 t_0}}{\pi\upsilon\sigma_y\sigma_z\mathrm{MD_{50}}}\right)^{\beta}\right]\right\}_{\mathrm{i}} = \mathrm{IEFF} \quad (41)$$

Therefore,

$$\left\{1-\exp\left[-0.693\left(\frac{592\mathrm{B}\varepsilon \mathrm{q}\lambda_l \mathrm{C_0 t_0}}{\pi\upsilon\sigma_y\sigma_z\mathrm{MD_{50}}}\right)^{\beta}\right]\right\}_{\mathrm{i}} \leq 0.0625 \quad (42)$$

The individual early fatality risk provides a measure of the average probability that a specific individual within 1.6 km of the plant would be exposed to a lethal radiation dose (given that a release of sufficient magnitude to produce lethal doses occurs from the plant) and assuming that the individual does not evacuate.

3.4 Factors Affecting Early Fatality Calculation

Although the unique accidents sequence and source term release rate are considered in the consequence analysis, the resulting offsite health effects may be different if the source term release parameters and weather conditions are different. Therefore, we have made basic spectra based on the relative importance of source term release parameters on offsite health effects. We then investigated the variation of health effects resulting from the severe accidents of the UCN 3&4 nuclear power plants from spectrum to spectrum by using MACCS2 code.

The notion of a "large release" implies the existence of a threshold in release space; that is, given a spectrum of possible releases there exists a boundary, which distinguishes "large" from "not large". Attempts have been made to define a large release magnitude based on off-site health effects. There is an inherent arbitrariness in their definition since off-site health effects depend not only on release magnitude but also on site-specific parameters, such as population. Thus, what would be a large release at one site (or scenario) would not necessarily be one at some other site (or scenario). Weather variability and wind rose are other site-specific factors.

In a hypothetical calculation, the population was fixed by assuming one person located at the site boundary in each of the 16 angular sectors around the site, and the population-weighted risk of early fatalities was calculated for large releases. The individual early fatality risk is the number of early fatalities (which is one in this case since for a particular release the wind only blows into one angular sector) divided by the total population "at risk" which is 16 in this particular case. This study explores the implications of some potential definitions.

The source term profiles, which were derived from level 2 PSA of the UCN 3&4 nuclear power plants, were used to evaluate health consequences. According to the level 2 PSA results, 19 source term categories (STC) are defined by categorizing similar containment failure modes. The MACCS2 code is used to evaluate the health effects resulting from the source terms of the UCN 3&4 nuclear power plants. In MACCS2, the dispersion and deposition of radionuclides released from the reactor containment to the atmosphere were modeled with a Gaussian plume model. The site was selected as the center of a polar grid, and the grid was divided into 16 equally spaced sectors. Evacuations are considered as NUREG-1150 emergency response actions. These actions are to mitigate the effects of a release of radioactivity during a severe accident and are designed to reduce radiation exposures, and public health effects.

For site-specific analyses, it has been our practice to consider the individual early fatality as a scenario and site-specific constant. In this analysis, the individual early fatality is treated as a random variable, which represents the variability (across the spectrum of sites) of the scenario and site-related parameters important to early fatality risk. The individual early fatality can be represented as:

$$\mathrm{IEF} = f(\lambda_l, C_0, t_0, \sigma_y, \sigma_z) \quad (43)$$

where $\sigma_y\sigma_z$ = horizontal and vertical dispersion coefficients
λ_l = the rate at which the fission product is released (release fraction)
C_0 = power level
t_0, = exposure duration (release time)

This section discusses those parameters that are important to early fatality risk. This includes those parameters that are determined by plant design and operations. The second part denotes those parameters that are determined by site characteristics. The plant design and operations have the following parameters that are potentially important to early fatality risk:

1) Source Term Characteristics
   A. magnitude of the fission product release from containment (particularly the volatile I and Te groups)
   B. release thermal energy and release height,
2) Timing Characteristics
   A. timing of release (relative to the start of protective actions) – effective evacuation begins before the start of radionuclide release.
   B. absolute time of release relative to reactor shutdown (for radionuclide decay considerations)

Figure 1 shows three primary factors: release magnitude, release timing, and emergency response, along three axes. With site demographics, meteorology, etc. fixed, various combinations of values of the depicted variables, for example, low magnitude/early timing/conservative (slow) emergency response, or, alternatively, large magnitude/late timing/NUREG-1150 type (prompt) response, could give rise to one early fatality. This schematic depiction is simply meant to show that there is no unique answer (a single point in the parameter space) to the question of what source term leads to one early fatality. There are,

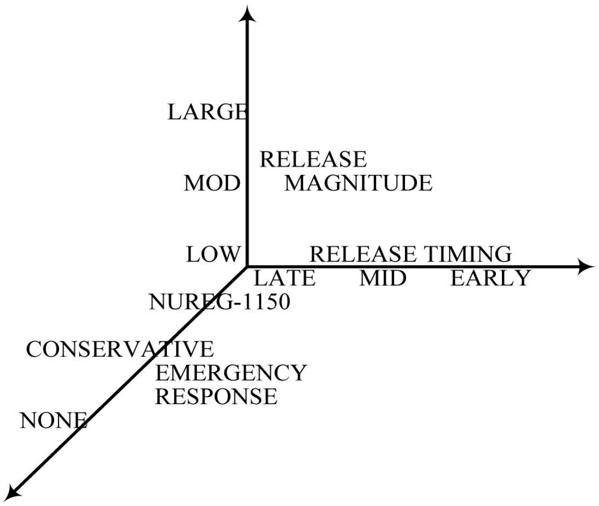

Fig. 1. Release Magnitude, Release Timing, and Emergency Response.

instead, a number of possible answers, or sets of points whose locus would define a hypothetical early fatality surface in the schematic parameter space of Fig. 1.

The source term, as traditionally specified by a set of fractional releases of the initial core inventory, and the time of release to the environment, defines the magnitude of the release to the environment. The timing is significant in that physical removal processes occurring in containment can lower the fractional releases and radioactive decay can significantly lower the amounts of short-lived isotopes which reach the environment, and which are known to contribute significantly to early dose and therefore early fatalities. On the other hand, timely and effective evacuation can in many instances mitigate early consequences, which in the absence of such emergency actions, would be quite serious. Hence, it is to be expected that a spectrum of source terms exists, given their magnitude and timing, and given the type and effectiveness of emergency responses that could occur, which will result in the potential to lead to one mean early fatality, and could be candidates for a large release source term definition.

The MELCOR Accident Consequences Code System (MACCS2) was used to validate off-site early fatalities. The protective actions are modeled in MACCS2. The emergency responses modeled in the EARLY module of the MACCS2 code are of importance. In this study, three release category timing subgroups were defined for each release category. For subgroup 1, it was assumed that evacuation commenced at least 30 minutes prior to the start of radionuclide release. For subgroup 3, the start of evacuation was delayed until one hour or more after radionuclide release had begun. For subgroup 2, the start of evacuation was assumed to occur within a time window from 30 minutes before, to one hour after, the start of release.

4. RESULTS

The final set of calculations reported here was to determine the amount released of a particular isotope, Iodine-131 that would have the potential to cause one early fatality regarding population, meteorology, exclusion boundary, power level, emergency response, time of release, warning time, etc. The specific assumptions for the two hypothetical base cases developed are shown in Table 1. Case 1 assumed a time of release two hours after accident recognition, a release duration of one hour, and a delay time of one hour for the emergency response. In Case 2, the corresponding times were assumed to be two, a half hour, and a half hour, respectively. Both cases assumed an exclusion boundary distance of 1 mile, random sampling weather, reactor power level of 3800 MW_{th}, and an emergency response consisting of three release category timing subgroups were defined for each release category.

The first set of calculations based on Case 1 and Case 2 considered only I-131 releases; the inventory of all the other isotopes was set to zero. The results for the number of this methodology, which estimated early fatalities within 1.6 km of the exclusion area boundary, are shown in Tables 1 for Cases 1 and 2, respectively as a function of the I-131 inventory released. The mean values of the early fatalities for both cases are also depicted graphically in Figure 2. For these calculations, the power level (or equivalently, the inventory of I-131) was sealed from 0.01 to 1.0 times the 2800 MWT level. The I-131 inventory released needing to have the potential to lead to one mean early fatality corresponds to between 30 and 40 percent of the basic 2800 MWT reactor inventory of I-131. For Case 2, as shown in Table 1, the corresponding fraction of the I-131 inventory release that potentially leads to one mean early fatality lies between 60 and 70 percent of the I-131 inventory of a 2800 MWT reactor.

The release categories, which are included in the summations, are either those with an early release fraction of iodine-131 greater than 10% of the core inventory, or for which evacuation is delayed. A ten percent iodine-131 release fraction has been selected as the threshold for source term magnitudes, which can result in lethal doses. The selection of 10% was based principally on the previous equation, which indicated that a threshold for early fatalities occurs at a release fraction of approximately 10% of volatile fission product (I, Cs, Te) release fraction. Figure 2 illustrates this behavior. An early release is defined as the release, which occurs at the time of containment failure (assuming core damage has occurred prior to containment failure). For sequences where containment integrity has been lost prior to core damage, early release begins when core damage commences. This definition for early release is identical to that provided in NUREG-1150. Typical periods of release duration for early release in the NUREG-1150 study are from several minutes to several hours.

The sets of calculations displayed in Table 1, and Figure 2, were designed to evaluate the site boundary dose as a function of the I-131 released inventories for the cases 1 and 2. Assuming one person is located at the site boundary in each of the 16 angular sectors, the population-weighted risk of early fatality was calculated (this is the number of early fatalities divided by the total population "at risk" which is equal to 16 in this particular case). For most weather conditions, the plume at the site boundary spreads over only a fraction of a sector, and since there is only one person in each sector, the code calculates a fractional fatality in most cases. (If one early fatality occurred in each case, the population-weighted risk would be identical to 1/16 or 0.0625). That is why Table 1 shows the mean values of the population weighted risk below 0.06 in all cases.

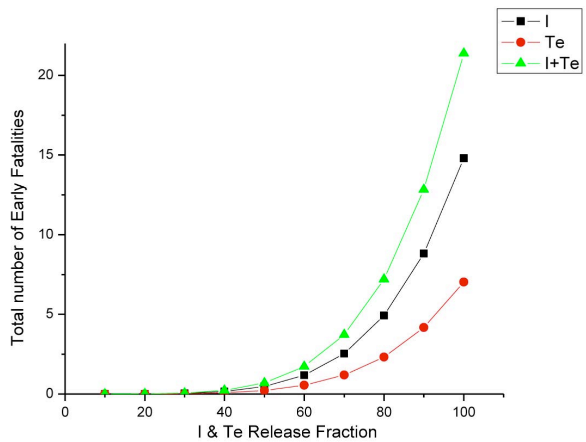

Fig. 2. Total Number of Early Fatalities as a Function of I-131 & Te-132 Release.

Table 1. Early Fatalities for I-131 Release: Case 1 vs. Case2

I – 131 release fraction	Early Fatalities	
	Case 1	Case 2
0.01.	2.08E-08	6.49E-10
0.05	6.49E-05	2.03E-06
0.1	2.08E-03	6.49E-05
0.2	6.65E-02	2.08E-03
0.3	5.05E-01	1.58E-02
0.4	2.12E+00	6.65E-02
0.5	6.42E+00	2.03E-01
0.6	1.57E+01	5.05E-01
0.7	3.27E+01	1.09E+00
0.8	6.00E+01	2.12E+00
0.9	9.81E+01	3.81E+00
1	1.44E+02	6.42E+00

Regardless of the magnitude of the source term, if evacuation commences sufficiently prior to the time when the release of radionuclides begins, then the probability of early fatalities is dramatically reduced. In the NUREG-1150 study, three release category timing subgroups were defined for each release category. For subgroup 1, it was assumed that evacuation commenced at least 30 minutes prior to the start of radionuclide release. For subgroup 3, the start of evacuation was delayed until one hour or more after radionuclide release had begun. For subgroup 2, the start of evacuation was assumed to occur within a time window from 30 minutes before, to one hour after, the start of release.

Figure 3 illustrates the impact of these various evacuation assumptions on the early fatality risk. This figure plots the early fatality risk against the iodine-131release fraction. Individual data points for the three release category

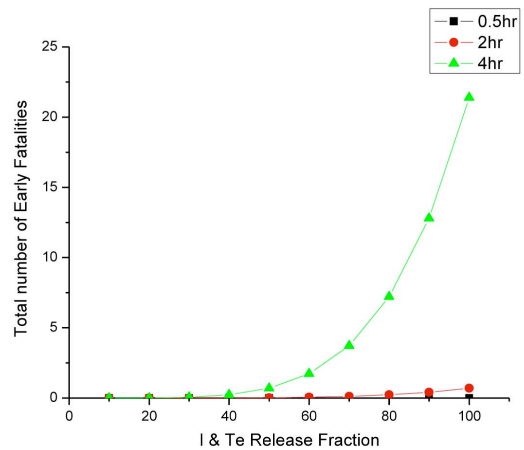

Fig. 3. Early Fatalities as a Function of I-131 Release and Evacuation Assumptions.

Table 2. Population Weighted Risk of Early Fatality for I-131 Release: Case 1 vs. Case 2

I – 131 release fraction	Population weighted risk of early	
	Case 1	Case 2
0.2	1.52E-05	4.74E-07
0.3	1.15E-04	3.60E-06
0.4	4.84E-04	1.52E-05
0.5	1.46E-03	4.63E-05
0.6	3.58E-03	1.15E-04
0.7	7.46E-03	2.49E-04
0.8	1.37E-02	4.84E-04
0.9	2.24E-02	8.69E-04
1	3.28E-02	1.46E-03

subgroups are shown with different symbols. This figure illustrates the effectiveness of early evacuation in reducing the Individual Early Fatality Risk. The diamond shaped symbols represent sequences for which evacuation was delayed until one hour or later, after the start of radionuclide release (subgroup 3). These results are dominated by the fraction of the affected population who are assumed not to evacuate. For sequences characterized by evacuation commencing at about the same time as the start of radionuclide release (subgroup 2), the results (shown as circles) generally fall between the results for subgroups 1 and 3.

5. CONCLUSIONS

This study defines a simple relationship between the Safety Goal QHO for Individual Early Fatality and a Large Early Release Frequency that can be used to estimate the Safety Goal QHO for a specific plant. It also provides a quantitative definition of the LERF. The relationship utilizes simple site-specific characteristics and results from a Level 2 plant-specific Probabilistic Safety Assessment (PSA), which includes the release category frequencies and source term characteristics.

It also results in producing the correlation factor between early fatality and LERF. The early fatality is calculated to validate the results according to the actual residential distances, using the MACCS2 code. The methodology presented in this study includes an approach for providing reasonably robust estimates for the individual early fatality frequency for PSA analyses lacking a detailed Level 3 offsite consequence analysis. This methodology has been applied to a broad spectrum of PSA accident sequences and in all cases a comparison with the PSA results have been performed. The results of this study may contribute to defining LERF and finding a measure for Risk-Informed Regulations and Risk-Informed Design for the next generation nuclear power plants, such as VHTR (Very High Temperature Reactor).

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