NRC Responses to Public Comments on NUREG-0800, *Standard Review Plan*, Chapter 18, "Human Factors Engineering" and Related Documents

January 14, 2004

1 Introduction

The staff of the US Nuclear Regulatory Commission (NRC) reviews the human factors engineering (HFE) aspects of nuclear power plants to evaluate whether reasonable assurance exists that personnel performance and reliability are appropriately supported. The guidance for the review of the HFE aspects of nuclear plants exists at several levels. Chapter 18, Human Factors Engineering, of the *Standard Review Plan* (NUREG-0800) provides a high-level review framework for the conduct of HFE reviews. The SRP references the *Human Factors Engineering Program Review Model* (NUREG-0711) for detailed review criteria. It also references *Guidance for the Review of Changes to Operator Actions* (NUREG-1764) for review criteria that are specifically tailored to the review of plant modifications and license amendment requests involving credited operator actions. In addition, these documents reference the *Human-System Interface Design Review Guidelines* (NUREG-0700), for review addressing the characteristics and functions of human-system interfaces. These reports have been recently revised and the changes are briefly outlined below. The SRP Chapter 18 was revised to address the following: (1) to incorporate the staff's general approach to HFE reviews in accordance to that developed in NUREG-0711; (2) provide guidance for the review of the HFE aspects of new plants, control room modifications, and modifications affecting risk-important human actions; and (3) provide guidance for a graded approach to HFE review.

NUREG-0711 was revised to: (1) make it more generally applicable to all human factors reviews and not limited to advanced reactors; (2) make it the one source of review procedures (in the past procedures were divided between NUREG-0711 and NUREG-0700); address industry initiatives for digital I&C and control room modernization; and (3) to update the technical content of the individual elements to reflect the current state-of-the-art.

NUREG-0700 was revised to: (1) address important HSI topics, such as soft controls and computer-based procedures that were not included in Rev. 1; and (2) limit its content to HSI review guidelines and not review procedures (which have been integrated into NUREG-0711).

NUREG-1764 is an new document developed to: (1) consolidate in one document review guidance for changes to credited human actions, and (2) include a risk-informed screening method to provide a graded approach to make the level of human factors reviews commensurate with the risk importance of the human action.

These documents were submitted for public comments. In March, 2003, NRC received public comments on these documents from Dominion, the Nuclear Energy Institute (NEI), the Strategic Teaming and Resource Sharing (STARS) Alliance, and Jeffery A. Julius of Scientech.

NRC's response to the public comments are described in this document. The responses are organized by commentor. Each individual comment is presented, followed by a discussion that

addresses the material in question. Then, the staff's changes, if any, are described.

The revision to Chapter 18 of the SRP, NUREG-0711, and NUREG-1764 have been completed and reflect the changes identified below.

2 Comments Provided by Dominion

Comment 1

The draft NUREG-1764 provides a screening approach to the review of operator actions assumed in the safety analysis. For those actions assigned a medium risk level certain review steps are necessary. However, under certain conditions more review is required. One of the reasons for requiring more review is if the human action was developed with a poor human reliability analysis (HRA). One of the reasons for concluding that the HRA is poor is that it was not based on simulator or operating crew survey as required by the ASME PRA standard. For a new human action, simulator data or an operating crew survey is generally not available because the action is new. The use of expert judgement is the only practical way to do the analysis. The use of expert judgement should not be the basis for concluding that additional review of the human action is required.

Discussion

For any given application, only those requirements of the PRA Standard that are needed for the application need be met. The comment seems to imply that the use of a simulator or operating crew survey would be required for this application, and that the use of expert judgement implies an inadequate HRA. The parts of the HRA required for this application (the assessment of the risk significance of an HA) include those that are associated with the appropriate inclusion of a human failure event in the logic model. If simulator exercises were possible they could be used to provide information that would be useful in the estimation of the human error probabilities (HEP), but they would not be sufficient. Some degree of expert judgement would be required. What is important, however, is that the uncertainty on the HEPs be characterized in such a way that meaningful sensitivity analyses can be performed to determine the potential impact of the HAs.

Change Made in response to Comment

The category of "Poor HRA Analysis" was removed from Step 3 (Qualitative Assessment) of the screening process. The quality of the PRA (including the HRA) and dealing with the uncertainty in the HEPs are now dealt with in a more appropriate way as part of the PRA-based determination of the level of risk significance, rather than as a separate qualitative adjustment.

Comment 2

Comment

References should be provided to what are considered current, accepted PRA/HRA principles and practices. On p. 35 it is stated "…(2) performed an analysis of the potential effects of the proposed changes on plant safety and reliability, in a manner consistent with current, accepted

PRA/HRA principles and practices…". Would these be PRA/HRA principles and practices accepted by the industry, the NRC or both? Has the NRC endorsed any HRA methods? The ASME PRA Standard should be referenced.

Discussion

The PRA and HRA quality need to be addressed.

Change Made In Response to the Comment

The discussion on PRA quality has been changed to include a reference to the use of a PRA Standard as endorsed by the NRC staff as a means of demonstrating that the quality of the PRA and its embedded HRA is adequate for the assessment of the significance of the HA.

3 Comments Provided by NEI

In a letter dated March 28, 2003 from James Davis (NEI) to Richard Eckenrode (NRC), industry comments were provided on Draft NUREG-0800, Revision 2 (December 2002). These comments were endorsed by the Strategic Teaming and Resource Sharing (STARS) Alliance in a letter dated March 28, 2003 from D. R. Woodlan (STARS) to Richard Eckenrode (NRC). Five separate comments on Chapter 18, Human Factors Engineering, were included addressing five separate aspects of the guidance.

Comment 1

Comment

Section II.A.1, page l8.0-6. In discussing the HFE design team, Draft NUREG-0800 states that the applicant must have one, "as required by 10CFR50.34 (f)(2)(iii)." This suggests that the law requires such a team, when in fact it does not. Actually, in response to industry comments, NUREG-0711 explicitly states that, "there is, however, no assumption that HFE is the responsibility of a single organization or that there is an organizational unit called the HFE design team." To avoid confusion, this reference in NUREG-0800 to 10 CFR should be deleted from the design team discussion, and the above caveat from NUREG-0711 should be added to NUREG-0800.

Discussion

The sentence was ambiguous in that the reference to 10 CFR was relevant to the commitment to HFE and not the design team itself. This ambiguity should be clarified by the change recommended below.

Change Made In Response to the Comment

The sentence in NUREG-0800 was changed to read as follows.

> The applicant has an HFE design team with the responsibility, authority, placement within the organization, and composition to ensure that the design commitment to HFE is achieved. There is, however, no assumption that HFE is the responsibility of a single organization or that there is

an organizational unit called the HFE design team.

NUREG-0711 does not have this wording so no change is needed there.

Comment 2

Comment

The further requirement that the HFE Program Plan follow “a structured top-down systems analysis,” imposes an impractical and outdated design model (i.e., the Systems Approach to Design) as the “state-of-the-art.” The Systems Approach has been recommended since TMI (NUREG-0700 APP. B). With ALWR licensing in the 1990's, government scientists found that the Systems Approach was not necessarily the safest or most effective, but merely that it was the oldest, most formal and extensive (NUREG-0711, 1994; p.1-7). As to the potential inefficiencies of the Systems Approach, a mountain of evidence (Hoos, 1972; Tribe, 1973; Sutherland, 1978; March & Weissinger-Baylon, 1986; Wildavsky, 1986; Feynman, 1988; De Greene, 1990; Ferguson, 1992; Fainstein & Fainstein, 1996; Hughes, 1998; Fuld, 2000; Sheridan, 2000; Andrews, 2001, among others) is compelling. To avoid unnecessary regulatory burden, NUREG-0800 should delete the requirement for applicants to apply a “structured top-down systems approach”..

Discussion

The notion of a top-down, systems approach in NUREG-0800 and -0711 is at the core of the NRC's regulatory approach to HFE. The NUREG-0800 and -0711 elements reflect such an approach throughout.

NUREG-0711 states that:

> State-of-the-art human factors principles are defined as those principles currently accepted by human factors practitioners. *"Current" is defined with reference to the time when a program management or implementation plan is prepared*. "Accepted" is defined as a practice, method, or guide that is (1) documented in the human factors literature within a standard or guidance document that has undergone a peer-review process or (2) can be justified through scientific research and/or industry practices. (Italics added)

Overall, it does not seem that NEI is requesting fundamental and wholesale rework of either -0800 or -0711. The cover letter transmitting the comments indicates that: "It should be noted that there are few comments, indicating that the draft sections provide adequate information to successfully develop and implement the targeted programs and plans."

The comment itself is unclear with respect the following:

- the statement is made that a top-down systems analysis is "impractical and outdated," yet it does not identify any more practical and current model(s)
- NUREG-0711 is referenced for the statement that "government scientists found that the Systems Approach was not necessarily the safest or most effective...," yet NUREG-0711 makes no such statement. Given the reference, it appears that the "government

scientists" identified are the NRC and BNL authors of NUREG-0711 and they do not share this opinion.

- The "compelling" "mountain of evidence..." that is generally referenced in support of the conclusion that the systems approach is *potentially* inefficient, includes many references that do not address this subject, e.g., DeGreene (1990) discusses changes needed in human factors research and practice as a science. Other articles address public policy and its analysis. Others somewhat more related address only a very limited aspect of the systems approach, e.g., function allocation.

NUREG-0800 and -0711 contains review elements that were developed to support a wide range of review needs. In the application of their criteria for a *specific* review, the staff tailors the full set of review elements and criteria to select those that are most relevant (as per NUREG-0800, Chapter 18, IB, Graded Approach to Review and NUREG-0711, Section 1.4, Graded Approach to Review). This is accomplished to reflect differing review needs and to reduce regulatory burden by focusing staff review only on those elements and criteria that are pertinent to the review.

Change Made In Response to the Comment

The sentence addressing this topic in NUREG-0800 was changed to read as follows.

> This HFE program plan should describe the technical program in sufficient detail to ensure that all aspects of the HSIs, procedures, and training are developed, designed, and evaluated on the basis of a structured analysis using accepted HFE principles.

Three sections of NUREG-0711 was revised to read as follows.

Executive Summary Introduction

> The HFE aspects of the plant should be developed, designed, and evaluated based on the basis of a structured analysis using accepted HFE principles.

Executive Summary Element 1 Description

> This plan should describe the technical program elements ensuring that all aspects of the HSI, procedures, and training are developed, designed, and evaluated on the basis of a structured analysis using accepted HFE principles.

Element 1 Description

> The objective of this review element is to ensure that the applicant has an HFE design team with the responsibility, authority, placement within the organization, and composition to ensure that the design commitment to HFE is met. Also, the team should be guided by a plan to ensure that the HFE program is properly developed, executed, overseen, and documented. This plan should describe the technical program elements ensuring that all aspects of the HSI, procedures, and training are developed, designed, and evaluated on the basis of accepted HFE principles. In addition, the HFE program as a whole should appropriately consider and address the deterministic aspects of design, as discussed in RG 1.174.

Comment 3

Section II.A.5, page 18.0.7. In discussing Staffing Analysis, draft NUREG-O800 requires reviewers to verify the applicant's Staffing Analysis in accordance with NUREG-0711 criteria. However, the proposed analysis has not been demonstrated or shown cost-beneficial to safety. In addition, the proposed analysis is required by NUREG-0711 whether or not 10 CFR 50.54 requirements are met. NUREG-0800 should specify that analysis of control room staffing is not required if 10 CFR 50.54 requirements are met.

Discussion

The various detailed parts of NUREG-0711 were not subjected to cost benefit analyses and there is no requirement or plan to do so at this time.

In general, NUREG-0711 and Element 6, Staffing and Qualifications, are not limited to control room staffing only. They address the entire plant and the staff needed to ensure safety. This is discussed in Section 2.4.1 of NUREG-0711.

With respect to minimal control room staffing for power reactors,10 CFR 50.54 is the definitive starting point and compliance with it is the first review criterion the NUREG-0711 staffing element. However, the 10 CFR requirements only address licensed staff. For the circumstances in which the staff is conducting an HFE review, this review element has a broader focus than simply ensuring compliance with 50.54 requirements . For example, when plant modifications impact credited operator actions, the applicant may review the staffing needed to successfully accomplish that action. Many such actions require teamwork and communication between control room staff, auxiliary operators, and other plant staff. The NRC would review the applicant's analysis used to determine the staffing needs for accomplishing that action using guidance contained in SRP Chapter 13, "Conduct of Operations."

As a second example, when a plant and control room modernization program is proposed and the technology underlying control room operations changes significantly, the applicant may consider the impact of this change on the qualifications of plant staff. Here too, this element is used to review the applicant's analysis. The same type of review will be made for new designs.

In all of these cases, a methodology is needed to determine and to review the adequacy of staffing decisions. NUREG-0711 provides such a structured methodology.

Change Made In Response to the Comment

Section 6.1, Background, of NUREG-0711, Element 6, Staffing and Qualifications, was changed to read as follows.

Plant staff and their qualifications are important considerations throughout the design process. Initial staffing levels may be established based on experience with previous plants, staffing goals (such as for staffing reductions), initial analyses, and government regulations. Staffing levels are also an important consideration when plant modifications are designed. For example, when plant modifications impact credited operator actions, the applicant may review the staffing needed to successfully accomplish that action. Many such actions require teamwork and communication between control room staff, auxiliary operators, and other plant staff. The NRC reviews the applicant's analysis used to determine the staffing

requirements for accomplishing that action.

As a second example, when a plant and control room modernization program is proposed and the technology underlying control room operations changes significantly, the applicants evaluate the impact of the change on the qualifications of plant staff. Here too, this element is used to review the applicant's analysis.

The review criteria in this element address these situations.

Comment 4 - HRA

Comment

Section II.A.6, page 18.0-7. In discussing Human Reliability Analysis, draft NUREG-0800 refers to, "PRA/HRA analyses required by 10CFR50.34(f)(1)(i)." This suggests that the law requires HRA. To avoid confusion, NUREG-0800 should indicate instead that the PRA, if imposed, should require HRA as described in the applicable lower-level guidance.

Discussion

10 CFR 50.34(f)(1)(i) does require plants to have a PRA. And while PRAs do include HRAs the latter is not formally specified in the rule. Therefore, a change to the wording is provided below.

Change Made In Response to the Comment

The sentence in NUREG-0800 was changed to read as follows.

In addition, the review should verify that HRA activities performed in support of the HFE design are coordinated with PRA analyses required by 10 CFR 50.34(f)(1)(i) and addressed in Section 19.2 and other sections of the SRP.

Comment 5

Comment

Section II.C, page 18.0-13. This section on review of Risk-Important Human Actions is itself based on draft NUREG-1764. Review of draft NUREG-1764 suggests that the methodology may encourage analysts to produce overly conservative evaluations of the risk contributed by human actions. This result, which would be consistent with some prior findings (NUREG/CR-6355, A Limited Assessment of ASEP), would be in direct conflict with the intent of risk-informed regulations, and the NRC mandate to reduce unnecessary regulatory burden. Concerns with this guidance, when resolved, should be reflected in NUREG-0800.

Discussion

It is not clear what aspects of the review suggest the "methodology may encourage analysts to produce overly conservative evaluations of the risk contributed by human actions." Without the methodology proposed in NUREG-1764, the staff practice is to review all applications. In

contrast, our analysis using the methodology has shown that many of the evaluations result in a Region III placement which requires very little or no NRC review. Only a small percentage are placed in Region I. Therefore the net effect is an overall reduction in regulatory burden to the industry which is consistent with the intent of risk-informed regulations.

Change Made In Response to the Comment

No change was made.

3 Comments Provided by Scientech

In a email dated March 17, 2003 from Jeffery A. Julius to James Bongarra (NRC) comments were provided on Draft NUREG-1764.

Comment 1

Comment

Although the formulas for FV importance are correct, the "definitions" are stated incorrectly. On p. 16, it is stated incorrectly "FV is the fraction of the total core damage cutsets (or sequences) that contain the HA in question." It would be more correct to state that "FV is the CDF (or LERF) of the cutsets (or sequences) containing the HA divided by the total CDF (or LERF). As stated in the document, it may appear that the FV is obtained by counting the number of cutsets instead of using the cutset probabilities (or frequencies).

Discussion

The NRC reevaluated the explanatory wording used in the definition of the FV formula. This wording has been changed in the updated NUREG-1764.

Change Made In Response to the Comment

We have utilized the commenter's recommendation to update the definition.

"The FV importance measure represents a different way of expressing risk significance than RAW and is included to obtain a more robust evaluation of risk importance. FV is defined as the CDF of core damage cutsets (or sequences) that contain the HA in question, divided by the total CDF. This is expressed for HA "x" in the following equation:

$$FV(x) = \sum \text{CDF of cut sets containing x} / CDF_{\text{new BL}}"$$

Comment 2

Comment

The bases for the FV and RAW criteria displayed in Figures 2.5, 2.6, 2.7 and 2.8 should be provided either by reference or by derivation in the document. It would also be desirable to

provide the mathematical equations plotted in these figures to facilitate implementation in software, for example.

Discussion

BNL and NRC agree with this comment. A short discussion of the basis has been provided in Section 2 of the updated NUREG-1764. A more detailed discussion of the basis will appear in the separate technical basis report, [Higgins, J., O'Hara, J.(2003). The Development of an Approach for the Review of Changes to Risk-Important Human Actions-DRAFT. (Draft Report No. Y6022-2003). Upton, New York: Brookhaven National Laboratory]. For information, the portion of the draft report that addresses the particular aspect of the technical basis addressed by the comment is attached to here, as "Development of the New Split Criteria for RAW and FV Importance Measures for the Step 2 HA screening evaluation."

Change Made In Response to the Comment

The following was added to Section 2 of the updated NUREG-1764:

> Figures 2.5 and 2.6 show the Level assignments for RAW and FV analogous to the RG 1.174 Regions. The term Level was chosen to represent an amount of risk due to the HA and the corresponding amount of human factors engineering review. The levels are distinct and different from the RG 1.174 Regions. We note that these levels are not related to the three PSA Levels of CDF, source term releases from containment, and offsite consequences. RAW and FV values, which should be computed for the HA being evaluated, together with the new baseline CDF, will determine where on the Figures the HA is placed and which level of human factors review is assigned by Step 2. If an HA falls very close to the dividing line between two levels, then the reviewer may want to use the qualitative criteria in Step 3 to finally decide on the level of review for the HA (in Steps 3 & 4 of the screening process).
>
> The curve delineating the split between the Level I and Level II areas of Figure 2.5 is roughly based on a CDF of 1 E-4 core damage events per reactor-year, the subsidiary objective of the Commission's safety- goal policy statement. Performance deficiencies associated with actions in the Level I area would generally be colored as Red in the new Reactor Oversight program. Similarly, the curve delineating the split between the Level II and Level III areas of Figure 2.5 are roughly based on a CDF of 1 E-5 core damage events per reactor-year. Performance deficiencies associated with actions in the Level II area would generally be colored as Yellow, those in the Level III area would be colored as White or Green. Figure 2.6 and the FV importance measure were added to have a second, different risk importance measure that would add robustness to the method. This measure addresses HAs that may not have a high RAW value (e.g., due to a relatively high base case HEP), but which contribute notably to the base- case CDF.

Comment 3

Comment

Guidance should be provided on the risk classification of systems either by reference or explicitly in the document. On p. 25, paragraph 2 it is stated "The risk-important systems can be obtained from the plant's IPE or latest updated PSA. This information can also be extracted from the plant-specific risk-informed inspection notebooks and related benchmarking reports that have been completed by the NRC". The plant-specific risk-informed inspection notebooks produce a color designator to classify system importance, which is analogous to risk achievement. These notebooks do not produce an FV based importance. The HA risk significance is obtained by considering FV and RAW importances for CDF and LERF. It would be prudent to determine the system importances in a similar manner. With no guidance or criteria on the risk classification of systems, the classification could be arbitrary and subjective.

Discussion

The risk significance of systems and components is plant-specific. Although guidance is available on how to calculate different risk measures, risk-important systems can only be determined within the context of a licensee's IPE or latest updated PSA. The risk classification of systems itself using IPE/PSAs removes subjectivity that would otherwise enter into the classification.

Change Made In Response to the Comment

No change was made.

Comment 4

Comment

References should be provided to what are considered current, accepted PRA/HRA principles and practices. On p. 35 it is stated "…(2) performed an analysis of the potential effects of the proposed changes on plant safety and reliability, in a manner consistent with current, accepted PRA/HRA principles and practices…". Would these be PRA/HRA principles and practices accepted by the industry, the NRC or both? Has the NRC endorsed any HRA methods? The ASME PRA Standard should be referenced.

Discussion

See response to Dominion Comment 2 above.

Change Made In Response to the Comment

See response to Dominion Comment 2 above.

Comment 5

Comment

It should be clarified how a licensee without a full scope PRA is to do a Region I review. On p. 36 it is stated "Risk-important HAs associated with the modification should be identified from the PRA/HRA and used as input to the design of procedures, HSI components, and training. These actions should be developed from the Level 1 (core damage) PRA and Level 2 (release from containment) PRA including both internal and external events. They should be developed using selected (more than one) importance measures and HRA sensitivity analyses to provide adequate assurance that an important action is not overlooked because of the selection of the measure or the use of a particular assumption in the analysis". Does this imply that a licensee without an external events PRA can not do a Region I review/

Discussion

This was not intended in the original version.

Change Made In Response to the Comment

This was clarified by adding "if available." The sentence in question from Section 3.6 now reads:

> "These actions should be developed from the Level 1 (core damage) PRA and Level 2 (release from containment) PRA including both internal events and external events (if available)."

Attachment
Development of the New Split Criteria for RAW and FV Importance Measures for the Step 2 HA screening evaluation

In NUREG/CR-6689, "Proposed Approach for Reviewing Changes to Risk-Important Human Actions" (ref. 1), BNL developed a risk screening methodology that placed human actions (HAs) into risk regions (1, 2, 3) based on their risk importance. The splits between the regions were derived from the figures of RG 1.174 (ref. 2). NRC had commented that a new approach was needed that did not completely depend on using RG 1.174, particularly when examining the RAW values of the HAs. This Attachment discussed the approach to using RAW and FV importance measures for Step 2 of the screening method for NUREG-1764.

Step 2 of the method evaluates the importance of the HA once it has already been determined that the overall modification is risk important. This Attachment also discusses the criteria that were used to split the risk plane into the three regions.

For the new criteria, there are two steps, one measures the Risk Achievement Worth (RAW) value of the human action versus the new baseline CDF and the second measures the Fussell-Vesely (FV) value of the human action versus the new baseline CDF.

Figures 1 and 2, attached, present the proposed split criteria. These figures also plot a number of data points for HAs modeled in the PSAs of five NPPs so that one can see how a sample of HAs will fall into the three regions.

Below we discuss the logic behind the selection of split criteria presented in the graphs.

RAW versus Baseline CDF

Recall that RAW measures importance by computing the increase in CDF when the action in question fails. One then either takes a ratio or a difference to compute the RAW. We will use the more common ratio method of expressing RAW here (see equation 1 below). We note that NUREG/CR-6689 used the interval or difference method. They are equivalent and one can easily convert from one to the other. However, since the ratio method is more common, risk and plant personnel are more comfortable working with the ratio method. Also, most all PSA codes automatically calculate the ratio RAW values. Further, people are familiar with the various values of RAW ratios and so the split criteria will be more understandable.

Region 1 versus Region 2 line

For the graph of RAW versus the new baseline CDF, after the modification is in place, the purple line separates Region 1 from Region 2. This is based on the Commission's safety goal policy (ref. 3) of not exceeding 1x E-4 core damage events per reactor year (for shorthand we will refer to this simply as E-4 in the rest of this document). We note that exceeding the E-4 value also corresponds to a Red finding in the new Significance Determination Process (SDP) as part of the new NRC Reactor Oversight Program (ref. 4). To explain, using the newly developed Risk-Informed Inspection Notebooks, inspection findings are evaluated by failing the item (or HA) in question and developing a color (red, yellow, white or green). The notebooks

are being benchmarked against the licensees' PSAs so that a red finding corresponds to a RAW value equivalent to a CDF increase of > E-4. Further, a yellow finding corresponds to a RAW value equivalent to a CDF increase between E-5 and E-4. (See anyone of several plant specific Benchmarking reports issued by BNL.)

Thus, we want to ensure with high confidence that the delta CDF remains less than E-4 for a modified HA. As a result we will define Region 1 to be that combination of RAW values and baseline CDF that equate to a delta CDF of E-4.

The following table lists the combinations that equate to a delta CDF of E-4.

Baseline CDF	RAW
E-6	101
E-5	11
E-4	2.0
E-3	1.1

The purple line is a smoothed graph of the above data. The RAW criterion for any value of baseline CDF can be simply be calculated using the definition of RAW, in Equation 1 below.

$RAW_x = (CDF_{BL} + \Delta CDF_x) / CDF_{BL}$

Region 2 versus Region 3 line

Similar to RG 1.174 we will place this line one order of magnitude below the Region 1/Region 2 line. Using the terminology of above, this equates the lower portion of Region 2 to a delta CDF of E-5. Also Region 2 corresponds in risk importance to a Yellow finding in the new Significance Determination Process (SDP). Region 3 would correspond to a White or Green SDP finding.

The points that define a delta CDF of E-5 are given in the below table.

Baseline CDF	RAW
E-6	11
E-5	2.0
E-4	1.1

For the Region 2/Region 3 line we use these values with the one exception that we drop the line to a RAW value of 1.0 at E-4 for conservatism. Thus any HA for at a plant with a revised baseline CDF greater than E-4 the HA would always be at least in Region 2.

When the new proposed Regions were completed we compared them to the data points for the

127 HAs modeled in the PSAs of five NPPs. We see that the sample of HAs are distributed among the three regions. One concern may be that there seem to be "too many" HAs in Region 1. Also, we note that this sample of plants did have a moderate number of HAs with high RAW values and relatively high base case CDFs.

If one were to plot the RAW acceptance criteria from the risk-informed SERs for South Texas and Comanche Peak, they fall roughly in the same regime as these. However, they are horizontal lines and don't vary with baseline CDF. They have horizontal lines at RAW values of 100, 10, and 2.

FV versus Baseline CDF

There is less absolute basis already established in the industry for a FV criterion than for a RAW criterion. Nonetheless, FV definitely represents a different way of expressing risk than RAW and the NRC has requested that we incorporate it into our criteria. Recall that FV is the fraction of the total core damage cutsets (or sequences) that contain the action in question.

Region 1 versus Region 2 line

The initial recommendation for this split line started at a FV value of 0.5 (or 50%). This was judged by NRC as too high, Thus, the consensus was to change the 50% criterion that would allow one HA to contribute to 50% of the cutsets. It was felt that would be placing too many of our "core damage prevention eggs" in one basket. Therefore, we have lowered the Region I vs. Region II split line from 50% to 10% in order to improve defense in depth. Also, we now keep the (Reg. I - Reg. II) FV line constant, until we reach a higher value of base case CDF at 1 E-4.

At this point (base case CDF of E-4) the absolute contribution of the cutsets to CDF becomes noticeable. That is, we note that 10% of the CDF is E-5, and that E-5 is the Region I criterion of RG 1.174. Therefore, it becomes appropriate to decrease the criterion line as CDF increases further. So, for baseline CDFs above E-4 we will extend the line such that an E-5 (or greater) contribution of the cutsets would put you in Region I. Thus, for a baseline CDF of E-3, the FV value for the split line is 0.01. We thus extend the split line from the point (E-4, .1) to the point (E-3, 0.01).

Region 2 versus Region 3 line

To select this criteria line we first make a few observations. If one were to plot the South Texas Plant (STP) and Comanche Peak (CP) FV acceptance criteria (from the risk-informed SERs) they fall as horizontal lines and don't vary with baseline CDF. They have horizontal lines at FV values of 0.01, .0005, and 0.001. This gives us a first neighborhood for the split line. In this neighborhood, we generally don't have sufficient contribution of the cutsets to CDF to use CDF contribution as a setting basis. We would also like to be approximately one order of magnitude below the Reg. I - Reg. II split line. Thus, we select a constant value of 0.005 as the Reg. II - Reg. III split line.

The South Texas and Comanche Peak FV acceptance criteria (from the risk-informed SERs) have horizontal lines at FV values of 0.01, 0.005, and 0.001 and that don't vary with baseline CDF. Thus, they fall roughly across Regions 2 and 3. This also seems reasonable.

Combination of RAW and FV

As noted in the beginning of this document, the RAW and FV screening would be applied after the initial screening for the entire modification using the RG 1.174 methods. One should use both the RAW screen and the FV screen and then take the most conservative of the two. As a test or example case, we again use the 127 data points for the HAs modeled in the PSAs of five NPPs. The Table below shows the number of Region 1, 2 & 3 HAs that fall into each Region using only the RAW measure, only the FV measure, and then using a Two Step process that is the most conservative of the two measures.

	RAW	FV	Two Step Process	
			number	%
Region 1	35	11	42	33
Region 2	41	61	53	42
Region 3	51	55	32	25
Total	127	127	127	100

TABLE 1 Summary of HA distribution among the three Risk Regions

We see that the largest number of HAs fall into Region 2, with a fair percentage in each of Region 1 and Region 3. Two other points to note are first that not all of the submitted HAs will be risk significant enough to be modeled in the PSA, hence that would add some additional Region 3 items. Secondly, the qualitative screening criteria of Step 3 will have some effect and would change the overall distribution somewhat.

Attachment References

1. Proposed Approach for Reviewing Changes to Risk-Important Human Actions, NUREG/CR-6689, BNL-NUREG-52598, J. Higgins, J. O'Hara, and W. Stubler, October, 2000

2. Regulatory Guide 1.174, An approach to using probabilistic risk assessment in risk-informed decision-making on plant-specific changes to the licensing basis, 1998

3. Safety Goals for the Operation of Nuclear Power Plants; Policy Statement, Federal Register, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986.

4. New NRC Reactor Inspection and Oversight Program, NUREG-1649, February, 1999.

5. Qualitative Insights to be Applied in Risk Screening, J. O'Hara, Rev. 1, Update 12/27/01

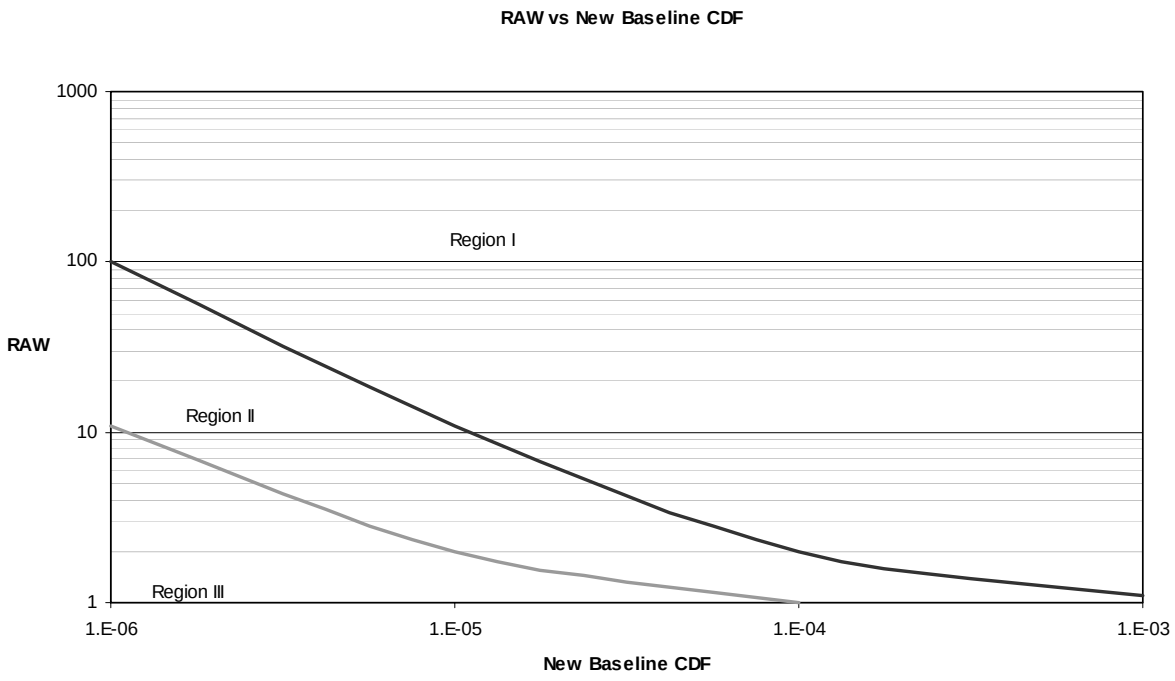

Figure 1. RAW vs. New Baseline CDF

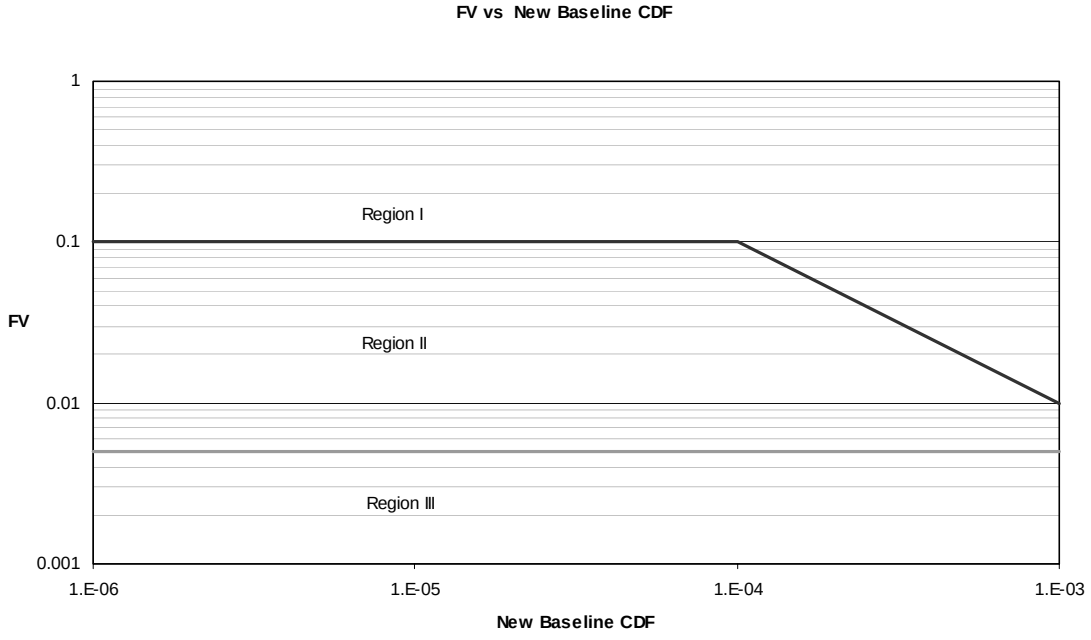

Figure 2. FV vs. Baseline CDF