UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 11, 1997

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Jackson:

SUBJECT: RISK-BASED REGULATORY ACCEPTANCE CRITERIA FOR PLANT-SPECIFIC APPLICATION OF SAFETY GOALS

In our December 6, 1996 meeting with the Commission, we committed to provide an example of how risk-acceptance criteria could be developed directly from the Safety Goals. Additionally, in a Staff Requirements Memorandum dated January 14, 1997, the Commission asked for our views on the relationship between the concept of "adequate protection," as used in the NRC regulations, and the NRC Safety Goals, from the standpoint of level of risk.

During the 440th meeting of the ACRS, April 3-4, 1997, we completed our deliberations on plant-specific application of NRC Safety Goals and the relationship between the concept of "adequate protection" and the Safety Goals. In our November 18, 1996 report on this subject, we stated that "the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications." We noted that full-scope Level 3 probabilistic risk assessments (PRAs) would be necessary to use the quantitative health objectives (QHOs) directly to assess the acceptability of plant-specific risk. We also stated that this assessment of risk could be done in terms of the QHOs, along with the core damage frequency (CDF), or in terms of the CDF and large, early release frequency (LERF).

This report further discusses the need for plant-specific application of risk-acceptance criteria and the appropriateness of these criteria being derived from the Safety Goal QHO on early fatalities. The additional comments to this report provide examples of approaches that could be used to quantify lower tier acceptance criteria (i.e., LERF, or CDF and conditional containment failure probability) that will ensure that the early fatality QHO is met at each site. Quantification of the LERF at each site is needed to ensure the appropriateness of the choice of the LERF acceptance criterion proposed in draft Regulatory Guide DG-1061 and draft Standard Review Plan sections that support risk-informed, performance-based regulation.

Need for Plant-Specific Application

The Safety Goal Policy Statement makes it clear that the QHOs and the subsidiary goal on CDF were intended only to provide standards for the NRC to judge the overall effectiveness of its regulatory system. The Policy Statement specifically precludes enforcement of the Safety Goals on a plant-specific basis.

In the development of draft Regulatory Guide DG-1061 and the associated draft Standard Review Plan sections in support of risk-informed, performance-based regulation, the staff has found it necessary to propose risk-acceptance guidelines that can be applied on a plant-specific basis. These guidelines would be used, along with other considerations and inputs, for making judgments on the acceptability of requested changes to a licensee's current licensing basis. Reviewing plant-specific license amendments by using risk-acceptance guidelines is a positive action toward risk-informed, performance-based regulation.

We also note that, in the longer term, the Commission may want to consider having a quantified acceptable risk level to replace the current concept of "adequate protection." This risk level could eventually serve as an objective risk-acceptance criterion for many enforcement decisions.

Risk-Informed, Performance-Based Regulation

The Commission has directed the staff to increase the use of PRA in the regulatory process. We have endorsed this because we believe that a risk-informed, performance-based regulatory approach will lead to increased coherence in the regulatory system, to enhanced decision-making ability, and to technically defensible bases for granting regulatory relief.

A risk-informed, performance-based regulatory system ought not be implemented without the existence of top-level risk-acceptance criteria. The obvious choices for these criteria are the NRC Safety Goal QHOs. As it is the responsibility of the NRC to license individual plants and ensure adequate protection, there seems to be no alternative to plant-specific applications.

Relationship Between Adequate Protection and the Safety Goals

Currently, licensing acceptance criteria are embodied in the concept of "adequate protection." With this concept, a plant that is licensed and complies fully with the applicable rules and regulations, is considered to meet the "adequate protection" standard. "Adequate protection" embodies protection of public health and safety against threats that can be quantified in terms of risk as well as threats, such as sabotage and diversion of special nuclear material, for which the risk cannot now be quantified. In the discussion that follows, the nonquantifiable aspects of adequate protection are set aside. Since there are many ways in which plants can be designed and operated within the confines of the regulations, the natural result is a spectrum of risk levels across the population of operating plants. This conclusion is consistent with the results of the recent Individual Plant Examination Program. Since each licensed plant

must, by definition, provide adequate protection, the licensed plant that poses the highest level of risk places a bound on the quantified level of risk to be associated with "adequate protection."

Within the spectrum of risk, it is likely that there are plants with risk levels above the Safety Goals and other plants with risk levels below. If this is indeed the case, a single risk level that bounds "adequate protection" would be a risk level greater than the Safety Goal level. For those plants with risk levels below the Safety Goals, the difference between the plant risk and the Safety Goals can be viewed as margin. It is from some portion of this margin that plant-specific regulatory relief could be granted. For those plants with risk levels greater than the Safety Goals, the challenge will be to eventually reduce their risk to below the Safety Goal level within the confines of the backfit rule.

Regulatory Transparency

The unquantified "adequate protection" concept is not well understood by the general public because the public is unfamiliar with the regulatory process, the body of nuclear regulations, and associated underlying technical bases. We believe that a long-term objective of replacing the "adequate protection" concept with a well articulated and quantified "acceptable level of risk" if achievable, would enhance the public's understanding and acceptance of the regulatory process and would lead to a more uniform level of protection for all individuals living in the vicinity of nuclear plants.

We note that the use of risk-acceptance criteria such as the QHOs will add stability to the regulatory process. This is because the Safety Goals are determined primarily from considerations of societal risk, while the NRC rules and regulations, which are now used to specify adequate protection, change with time as our understanding of reactor safety issues evolves.

Safety Goals as Risk-Acceptance Criteria

It is our opinion that the QHOs are the appropriate choices for risk-acceptance criteria for plant-specific applications. The Safety Goals are the expression by NRC for "how safe is safe enough." In our opinion, this is what risk-acceptance criteria ought to be. As we stated in our August 15, 1996 report, the subsidiary CDF goal should be elevated to the status of a fundamental goal. Elevating the CDF subsidiary goal to the status of a fundamental goal can be considered as a defense-in-depth principle that provides balance between prevention and mitigation.

The early fatality QHO generally controls the risks from nuclear plant operations. Our understanding of risk associated with low-power and shutdown operations, or accidents initiated by external events in which emergency response

is impeded, is not yet sufficient to draw definitive conclusions concerning the limiting QHO in these situations.

Additional comments by ACRS Member T. S. Kress are presented below.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

Additional Comments by ACRS Member T. S. Kress

While I agree completely with the Committee's report, I think it could be augmented in two respects. First, it could make it clearer that, with respect to plant-specific application of the Safety Goals, we are making two related, somewhat radical proposals — the second more so than the first:

1) That lower tier risk-acceptance criteria (CDF and LERF), now being proposed in Draft Regulatory Guide DG-1061 for use in making decisions regarding requested changes to a licensee's current licensing basis, be derived directly from the prompt fatality QHO and be of such value as to bound all current sites.

2) That, in the long run for enforcement purposes, the prompt fatality QHO be considered as the quantification of a risk level to replace "adequate protection."

Second, guidance on how lower tier criteria are to be derived from the QHO is needed. Consequently, I am including two attachments to these additional comments (one developed by me and a complementary one developed by ACRS Senior Fellow Rick Sherry). These provide examples of how to more rigorously derive the lower tier criteria. It is suggested that the staff consider these for use if the first proposal above is to be implemented.

Attachments:
1. Kress, T. S., "Risk-Based Regulatory Acceptance Criteria for Plant-Specific Application of Safety Goals," March 1997
2. Sherry, R. R., "Methodology for Estimating Offsite Early Fatality Risk in the Absence of a Level 3 PRA," March 1997

References:
1. Staff Requirements Memorandum dated January 14, 1997, from John C. Hoyle, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Meeting with ACRS, 9:30 A.M., Friday, December 6, 1996, Commissioners' Conference Room.

2. Report dated November 18, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Plant-Specific Application of Safety Goals.
3. Report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters.
4. U.S. Nuclear Regulatory Commission, NUREG-1560, Volume 1, Part 1, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Summary Report, Draft Report for Comment, October 1996.
5. U.S. Nuclear Regulatory Commission Draft Regulatory Guide, Draft DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," dated February 28, 1997 (Predecisional).
6. U.S. Nuclear Regulatory Commission, Draft Standard Review Plan Chapter 19, Revision L, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," dated March 3, 1997 (Predecisional).

ATTACHMENT 1

Risk-Based Regulatory Acceptance Criteria for Site Specific Application of Safety Goals

T.S. Kress

The purpose of this discussion paper 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 working definition for the risk of early fatalities for any specific plant in terms of the normal determinations of probabilistic risk assessments (PRAs).

$$\text{Mean number of early fatalities} = \sum_i (CDF)_{i,k}(CCFP)_{i,k}(C_{ef})_{i,k} \tag{1}$$

where
- k refers to a specific plant,
- i refers to the spectrum of accident sequences,
- $(CDF)_{i,k}$ is the core damage frequency for sequence i of plant k,
- $(CCFP)_{i,k}$ is the conditional containment failure probability for sequence i and plant k,
- $(C_{ef})_{i,k}$ is the early fatality consequences at site k given the sequence I which has associated with it a source term St_i, that may be defined in terms of the equivalent release of iodine *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 one mile* of the site boundary, C_{ef1}, and dividing that by the population within that same one mile region, P_{k1}. Therefore, Equation 1, for purposes of comparing with the early fatality QHO, should be rewritten as

$$\text{(IR) Individual risk} = \sum_i (CDF)_{i,k}(CCFP)_{i,k}(C_{ef1})_{i,k}/P_{k1} \tag{2}$$

In order to proceed further, we first note that, in general, the early fatality consequences within one mile of the site boundary can be related to the source term expressed in terms of the equivalent release of iodine by the relationship

$$C_{ef1} \sim (St)^n$$

An appropriate exponent is n = 0.9, which can be obtained from Figure 2 of Attachment 2. Consequently, if a calculation were available that gave the expected early fatalities within one mile of the site boundary, C_{k1r}, for any reference source term, St_r, then Equation 2 would be rewritten as

$$IR = \sum_i (CDF)_{i,k}(CCFP)_{i,k} \quad [\frac{C_{klr}}{P_{kl}} (\frac{St_{ik}}{St_r})^n] \tag{3}$$

For our present purposes, Equation 3 can be rewritten as

$$\sum_i (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n = \frac{(IR)\ P_{kl}\ (St_r)^n}{C_{klr}} \tag{4}$$

The items on the left of Equation 4 are those that are determined by a full-scope Level 2 PRA with source term (to the environment) capability. The items on the right contain the result of a Level 3 consequence analysis for IR and a site characterization parameter, $(P_{k1})(St_r)^n/C_{k1r}$. This parameter can easily be determined using an appropriate "consequence" code such as MACCS or CRAC and, in fact, may already exist as part of NUREG/CR-2239, "Technical Guidance for Siting Criteria Development."

NUREG/CR-2239 basically has a characterization of each US site with respect to C_{el}. The CRAC code was used to determine site-specific early fatality consequences for

- a standard 1120 Mwe plant (for inventory),
- summary evacuation assumptions,
- actual site population and wind rose,
- best-estimate meteorology, and
- a variety of source terms.

Although the values for P_{k1} and C_{k1r} out to the one mile boundary were not specifically reported in NUREG/CR-2239, the information may still exist in the archival print-outs of the computed output. In case these data are not currently available, Attachment 2 presents a convenient and robust method for estimating the prompt fatalities at each site for any reference source term and provides an appropriate definition of LERF.

<u>Direct Comparison with the QHO Using Level 2 PRAs with Source Term Capability</u>

Equation 4 is an abbreviated form of the early fatality IR that can be used to derive an acceptance criterion for plants with full-scope, Level 2 PRAs with source term capability. For the QHO criterion to be met, the IR must be $\leq 5 \times 10^{-7}$/yr. Therefore the acceptance criterion parameter is clearly

$$\sum_i (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n \leq \frac{(5x10^{-7})\ P_{kl}\ (St_r)^n}{C_{klr}} \tag{5}$$

The criterion, Equation 5, can also be cast in terms of a conditional early release probability criterion, CERP, that can be defined as

$$[\sum (CDF)_{i,k}]\ x\ CERP = \sum (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n$$

or

$$CERP_k \equiv \frac{\sum (CDF)_{i,k}(CCFP)_{i,k}(St_{i,k})^n}{\sum_i (CDF)_{i,k}}$$

In terms of acceptance criteria similar to what was recently proposed by the staff with coordinates of CDF and LERF with a basic tenet that CDF should be $\leq 1 \times 10^{-4}$/yr, the appropriate plant specific CERP acceptance criterion would be

$$CERP_{ka} \leq \frac{(5x10^{-7})(St_r)^n(P_{k1})}{(10^{-4})C_{k1r}}$$

If a single $CERP_a$ is desired that would bound all current sites, the site with minimum P_{k1}/C_{k1r} would have to be chosen.

<u>Level 2 PRA Without Source Term Capability</u>

For the case of a plant that has Level 2 PRA capability to determine CCFP but not to establish the various associated sequence source terms, it is desired to develop a lower tier criterion in terms of a defined CCFP that would be bounding with respect to ensuring that the early fatality QHO is met on a plant-specific basis.

To do this from Equation 5, it will be necessary to use a single source term that is sure to bound each sequence source term. A possible value that would be bounding (for PWRs only) is 0.5 (release fraction of iodine equivalent). With this value, Equation 5 as a PWR plant-specific bounding criterion becomes

$$\sum_i (CDF)_{i,k}(CCFP)_{i,k} \leq \frac{(5x10^{-7})(P_{k1})(St_r)^n}{(C_{k1r}) \quad (.5)^n}$$

The parameter on the left is what I believe the staff used in its decision chart and called a "large early release frequency" (LERF). This would more aptly be called a containment failure frequency.

To express this in terms of a conditional containment failure probability, we could use the usual definition of CCFP.

$$CCFP_k \equiv \frac{\sum (CDF)_{i,k}(CCFP)_{i,k}}{\sum (CDF)_{i,k}}$$

Then, an acceptance value (for PWRs only) for use in a "decision" chart along with CDF = 10^{-4}/yr becomes

$$CCFP_{ka} \leq \frac{(5x10^{-7})}{(10^{-4})} \frac{(P_{k1})}{(C_{k1r})} \frac{(St_r)^n}{(.5)^n}$$

A separate acceptance criterion could be developed for BWRs using a bounding St that would have to recognize the effects of the suppression pool and its bypass. An optional way to account for the differences between BWRs and PWRs could be to make different choices for the sequences to go into calculating the CCFP. This is what the staff chose to do - thereby preserving the same acceptance criterion for both reactor types.

For site bounding values, the minimum P_{k1}/C_{k1r} for each reactor type would be chosen. It would be this bounding value that should, in the case of PWRs, be compared with the CCFP = 0.1 that is often mentioned as an appropriate criterion to use along with CDF = 1 x 10^{-4} /yr.

ATTACHMENT 2

Methodology for Estimating Offsite Early Fatality Risk in the Absence of a Level 3 PRA

Rick Sherry- Senior Fellow
Advisory Committee on Reactor Safeguards

Introduction

This paper defines a simple relationship between the Safety Goal Quantitative Health Objective (QHO) for Individual Early Fatality Frequency (IEFF) and a Large, Early Release Frequency (LERF) that can be used to estimate the Safety Goal QHO for a specific plant at a specific site. This paper also provides a quantitative definition of the LERF. The relationship utilizes simple site-specific characteristics (wind direction frequencies and population demographics) and results from a Level 2 plant-specific probabilistic risk assessment (PRA) (release category frequencies and source term characteristics).

Estimates using this methodology have been compared with results from detailed calculations performed using the MELCOR Accident Consequence Code System (MACCS) (Ref. 1) for the NUREG-1150 study (Ref. 2), the Grand Gulf Shutdown study (Ref. 3) and the Maine Yankee PRA/IPE. This comparison includes internally initiated sequences, seismic sequences, and shutdown sequences.

Summary of Methodology

The relationship between the IEFF Safety Goal QHO and the LERF is defined as:

$$IEFF = LERF \times EI \qquad \textit{Equation 1}$$

$$\text{where: } EI = \text{Exposure Index} = \frac{\sum_{i=1}^{16} P_i \times F_i}{\sum_{i=1}^{16} P_i} \qquad \textit{Equation 2}$$

F_i = *the relative frequency wind blows toward sector i*

P_i = *population in sector i within one mile of the plant*

$$\text{and: } LERF = \sum_{k=1}^{N_{rc}} RC_k(\text{: early } I > 10\% \text{ and Evacuation Delayed}) \times (1 - F_{no\text{-}evac})$$

$$+ \sum_{k=1}^{N_{rc}} RC_k(\text{: total } I > 10\%) \times F_{no\text{-}evac} \qquad \text{Equation 3}$$

RC_k = the frequency of release category k

N_{rc} = number of release categories

$F_{no\text{-}evac}$ = the population fraction not evacuating

Discussion

The Exposure Index (EI) provides a measure of the average probability that a specific individual within one mile of the plant would be exposed to a lethal radiation dose (given that a release occurs from the plant of sufficient magnitude to produce lethal doses) and assuming that the individual does not evacuate. The exposure index defined above is a slight variation from that initially developed for the Generic Environmental Impact Statement for License Renewal of Nuclear Plants(Ref. 4). Table 1 shows the calculated values for the exposure index for a number of nuclear plant sites using site specific wind direction frequency and population data. Note that all calculated values of the EI fall well within a factor of two of the nominal value of 1/16 which would result if the population within one mile of the plant and the wind direction frequency were uniformly distributed in the sixteen wind direction sectors.

In the above definitions, the release categories which are included in the summations are either 1) those with an early release fraction of iodine greater than 10% of the core inventory and for which evacuation is delayed or 2) release categories where the total iodine release fraction (sum of early and later releases) exceeds 10% and evacuation does not occur. The second summation accounts for the fraction of individuals that do not evacuate and are assumed to be exposed to all time phases of the radionuclide release. This second summation also governs severe seismic sequences where evacuation is assumed to be ineffective.

A ten percent iodine 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 work of Kaiser (Ref. 5) 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 1 from Kaiser 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.

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 2 illustrates the impact of these various evacuation assumptions on the Conditional Individual Early Fatality Probability[1] (CIEFP). This figure plots the CIEFP against the iodine release fraction. Individual data points for the three release category 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). For these sequences, the peak CIEFP is on the order of $3 \times (10)^{-2}$. The triangle shaped symbols represent sequences for which evacuation was initiated at least 30 minutes prior to the start of the release of radionuclides (subgroup 1). These sequences have a CIEFP of order $2 \times (10)^{-4}$ or less. 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.

Equation 1 can be rearranged and used to estimate a plant specific LERF subsidiary objective:

$$LERF = \frac{IEFF}{EI} \qquad \textit{Equation 4}$$

This equation along with the plant specific Exposure Index values shown in Table 1 have been used to calculate a plant specific LERF objective. These results are also shown in Table 1. Note

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1 The CIEFP is conditional on the occurrence of sequences (release categories) with the indicated volatile fission product release magnitude.

that for these calculations, the assumed value for the Safety Goal QHO IEFF is taken as $3 \times (10)^{-7}$ per year.[2]

Example Application Results

The simple methodology has been applied to six cases. These cases are:

Surry Internal Event Initiated Sequences (NUREG-1150)
Surry Seismic Sequences (NUREG-1150)
Peach Bottom Internal Event Initiated Sequences (NUREG-1150)
Peach Bottom Seismic Sequences (NUREG-1150)
Grand Gulf Shutdown Sequences (NUREG/CR-6143)(Ref. 3)
Maine Yankee Internally Initiated Sequences (PRA/IPE submittal)

The results of the example application are summarized in Figure 3. In this figure, the Individual Early Fatality Risk Frequency calculated using the simple methodology is compared with the mean and 95th percentile values from the PRA offsite consequence (MACCS) code calculations. In all cases, the simple methodology produces estimates for the IEFF that are above the PRA mean values. For the two seismic cases, the results fall between the PRA mean and 95th percentile values, and in the remaining cases the results lie above the 95th percentile(but within a factor of three of the 95th percentile value).

The limited comparison summarized in Figure 3 indicates that this methodology may provide a simple and easy to use approach for providing reasonably robust estimates for the IEFF for PRA analyses lacking a detailed Level 3 offsite consequence. This methodology has been applied to a broad spectrum of PRA sequence classes and in all cases the comparison with the PRA results have been favorable.

Discussion of Results

<u>Internally Initiated Sequences (NUREG-1150)</u>

For Peach Bottom internally initiated sequences, all release categories that met the criteria for early iodine release greater than 10% also had evacuation beginning prior to the start of release. This is the principal reason that the MACCS calculated IEFF is low for Peach Bottom. It is only the small fraction of the population that does not evacuate which contributes to the calculated

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2 Based on data from the National Center for Health Statistics the total U.S. societal accident fatality rate was 34.4 and 34.6 deaths per 100,000 individuals in 1993 and 1994, respectively. This translates into an individual early fatality Safety Goal QHO of $3 \times (10)^{-7}$ per reactor year.

IEFF. Consequently, in this study the estimated IEFF for Peach Bottom is controlled by the second summation term in Equation 3 (as for the NUREG-1150 MACCS analyses it was assumed that 0.5% of the population did not evacuate and $F_{no-evac}$ was set to this value).

For the Surry plant, the dominant contributors to a large early fission product release and early fatalities for internally initiated sequences were containment bypass events (12% of CDF). These sequences contribute about 80% of the early fatality risk. For the fast developing interfacing system LOCA bypass sequences, warning times were typically within one hour of the start of radionuclide release. Consequently, evacuation would generally begin after the start of release.

Seismic Sequences (NUREG-1150)

The NUREG-1150 results for seismic sequences for the Peach Bottom and Surry plants using the LLNL seismic hazard curves were compared for this study. The NUREG-1150 seismic sequence consequence analyses were separated into two categories - low Peak Ground Acceleration (PGA)(< 0.6 g) and high PGA (> 0.6 g). For seismic sequences, the offsite emergency protective action assumptions input into the MACCS analyses differed from that for internally initiated events. Particularly important was the assumption that for the high PGA category sequences evacuation was assumed to be ineffective (i.e., does not occur). Consequently, for this study the total iodine release fraction for high PGA seismic sequence classes was used (i.e. the second summation in Equation 3 with $F_{no-evac}$ set to one).

Grand Gulf Shutdown Sequences (NUREG/CR-6143)

For shutdown sequences, the activity level of important volatile fission product species will be reduced by decay from their full power levels and this reduction should be accounted for in analyses. However, for this application this was not done. Even with this limitation the estimated IEFF using the simple methodology was in good agreement with the MACCS results. As for the Peach Bottom internally initiated sequences, warning times for Grand Gulf shutdown sequences were sufficiently early that evacuation commenced prior to the beginning of fission product release, and it is only that small fraction of the population that does not evacuate which contributes to the calculated IEFF. Consequently, the estimated IEFF is controlled by the second summation term in Equation 3.

Maine Yankee Internally Initiated Sequences (PRA/IPE)

The IEFF for Maine Yankee is dominated by sequences where containment failure occurs as a result of a hydrogen combustion event near the time of reactor vessel meltthrough. This type of failure was predicted to occur for about 8% of the core damage frequency. For those sequences contributing to the IEFF, the radionuclide release is predicted to begin about one hour after a general emergency would be expected to be declared. In the Maine Yankee PRA, delay times

ranging from 1 to 2.5 hours were assumed (dependent on the time of year). Hence, for these sequences, radionuclide release generally begins prior to, or coincident with, the start of evacuation.

Enhancements - Use of Iodine Equivalent

Although this paper has shown that reasonably good correlation exists between the IEFF and the LERF calculated using Equation 3 (which uses an iodine release fraction of 10% as threshold for early fatalities), it was also observed that under certain conditions other fission product species groups can make significant contributions to the early fatality risk. It was observed that if the Ru or La group release fractions exceeded about 1 to 5% of core inventory, they began to make a significant contribution to early fatality risk. Furthermore, as discussed above, for sequences initiated during shutdown which may have significant decay periods, the contribution of iodine isotopes to the early fatality risk will be significantly reduced and other radionuclide groups will begin to dominate. For these reasons, a parameter that can be evaluated from the release fraction for all radionuclide groups (and as a function of shutdown time) is desirable. An example of the type of iodine equivalent parameter that is required is presented in NUREG/CR-5164 (Ref. 6) which is based on work reported in Reference 7. In these studies, weighting factors for the contribution to early fatality risk for each radionuclide species group were developed. These weighting factors (which are multiplied by the radionuclide group release fraction) can be used to develop a total iodine equivalent release.

References

1. Chanin, D. I., et al., "MELCOR Accident Consequence Code System (MACCS): User's Guide," NUREG/CR-4691,Vol. 1, Sandia National Laboratories, February 1990.

2. "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990. [Note, references to NUREG-1150 also imply reference to the NUREG-1150 supporting documents including applicable volumes of NUREG/CR-4551]

3. Brown T. D. et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," NUREG/CR-6143, Vol. 6, Part 1, March 1995.

4. "Generic Environmental Impact Statement for License Renewal of Nuclear Plants", NUREG-1437, Vol. 1, May 1996.

5. G.D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," ANS Executive Conference on the Ramifications of the Source Term, Charleston, SC, March 12, 1985.

6. Madni I. K., et al., "A Simplified Model for Calculating Early Offsite Consequences from

Nuclear Reactor Accidents'" NUREG/CR-5164, Brookhaven National Laboratory, July 1988.

7. Alpert J., et al, "Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions," NUREG/CR-4467, Sandia National Laboratories, March 1986.

5. Brown T. D. et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," NUREG/CR-6143, Vol. 6, Part 1, March 1995.

Table 1
Example of Plant Specific Exposure Index and Derived Subsidiary LERF Objective

Plant	EI	Derived Subsidiary LERF Objective
Grand Gulf	.065	5E-6
Surry	.077	4E-6
Sequoyah	.045	7E-6
Maine Yankee	.040	8E-6
Zion	.081	4E-6
Peach Bottom	.075	4E-6
Lasalle	.083	4E-6
Nominal	.063 (1/16)	5E-6

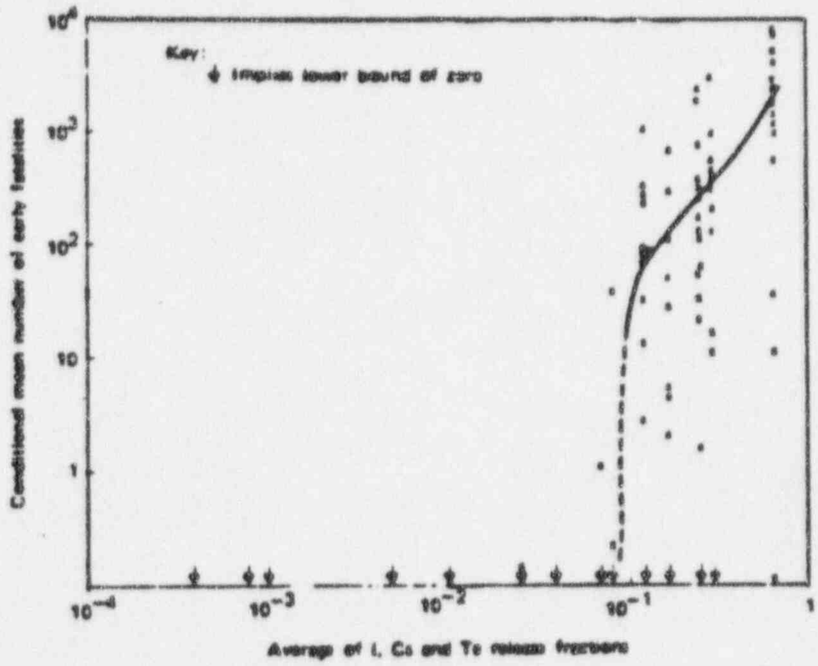


Figure 1. Early Fatalities

Figure 2

Surry Individual Early Fatatality Risk
Internal Initiators

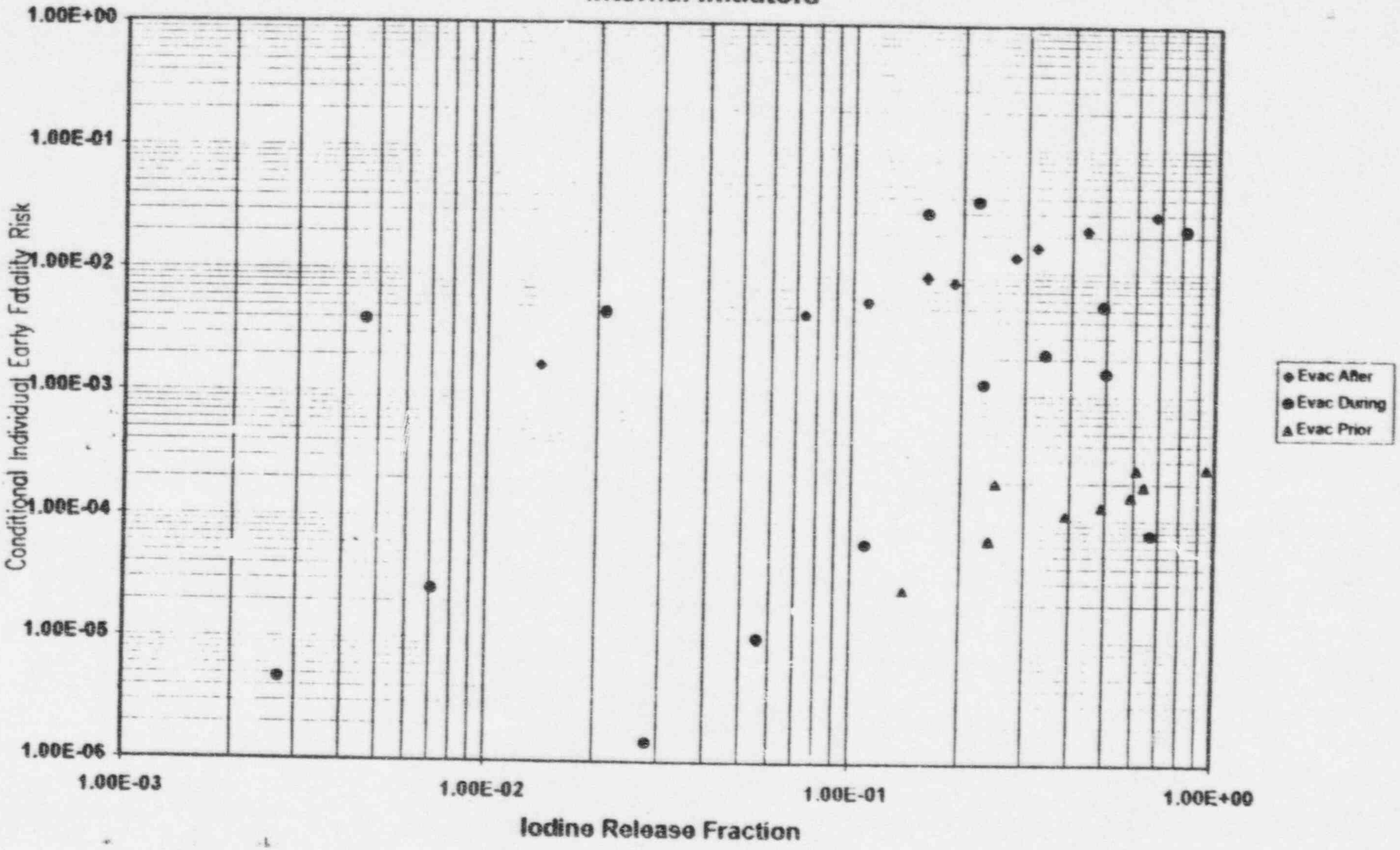
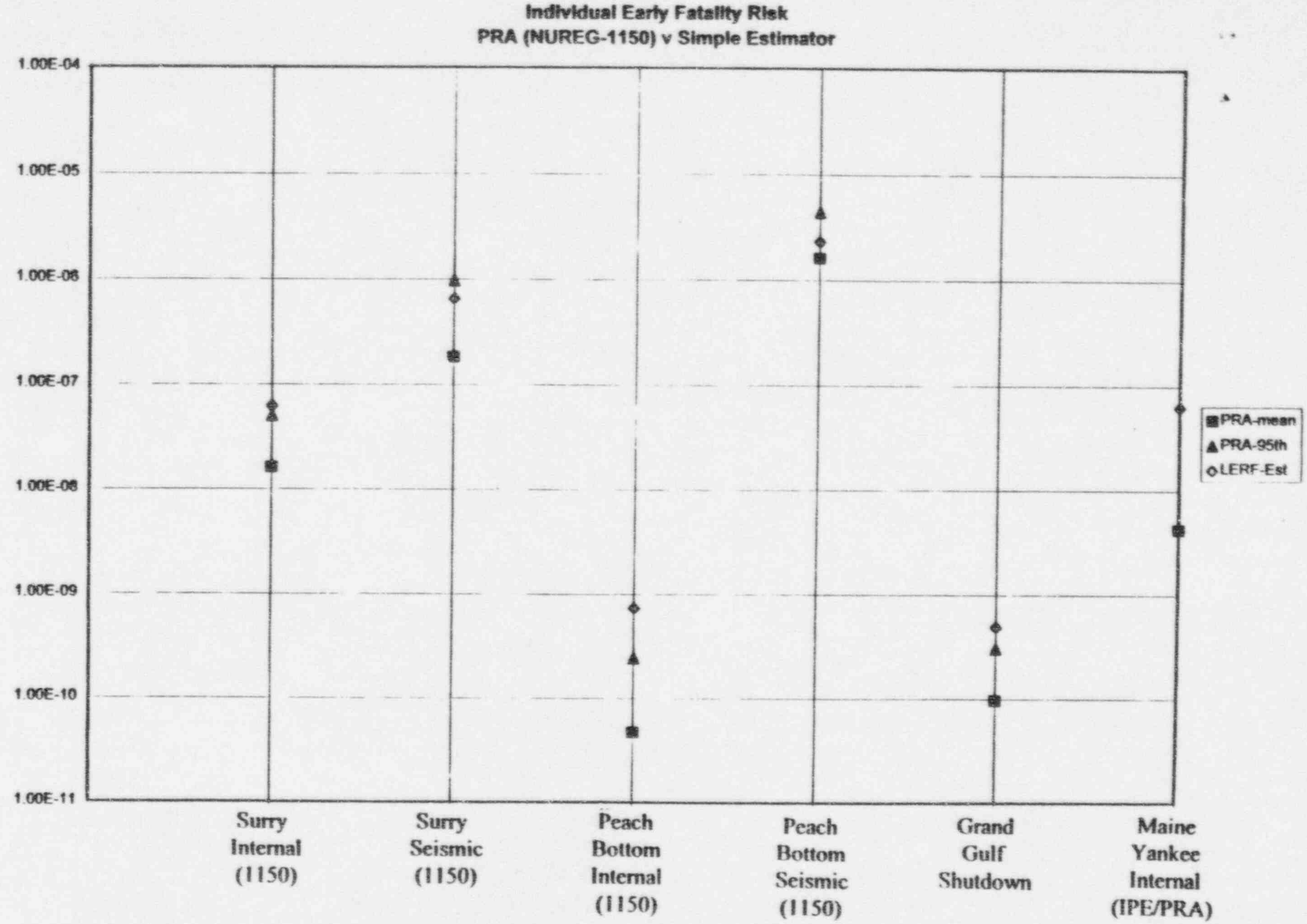


Figure 3



Individual Early Fatality Risk
PRA (NUREG-1150) v Simple Estimator
1.00E-04
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
1.00E-10
1.00E-11
PRA-mean
PRA-95th
LERF-Est
Surry Internal (1150)
Surry Seismic (1150)
Peach Bottom Internal (1150)
Peach Bottom Seismic (1150)
Grand Gulf Shutdown
Maine Yankee Internal (IPE/PRA)