NUREG-1764, Rev. 1

Guidance for the Review of Changes to Human Actions

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: U.S. Nuclear Regulatory Commission
Office of Administration
Mail, Distribution and Messenger Team
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

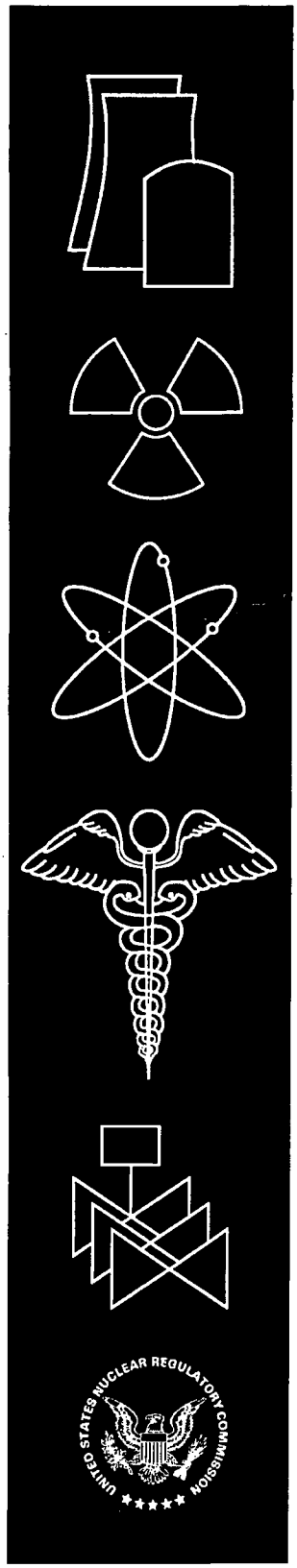
The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).



NUREG-1764, Rev. 1

Guidance for the Review of Changes to Human Actions

Manuscript Completed: January 2005
Date Published: September 2007

Prepared by:
J.C. Higgins and J.M. O'Hara
Energy Sciences and Technology Department
Brookhaven National Laboratory
Upton, NY 11973-5000

P.M. Lewis, J.J. Persensky, J.P. Bongarra, S.E. Cooper, and G.W. Parry
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Prepared for:
Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) reviews changes in human actions, such as those that are credited in nuclear power plant safety analyses. This document provides guidance for reviewing those changes. In this document, the terms "human action" and "operator action" are used synonymously because most of the types of actions discussed are performed by operations staff. The evaluation method uses a two-phased approach. The first phase is a screening analysis of the licensee's proposed modification and the affected human actions to assess their risk-importance. A graded, risk-informed approach is used to determine the appropriate level of human factors engineering review. This approach can be accomplished for licensee submittals that are either risk-informed or non-risk-informed. For risk-informed submittals, the first phase has four steps: (1) use of Regulatory Guide (RG) 1.174 to determine the risk-importance of the entire plant change or modification that involves the human action, (2) quantification of the risk-importance of the human action itself, (3) qualitative evaluation of the human action, and (4) integrated assessment to determine the appropriate level of human factors engineering review. For non-risk informed submittals, a similar process is used which includes the use of generic risk information to determine the safety significance of the HA in place of the first two steps used in a risk-informed submittal.

The proposed human actions are assigned to one of three risk levels (high, medium, and low) as a result of Phase 1. The level of human factors engineering review in the second phase corresponds to these risk levels. In the second phase, human actions are reviewed using standard human factors engineering criteria to ensure the appropriate conditions are in place so that the change in human action does not significantly increase the potential for risk. Human actions in the high risk level receive a detailed human factors engineering review, while those in the medium risk level undergo a less detailed review, commensurate with their risk. For human actions in the low risk level, there is a minimal human factors engineering review or none. The NRC's review of licensee submittals that involve changes to human actions is an iterative process. The final results of the human factors engineering review provide input to integrated decision-making and a safety evaluation report.

This revision (Rev. 1) contains relatively minor changes and clarifications in the text (virtually all of them in Chapter 2). Appendix B lists the most significant changes.

Paperwork Reduction Act Statement

This NUREG does not contain information collection requirements and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

FOREWORD

The United States Nuclear Regulatory Commission (US NRC) staff reviews license amendments and actions involving plant changes that affect human actions to determine whether the proposed human action can be reliably performed when needed. This document provides guidance for NRC staff to use in reviewing changes to human actions (HAs), such as those that are credited in nuclear power plant safety analyses. To better utilize staff resources, the process recommended in this document provides a risk-informed, graded approach to tailor the level of resources used by NRC staff for review to be commensurate with the level of risk associated with the proposed human action. NUREG-1764, Revision 0 was used as a basis for revisions to the Standard Review Plan (SRP), Chapter 18 and Chapter 19 in 2004 and is referenced in the SRP. This current revision is the result of experience with and further testing of the risk screening methodology described herein and will be referenced in the SRP, Chapters 18 and 19.

NRC developed this document to (1) consolidate and standardize review guidance that previously existed in several other documents, and (2) make the review guidance more risk-informed. A two-phased approach is used to review human actions and determine their acceptability.

Phase 1 is a risk-informed screening process to determine the level of human factors engineering (HFE) review that is appropriate in Phase 2. For risk-informed submittals, staff would use the risk information provided by the licensee in accordance with Regulatory Guide (RG) 1.174 as the Phase 1 screening. For non-risk-informed submittals, a different Phase 1 process is used to determine the safety-significance of the HA without the benefit of the risk inputs that would be provided in risk-informed submittals. In these cases the NRC staff performs a generic risk evaluation to estimate the risk-importance of the proposed change to the HA in accordance with the guidance in SRP, Chapter 19.0.

Phase 2 is the HFE review, which is divided into three levels (i.e. high, medium, low) depending on the risk classification from Phase 1, whether the submittal is risk-informed or non risk-informed. The HFE review criteria for high, medium, or low levels were adapted from NUREG-0711.

Following Phase 2, staff renders a decision as to the acceptability of the proposed change, considering several factors, including change in risk, human factors criteria, deterministic criteria, the time for which the modification will be in place, and supplemental decision factors related to the unique characteristics of the situation. The results of the HFE review provide input to integrated decision-making and a safety evaluation report as described in RG 1.174.

Farouk Eltawila, Director
Division of Risk Assessment and Special Projects
U.S. Nuclear Regulatory Commission

CONTENTS

CONTENTS (cont'd.)

Appendices

CONTENTS (cont'd.)

Figures

Tables

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) reviews changes in human actions (HAs), such as those that are credited in nuclear power plant safety analyses. Changes in credited HAs may result from a variety of plant activities such as plant modifications, procedure changes, equipment failures, justifications for continued operations, and identified discrepancies in equipment performance or safety analyses. Relevant considerations for review are described in NRC information notices and generic issues. Generic Letter 91-18 (NRC, 1991) discusses the conditions under which manual actions may be used in lieu of automatic actions for safety-system operations. Information Notice 97-78 (NRC, 1997) alerts licensees to the importance of considering the effects on human performance of such changes made to plant safety systems.

This document provides guidance for use in determining the appropriate level of human factors engineering (HFE) review of HAs based upon their risk-importance. This guidance uses a graded, risk-informed approach that is consistent with Regulatory Guide (RG) 1.174, Rev. 1 (NRC, 2002e). As such, this guidance uses risk insights to determine the level of regulatory review the staff should perform. This approach can be accomplished for licensee submittals that are either risk-informed or non-risk-informed. Human actions that are considered more risk-significant receive a detailed review, while those deemed less significant receive a less detailed review. In this document, the terms "human action" and "operator action" are used synonymously because most of the types of actions addressed are performed by operations staff. In keeping with RG 1.174, this guidance does not preclude licensees from using other approaches in requesting changes to a plant's licensing basis or HAs. Rather, this method of reviewing HAs is intended to improve consistency in regulatory reviews and decisions.

This guidance uses a two-phased approach to reviewing HAs. Phase 1 is a risk screening and analysis of the affected HAs identified by the licensee to determine their risk-importance and the level of HFE review that is appropriate in Phase 2. The second phase is an HFE review of those HAs that are found to be risk-important. The next two subsections describe these phases in greater detail.

Risk Screening Process

A plant change may include changes to equipment, as well as HAs. This approach focuses on the HAs, whereas risk screening of equipment changes can be accomplished using RG 1.174. Section 2.3 discusses the risk screening for risk-informed licensee submittals, while Section 2.4 discusses the risk screening for those that are non-risk-informed.

For risk-informed submittals, the staff uses a four-step screening process to locate the plant modification and its associated HAs in risk space, using guidance similar to that of RG 1.174. The first two steps are quantitative, while the third is qualitative and the fourth involves an integrated assessment that considers the results of the first three steps and determines the appropriate level of HFE review. Essentially, plant modifications and their associated HAs can be categorized into regions of high, medium, and low risk, and this categorization determines the appropriate level of graded HFE review. The following paragraphs discuss the important steps of this four-step screening process.

Before submitting a change request to the NRC the licensee reviews a proposed plant change to identify HAs that constitute new or modified actions, or involve modified task demands. The licensee also conducts an evaluation, in accordance with Title 10, Section 50.59, of the *Code of Federal Regulations* (10 CFR 50.59), to identify any changes that affect the plant's final safety analysis report (FSAR). This evaluation may identify associated activities that require the NRC's review and approval before they are implemented.

For a risk-informed review, the licensee would perform an initial risk screening calculation, which would then be submitted to the NRC with the licensee's request for approval of the change. The first step in the risk screening evaluates the full modification, including both equipment and HAs, and is conducted using RG 1.174. Risk calculations include the change in risk or core damage frequency (CDF) attributable to the full modification (ΔCDF_{mod}), including the HA. This assessment may determine that the full modification, including the HA, is in Region III (low risk). If so, in some cases, the only necessary NRC review is an evaluation of whether there is a valid technical basis for the low risk.

The risk screening calculations also consider whether the proposed change is permanent or temporary. If temporary, the screening considers the length of time the change will be in place. Then, the method assesses the integrated risk attributable to the change over the time that it will be in place (i.e., the integrated conditional core damage probability, or ICCDP). Similar calculations would be performed for large early release frequency (LERF), where appropriate.

The second step of the risk screening process uses both the Risk Achievement Worth (RAW) and the Fussell-Vesely (FV) risk-importance measures to determine the risk-significance of the HA. Toward that end, this step identifies the effect on risk of failing to perform the HA (using the RAW), as well as the action's relative contribution to risk (using the FV importance). Uncertainty about HAs is treated by setting the HA failure probability to 1.0 for the action under review. This second step of the risk screening process tentatively places the HA in one of three risk levels (high, medium, or low) to determine the level of HFE review to be performed by the NRC. The importance of the HA with respect to both CDF and LERF is then assessed. For HAs determined to be risk-significant (i.e., in the high or medium risk levels), the intent of the detailed HFE review is to ensure the appropriate conditions are in place so that the change in human action does not significantly increase the potential for risk.

The third step of the risk screening is qualitative, and allows the NRC to adjust the quantitative evaluation from Step 2, if necessary, considering factors for which the probabilistic risk assessment (PRA) model cannot quantitatively account. This step considers several factors, such as personnel functions and tasks, design support for task performance, and performance shaping factors.

The fourth step is an integrated assessment that considers the results of the first three steps and determines the appropriate level of HFE review. If the HA is verified to be in the low risk level, the NRC would likely allow the licensee's change with either minimal or no further HFE review. If the HA is in the medium risk level, the NRC would undertake a moderate, top level HFE review. If the action is in the high risk level, the NRC would conduct a more detailed review, including HFE, deterministic reviews, and risk analysis.

Human Factors Engineering Review

In this phase, the NRC reviews the proposed changes to HAs to ensure the appropriate conditions are in place so that the change in human action does not significantly increase the potential for risk. Again, the details of the review are commensurate with the risk. Three levels of risk and NRC review are presented. The review criteria are based on an adaptation of existing NRC review guidance for HFE, as found in NUREG-0800 (NRC, 2004a), NUREG-0711 (NRC, 2004b), NUREG-0700, (NRC, 2002b), and IN 97-78 (NRC, 1997).

A Level I review is used for HAs in the high risk category, which require the most stringent review. A Level I review includes most of the review elements from NUREG-0711, the NRC's "Human Factors Engineering Program Review Model."

The NRC conducts a Level II review for HAs in the medium risk category. While the guidance addresses topical areas similar to those in the Level I review, the extent of the staff's review is notably less for Level II. The evaluation processes for this level are less prescriptive and afford greater latitude to both the licensee and the NRC reviewers for collecting and analyzing information. The Level II evaluation process addresses general deterministic review criteria, analysis, human-system interface (HSI) design, procedures, training, and HA verification.

HAs in the low risk category receive a Level III review by the NRC, which is generally limited to verifying that the HA is in fact in Level III. Typically, no detailed HFE review is necessary. However, the NRC review may include a few review areas based on the results of Step 3 of the risk screening process. Licensees may choose to use the Level II guidance to address HFE considerations for HAs that fall into Level III.

Final Decision on Acceptance of Human Actions

The NRC's review of licensee submittals involving changes to HAs is an iterative process. That is, the final results of the HFE review provide input to integrated decision-making (see RG 1.174, Section 2.2.6), and may be documented in a safety evaluation report.

The results of the various analyses are considered in an integrated manner (i.e., the decision is not solely driven by the numerical results of the risk assessment). This approach complements the NRC's deterministic approach; supports its traditional defense-in-depth philosophy; considers traditional engineering, HFE, and risk information; and uses both qualitative and quantitative analyses and information. The decision-making process considers the following major factors:

- **Change in CDF**: The increase in CDF attributable to the modification (ΔCDF_{mod})
- **Change in LERF**: The increase in LERF attributable to the modification ($\Delta LERF_{mod}$)
- **Risk-Importance Measures for the HA**: The values of the RAW and FV risk-importance measures
- **Time and Integrated Risk**: Risk integrated over the length of time that a temporary change will be in place

- **Human Factors**: The basis that operators can perform the actions required for the modification, as determined by the HFE review criteria that the NRC used for the review.
- **Deterministic Criteria**: Satisfaction of the deterministic review guidance provided in Section 3.1 of the Level I review guidance or Section 4.1 of the Level II review guidance.

The guidance document also addresses additional/supplemental factors that could influence the acceptability of a given change.

ACKNOWLEDGMENTS

Several individuals were involved in developing this document. The principal contributors were James C. Higgins and John M. O'Hara of Brookhaven National Laboratory. In addition, Susan E. Cooper of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES), and Gareth W. Parry of the NRC's Office of Nuclear Reactor Regulation (NRR) updated and modified the risk screening methodology in Chapter 2. Paul M. Lewis and J. J. Persensky of RES provided overall management for preparation of this document, while James P. Bongarra, Jr., of NRR was the technical coordinator and a reviewer. Other NRC staff members who reviewed this document were Richard J. Eckenrode, Clare P. Goodman, Michael C. Cheok, Martin Stutzke, and Michael Franovich of NRR, as well as Nathan O. Siu and Erasmia Lois of RES.

ABBREVIATIONS

AC	alternating current
ADS	automatic depressurization system
AFW	auxiliary feedwater
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOT	allowed outage time
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BWR	boiling-water reactor
CCDF	cumulative core damage frequency
CDF	core damage frequency
CFR	*Code of Federal Regulations*
DBE	design-basis event
DC	direct current
DEP	depressurization
DG	draft regulatory guide
ECCS	emergency core cooling system
EOP	emergency operating procedures
FSAR	final safety analysis report
FV	Fussell-Vesely
FW	feedwater
HA	human action
HEP	human error probability
HFE	human factors engineering
HPCI	high-pressure coolant injection
HRA	human reliability analysis
HSI	human-system interface
IC	isolation condenser
ICCDP	incremental conditional core damage probability
ICLERP	incremental conditional large early release probability
IMC	Inspection Manual Chapter
IN	information notice
IPE	individual plant examination
JCO	justification for continued operations

ABBREVIATIONS (cont'd.)

LB	licensing basis
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPI	low-pressure injection
MSLB	main steam line break
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OER	operating experience review
OMB	Office of Management and Budget
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSF	performing shaping factor
PWR	pressurized-water reactor
RAI	request for additional information
RAW	Risk Achievement Worth
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RG	regulatory guide
RI	risk-informed
RIS	regulatory issue summary
SBO	station blackout
SG	steam generator
SGTR	steam generator tube rupture
SLC	standby liquid control
SPAR	standardized plant analysis of risk
SRA	senior reactor analyst
SROA	safety-related operator action
SRP	Standard Review Plan (NUREG-0800)
SRV	safety relief valves
SSC	structure, system, and/or component
TA	task analysis
USQ	unreviewed safety question
V&V	verification and validation

1. INTRODUCTION

In Enclosure 2 to Generic Letter 91-18 (NRC, 1991), the U.S. Nuclear Regulatory Commission (NRC) discussed the conditions under which manual actions may be used in lieu of automatic actions for safety-system operations. Information Notice (IN) 97-78 (NRC, 1997) alerted licensees to the importance of considering the effects on human performance of such changes made to plant safety systems:

> The original design of nuclear power plant safety systems and their ability to respond to design-basis accidents are described in licensees' FSARs [final safety analysis reports] and were reviewed and approved by the NRC. Most safety systems are designed to rely on automatic system actuation to ensure that the safety systems are capable of carrying out their intended functions. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual action for automatic system actuation or that modify existing operator actions, including operator response times, that were not reviewed and approved during the original licensing review of the plant may raise the issue of an unreviewed safety question (USQ). Such changes must be evaluated under the criteria of 10 CFR 50.59 to determine whether a USQ is involved and whether NRC's review and approval are required before implementation.... In the NRC staff's experience, many of the changes involving operator actions proposed by licensees do involve a USQ. (p. 3)

The authors recognize that the NRC updated Title 10, Section 50.59, of the *Code of Federal Regulations* (10 CFR 50.59) to remove the USQ wording. Nonetheless, the intent of IN 97-78 is still pertinent. That is, licensees still need to submit many changes in operator actions to the NRC for review and approval in accordance with the revised 10 CFR 50.59.

The guidance presented in this document can be used to address safety-related operator actions (SROAs), as well as other required operator actions. The American National Standards Institute/American Nuclear Society defined "safety-related operator action" in ANSI/ANS-58.8-1994, as follows:

> A manual action required by plant emergency procedures that is necessary to cause a safety-related system to perform its safety-related function during the course of any DBE (design-basis event). The successful performance of a safety-related operator action might require that discrete manipulations be performed in a specific order. (p.4)

Licensee requests may involve changes in human actions (HAs) that result from the following types of plant activities:

- plant modifications
- procedure changes
- equipment failures
- justifications for continued operations (JCOs)
- identified discrepancies in equipment performance or safety analyses

The licensee's request should include an evaluation of such plant activities to determine their effects on HAs. Specifically, the following types of HA effects may result from these changes:

- **New action**: An action that was not previously performed by personnel, such as when an action formerly performed by automation is allocated to the operators.
- **Modified action**: A change in the way an action was previously performed, such as introducing new task steps (e.g., as a result of new system components, a modification to a component, or failed components), or new control and display devices for performing the action.
- **Modified task demand**: Rather than affecting the task steps themselves, a change in the plant may affect the task demands, such as the amount of time available or the overall environment for the task.

This document proposes using a graded, risk-informed approach in conformance with Regulatory Guide (RG) 1.174 (NRC, 2002e) and provides guidance for reviewing the human performance aspects of changes to plant systems and operations. Risk insights are used to determine the level of regulatory review the staff should undertake. HAs that are considered more risk-significant receive a detailed review, while those deemed less risk-significant receive a less-detailed review commensurate with their risk. In this document, the terms "human action," and "operator action" are used synonymously because most of the types of actions discussed are performed by operations staff.

The evaluation method uses a two-phased approach. The first phase is a four-step screening process to locate the plant modification and its associated HAs in risk space, using guidance similar to that of RG 1.174. The first two steps are quantitative, while the third is qualitative and the fourth involves an integrated assessment that considers the results of the first three steps and determines the appropriate level of human factors engineering (HFE) review. Essentially, plant modifications and their associated HAs can be categorized into regions of high, medium, and low risk, and this categorization determines the necessary level of graded HFE review. The following paragraphs discuss the important steps of this four-step screening process.

In the second phase (the HFE review), the NRC reviews the proposed HAs to verify that they can reliably be performed when needed. Again, the details of the review are commensurate with the risk. There are three levels of NRC review, and the review criteria are based on an adaptation of existing NRC review guidance for human factors, as found in NUREG-0800 (NRC, 2004a), NUREG-0711 (NRC, 2004b), NUREG-0700, (NRC, 2002b), and IN 97-78 (NRC, 1997). The development of the review criteria was facilitated accomplished by analyzing past cases reviewed by the NRC (Higgins, et al., 1999).

A Level I review is used for HAs in the high risk category, which require the most stringent review. (See Chapter 3 of this document.) A Level I review includes most of the review elements from NUREG-0711, the NRC's "Human Factors Engineering Program Review Model." As such, it examines the licensee's planning, analysis, design activities, and verification and validation related to the change. While HAs in the high risk area (Region I) are generally not desired, there are certainly examples of such actions in plants today, such as the pressurized-water reactor (PWR) emergency core cooling system (ECCS) switchover. Also, there may be extenuating circumstances in which a licensee can adequately justify a modification to add a Region I HA [e.g., if the change is temporary or there are other changes that reduce the core

damage frequency (CDF).] Another important consideration is how well the licensee addressed the HFE aspects of the modification.

The NRC conducts a Level II review for HAs in the medium risk category. While the guidance addresses topical areas similar to those in the Level I review, the extent of the staff's review is notably less for Level II. (See Chapter 4 of this document.) The evaluation processes for this level are less prescriptive and afford greater latitude to both the licensee and the NRC reviewers for collecting and analyzing information. The Level II evaluation process addresses general deterministic review criteria, analysis, human-system interface (HSI) design, procedures, training, and HA verification.

Finally, HAs in the low risk category receive a Level III review by the NRC, which is generally limited to verifying that the HA is in fact in Level III. Such a verification can be accomplished by reviewing the licensee's analysis methods and risk results that show the placement of the action in Level III. Typically, no HFE review is necessary. However, the NRC may address a few review areas based on the results of Step 3 of the risk screening process. Licensees may choose to use the Level II guidance to address human factors considerations for HAs that fall into Level III.

In keeping with RG 1.174, this guidance does not preclude licensees from using other approaches in justifying changes to a plant's licensing basis or requesting changes in HAs. Rather, this method of reviewing HAs is intended to improve consistency in regulatory reviews and decisions in areas where the results of risk analyses are used to help justify regulatory action. RG 1.174 notes that the risk-informed principles, processes, and approaches provide useful guidance for applying risk information to a broader set of activities than plant-specific changes to a plant's licensing basis. This document was developed within the spirit of such applications.

RG 1.174 notes that the use of probabilistic risk assessment (PRA) technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data. Its application should complement the NRC's deterministic approach and support the NRC's traditional defense-in-depth philosophy. The NRC's review of HAs also considers this concept.

In addition, RG 1.174 notes that decisions concerning proposed changes are expected to be reached in an integrated fashion, considering traditional engineering and risk information. They may be based on qualitative factors, as well as quantitative analyses and information. Thus, the approach presented herein also considers such qualitative factors, both in Step 3 of the risk screening process, and in the final decision on acceptance of human actions. Thus, the approaches described retain some deterministic aspects (for example, dealing with defense-in-depth, meeting existing regulatory requirements, and addressing the HFE aspects of the HAs).

The NRC's review of risk-informed changes that involve HAs is an iterative process. The final results of the HFE review provide input to integrated decision-making (see RG 1.174, Section 2.2.6), and may be documented in a safety evaluation report.

By implementing the guidance in this document, the NRC will improve the regulatory process (1) by enhancing safety decision-making by using PRA insights, (2) through more efficient use of agency resources, and (3) by reducing unnecessary burdens on licensees. The use of risk

insights in licensee submittals that request plant changes to HAs will assist the staff in the disposition of such licensee proposals.

The technical basis and development of this review guidance is provided in Higgins and O'Hara (2004).

2. RISK SCREENING PROCESS

2.1 Licensee Change Requests Involving Human Actions

As discussed in Chapter 1 of this document, licensee submittals may involve changes to HAs resulting from a variety of plant activities. These activities may result in new or modified human actions, or modified task demands. This section provides guidance for use in determining the level of HFE review required for such HA changes using a risk-informed screening process. This approach can be accomplished for licensee submittals that are either risk-informed or non-risk-informed. Section 2.3 discusses the risk screening for risk-informed licensee submittals, while Section 2.4 discusses the risk screening for those that are non-risk-informed.

A particular modification or submittal may involve one or several HAs. Most submittals clearly identify the affected HAs, but some involve an HA that is not clearly identified. There may also be submittals that do not specifically identify any affected HAs, although the modification for which the licensee is requesting review actually affects one or more HA(s). (Section 2.3.4 discusses some considerations for use in ensuring that all appropriate HAs are identified and screened.) In addition, in some submittals, some of the HAs associated with the modification may appear to be unchanged. In such cases, the HAs should still be subjected to the screening process. Step 3 of the process includes provisions for verifying that a given HA has not been changed and reducing the level of associated review.

2.2 Overview of the Screening Process

The screening process for determining the appropriate level of human factors (HF) review of HAs has two major pathways, based on the type of information contained in the licensee's change request:

(1) risk-informed submittals
(2) non-risk-informed submittals

Regulatory Guide (RG) 1.174 and Chapter 19 of the Standard Review Plan (SRP, NUREG-0800) provide guidance for reviewing risk-informed (RI) licensee change requests, as well as guidance on when the NRC requests risk information from the licensee. For RI submittals, adequate risk information should be available to assist the NRC staff in determining the appropriate level of HFE review. Section 2.3 discusses the methodology for making this determination.

This document provides guidance in two areas for non-risk-informed (non-RI) licensee change requests. The first area involves assessing the appropriateness of a non-RI submittal for the HA(s), based on the guidance in RG 1.174 and SRP, Chapter 19 for when the NRC can request risk information from the licensee. If that assessment determines that the licensee should submit the appropriate risk information for the change to be considered, the submittal becomes RI, so the guidance for RI submittals should be used. The second area addresses those remaining submittals that are non-RI, for which risk inputs will not be available to assist the staff in determining the appropriate level of HFE review. The basis for such requests might be a licensee safety analysis using 10 CFR 50.59. In this case, guidance for determining the

appropriate level of HFE review in the absence of explicit, plant specific risk inputs is based on Chapter 19 of the SRP, as well as more general human reliability and risk concepts provided herein. Section 2.4 discusses the methodology for making this determination for non-RI licensee change requests.

Section 2.5 briefly describes the recommended levels of HFE review, which are the outputs of the RI and non-RI screening processes. Chapters 3 and 4 present the guidance for conducting the HFE review.

2.3 Screening Process for Risk-Informed Change Requests

RG 1.174 and SRP Chapter 19 provide guidance for use in evaluating RI requests for changes to a plant's licensing basis (LB). A PRA analyst performs this analysis, however, an HFE review is one of the inputs to the integrated decision-making described in RG 1.174. The guidance in this section supplements that in RG 1.174 and SRP Chapter 19 to support the determination of the appropriate level of HFE review.

2.3.1 Overall Screening Approach for Risk-Informed Requests

The overall screening approach for RI requests consists of the following four steps:

(1) assignment of the change request into Region I, II, or III using RG 1.174

(2) calculation of importance measures for an HA involved in the change request

(3) qualitative assessment of the safety-significance of an HA involved in the request for LB change

(4) integrated assessment of HA safety-significance for determining the appropriate level of HFE review (i.e., Level I, II, or II).

Sections 2.3.2 and 2.3.3 describe the first step in the screening process for RI requests. Sections 2.3.4 and 2.3.5 describe Steps 2 and 3 in the overall RI screening process for HA reviews.

The results of Steps 1, 2, and 3 are inputs to Step 4, the integrated assessment of HA safety significance. Section 2.3.6 describes how this integrated assessment is performed and the resulting evaluation of the appropriate level of HFE review.

2.3.2 Assignment into RG 1.174 Acceptability Regions (Step 1)

An NRC PRA analyst reviews the licensee's submittal and assigns it into one of three acceptability regions using RG 1.174 and SRP Chapter 19. For this first step in the screening process, there are two possibilities:

- The request is for a *temporary* LB change.
- The request is for a *permanent* LB change.

RG 1.174 does not directly address requests for temporary LB changes. Section 2.3.3 of this report describes how region assignments can be made for such requests for the purposes of HFE review screening. This approach is similar to that previously used by NRC staff to evaluate requests for temporary changes.

If the request is for a permanent LB change, the guidance provided in RG 1.174 is used directly to determine into which of the three "regions" (i.e., Region I, II, or III) the overall request will be assigned. This assignment is one input to the integrated assessment of the acceptability of the proposed change that involves the HA.

Permanent changes may include equipment only, human actions only, or a combination of both. Equipment only changes, which have no impact on HAs, are not within the purview of this approach. Changes that involve HAs only, HAs plus equipment changes, or equipment changes that affect HAs should be evaluated using RG 1.174 as the first step.

RG 1.174, Figure 3, "Acceptance Guidelines for Core Damage Frequency (CDF)," and Figure 4, "Acceptance Guidelines for Large Early Release Frequency (LERF)," have three risk regions (I, II, and III), the most risk-significant being Region I. These figures are reproduced here for convenience.

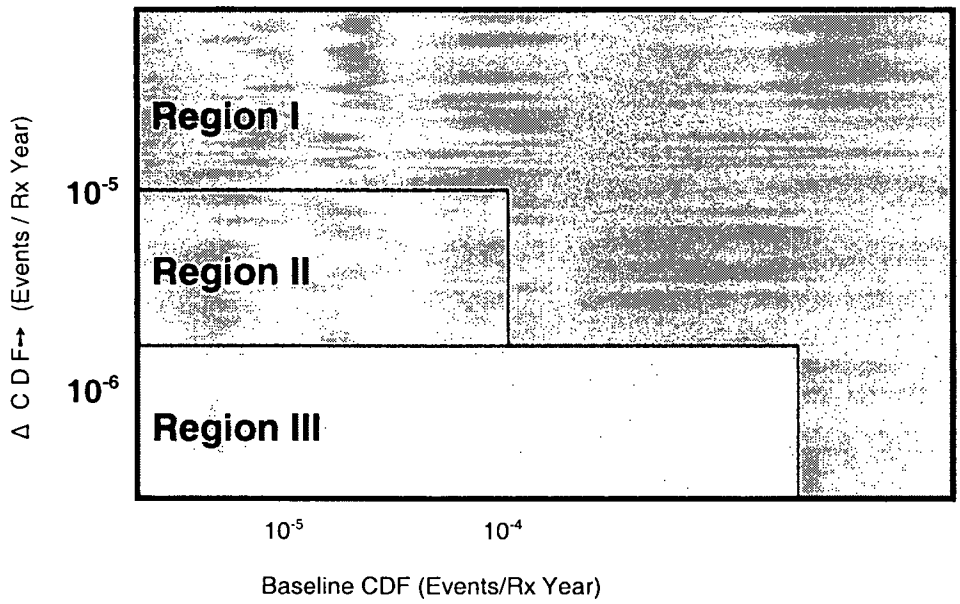


Figure 2.1 Acceptance Guidelines for Core Damage Frequency (CDF) (from RG 1.174, Figure 3)

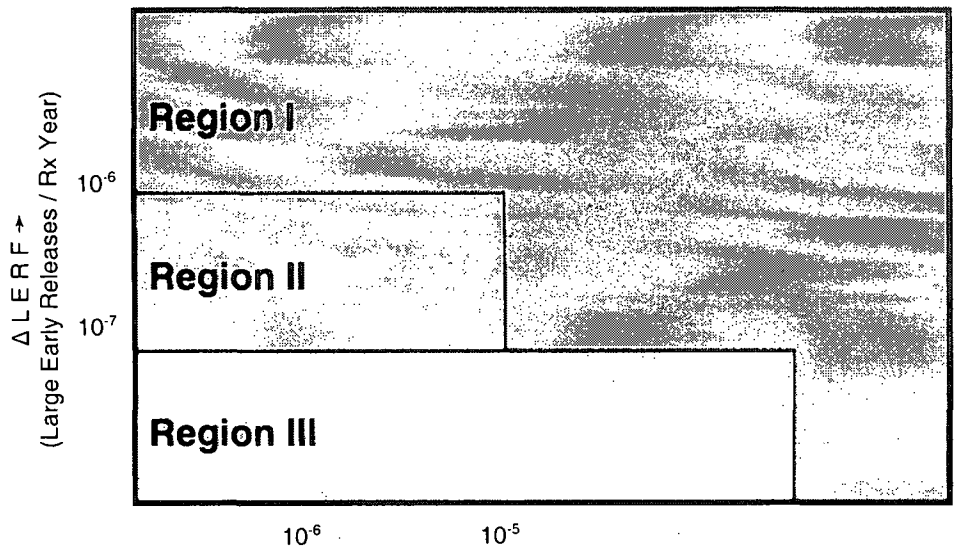


Figure 2.2 Acceptance Guidelines for Large Early Release Frequency (LERF) (from RG 1.174, Figure 4)

The change in core damage risk attributable to the modification (ΔCDF_{mod}) that includes the new HA is defined as follows:

$$\Delta CDF_{mod} = [\text{new CDF (with modification in-place)} - \text{current baseline CDF}]$$

where, ΔCDF_{mod} is the change in CDF attributable to the modification. This value of ΔCDF_{mod} is placed in one of the three regions of Figure 3 of RG 1.174, for Step 1 of the screening process.

Similarly, the change in risk attributable to LERF is evaluated using Figure 4 of RG 1.174. The change in large early release risk attributable to the modification ($\Delta LERF_{mod}$) that includes the new HA is defined as follows:

$$\Delta LERF_{mod} = [\text{new LERF (with modification in-place)} - \text{current baseline LERF}]$$

where, $\Delta LERF_{mod}$ is the change in LERF attributable to the modification. This value of $\Delta LERF_{mod}$ is placed in one of the three regions of Figure 4 of RG 1.174, for Step 1 of the screening process. The region for Step 1 of the screening is the most conservative of Figure 3 or Figure 4 of the RG. If it is in Region I, the NRC will likely (but not definitely) disapprove the modification; this is discussed in more detail below. If the modification is in Regions II or III, the reviewer proceeds to Step 2 of the screening process, as summarized in Table 2.1 below.

Table 2.1 Action on Completion of Step 1

Step 1 Results	NRC Review Action
Region I (Equip. + HA)	- Change generally not permitted. - If change not disapproved at Step 1, go to Step 2 of screening.
Region I (HA only)	- Change generally not permitted. - If change not disapproved at Step 1, perform Region I HFE review.
Region II or III	- Go to Step 2 of screening.
Note: When using this table for a temporary modification, use Region I^{intgr}, II^{intgr}, III^{intgr}.	

Modification in Region I for Step 1

RG 1.174 notes that licensee applications that are in Region I "would not normally be considered." However, it notes that "the acceptance guidelines should not be interpreted as being overly prescriptive." There may be extenuating circumstances where they would be considered and approved, including the following reasons:

- unquantified benefits that are not reflected in the quantitative risk results
- compensatory measures proposed to counter the impact of major risk contributors

If a Region I modification includes a combination of both equipment and HAs, the NRC may reject the overall modification, and no further screening is necessary. If such a modification is not rejected, screening proceeds to Step 2 to evaluate the risk-significance of the HA portion of the modification.

The NRC may likewise reject a Region I modification that has only HAs. If the change is not rejected, the reviewer proceeds directly to the Level I HFE review. Since the HA itself is very risk-significant, there is no need to perform Steps 2 and 3 of the screening, the NRC will review the HA using the guidance for Level I HFE review in Chapter 3 of this document.

Modification in Region II for Step 1

If the overall modification is in Region II, it is possible that the HA may still undergo either a Level I or Level III HFE review. Therefore, Steps 2 and 3 of the screening process must be performed.

Modification in Region III for Step 1

In the case where a Region III modification includes a combination of both equipment and HAs, there may be equipment improvements that cause a decrease in CDF and may mask the risk-significant contribution of the HAs. That is, although the overall modification is not risk-significant, the HA may be when considered by itself. Therefore, the reviewer must undertake Steps 2 and 3 of the screening.

If a Region III modification includes only HAs, Steps 2 and 3 of the screening method are still performed because the Step 1 risk calculation is based on the *base-case value* of human error probability (HEP) for the HA. Further, there may be a situation where a licensee is replacing a demonstrated reliable automatic component with a presumed reliable HA (with a low HEP). Step 2 will evaluate this and other possibilities.

2.3.3 Human Factors Screening Approach for Temporary Changes (Step 1)

Licensee requests for temporary changes often involve HAs that provide compensatory measures to offset the calculated increase in risk. For example, the request may propose substituting HAs for automatic equipment that is temporarily inoperable and cannot be restored within the time interval required by the plant's technical specifications.

For temporary changes, the calculated risk increase (or decrease) must consider the time that the modification will be in place. RG 1.174 generally applies to all changes (both permanent and temporary) to the LB. However, the acceptance guidelines in RG 1.174 that result in assignments to Region I, II, or III do not explicitly address the duration of a temporary change. If the time for a temporary change is 1 year or longer, it should be screened as a permanent change.

To support the screening process for temporary change requests, the guidance in RG 1.174 is supplemented to make assignments into Regions I, II, or III. The following guidance describes a method the PRA analyst can use to quantitatively evaluate, in an integrated fashion, both the increase in risk and the length of time of increased risk. For this application, the regions are named: Region I^{intgr}, II^{intgr}, III^{intgr} (shown in Figures 2.3 and 2.4).

The risk calculated by a PRA can be expressed in several ways: as an instantaneous value (often calculated for configuration risk management purposes), an average value of CDF over a reactor year (the value most commonly cited), or a cumulative core damage frequency (CCDF) computed over a defined time. The CCDF can be calculated accurately using statistical techniques. A simplified method of viewing the cumulative or integrated risk is to multiply the CDF by time. This gives reasonable results for the type of screening review the NRC performs for HAs that are risk-important. Thus, equations for integrated risk can be written as follows:

$$\text{Integrated CDF Risk (mod)} = \Delta CDF_{mod} \times \text{time (mod)} = \text{ICCDP}$$

or

$$\text{Integrated LERF Risk (mod)} = \Delta LERF_{mod} \times \text{time (mod)} = \text{ICLERP}$$

where:

Integrated Risk (mod) is the integrated risk attributable to the modification over the time that the change or modification is to be in place, expressed as CDF or LERF;

time (mod) is the length of time that the change or modification is to be in place;

ICCDP is incremental conditional core damage probability; and

ICLERP is incremental conditional large early release probability.

The value of Integrated ΔCDF Risk (mod) can be roughly interpreted as the change in the plant's expected core damage probability over the time the modification will be in place. RG 1.177 also uses this concept of integrated risk, where the Integrated ΔCDF Risk is the incremental conditional core damage probability (ICCDP), and the Integrated ΔLERF Risk (mod) is the incremental conditional large early release probability (ICLERP). In this situation, "incremental" refers to the incremental increase in risk over the time period for the modification.

To support the screening process used in determining the level of HFE reviews, acceptance criteria similar to those in RG 1.174 were developed. They were adapted from Section 2.4 of RG 1.177, which addresses the acceptability of integrated risk over periods when equipment is out-of-service for the allowed outage time (AOT). In Section 2.4 of RG 1.177, an acceptability limit of 5E-7 per reactor-year for ICCDP is considered a small risk increase for a single Technical Specification AOT. This 5E-7 value is chosen as the boundary between Regions II^{intgr} and III^{intgr} for ICCDP. The selected boundary for Regions I^{intgr} and II^{intgr} is 5E-6 events per reactor-year, an increase of one order of magnitude. These boundary values result in Figure 2.3, which shows the acceptability limits for ICCDP in terms of Regions I^{intgr}, II^{intgr}, and III^{intgr}. Similarly, RG 1.177 uses 5E-8 events per reactor-year for the limit on a small LERF increase. This value was adopted for the boundary between Regions II^{intgr} and III^{intgr} for ICLERP, while 5E-7 is used for the Region I^{intgr}/II^{intgr} boundary. Figure 2.4 shows the acceptability limits of ICLERP resulting from these choices.

Similar to the approach in RG 1.174, the PRA analyst uses both Figures 2.3 and 2.4 to determine acceptability if the modification affects LERF. If LERF is not affected by the change, Figure 2.3 alone will suffice.

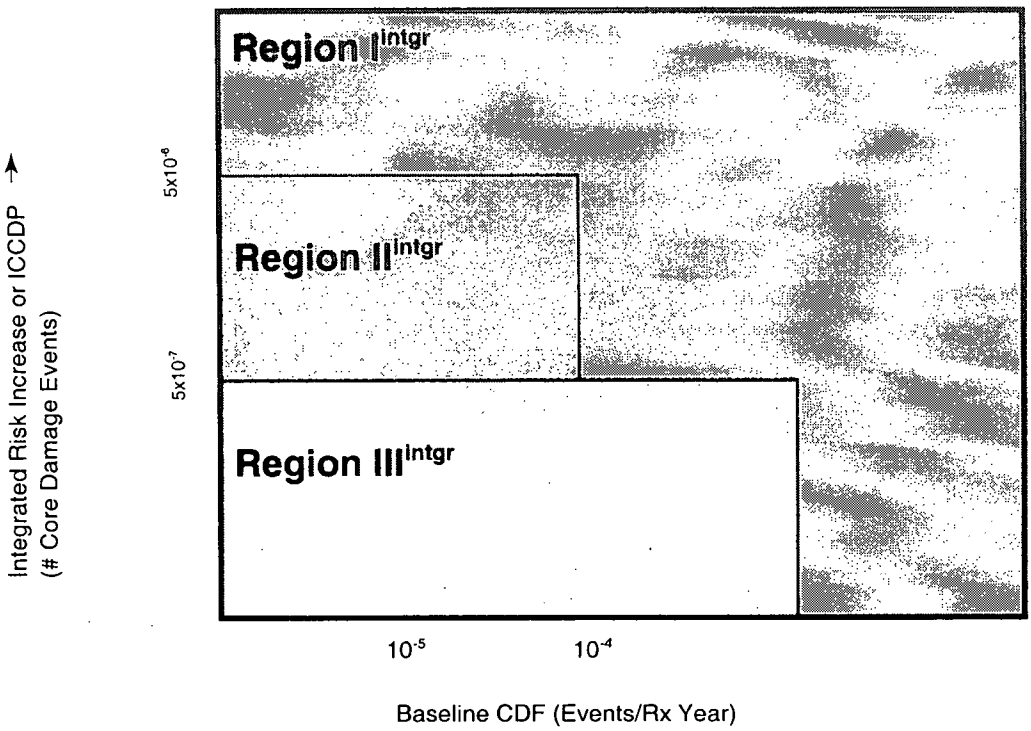


Figure 2.3 Guidelines for Integrated Risk Increase (ICCDP) (Product of ΔCDF_{mod} and Time)

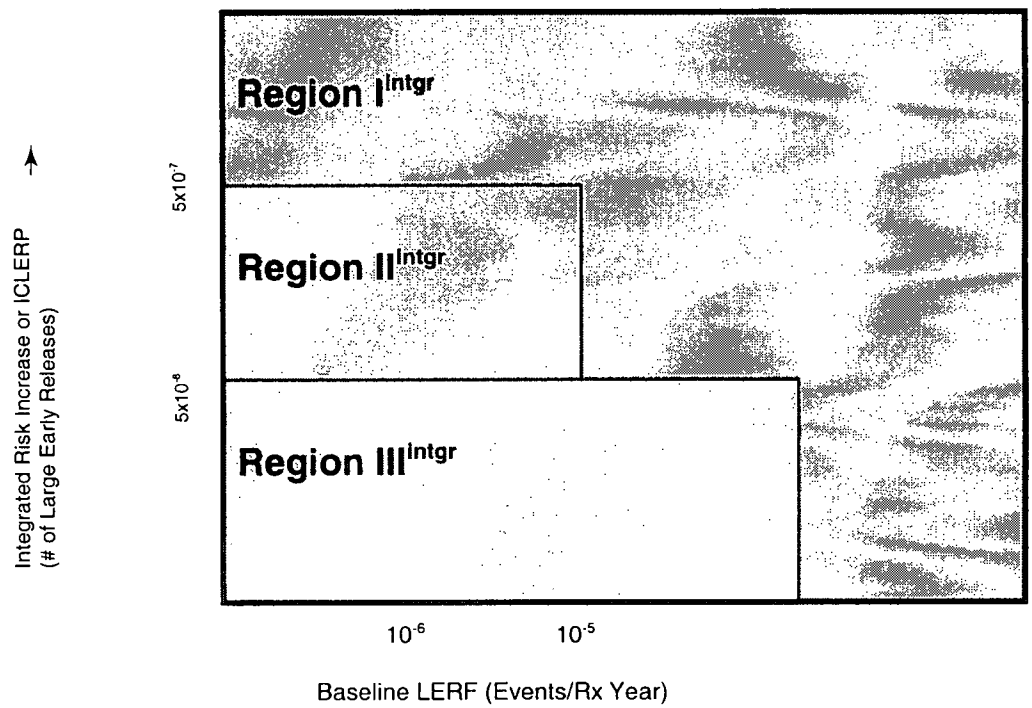


Figure 2.4 Guidelines for Integrated Risk Increase (ICLERP) (Product of $\Delta LERF_{mod}$ and Time)

Potential concerns about using integrated risk measures to address requests for temporary changes include:

- Strict interpretation of this approach would allow acceptance of potentially large increases in risk if the modification is in place for a short period of time.
- For larger values of integrated risk (e.g., in Region I^{intgr}), consideration should be given to the potential synergistic effect on the risk of CDF spikes attributable to changes in plant configuration, together with the effects of the temporary modification. For example, there may be a need to impose temporary restrictions on configurations and equipment out-of-service during the time period of the temporary modification.
- If a licensee makes multiple requests for changes (whether in one or multiple submittals), the cumulative effect of these requests should be considered, including possible dependencies and changes to the operating environment.

The result of this step for temporary changes is similar to that for permanent changes: a Region I^{intgr}, II^{intgr}, or III^{intgr} assignment is an input made by the PRA analyst to the integrated assessment discussed in Section 2.3.6. As with permanent changes, the modification is placed into one of the three risk regions. If the modification is in Region I^{intgr}, the NRC will likely disapprove it. If it is in Region II^{intgr} or III^{intgr}, the reviewer proceeds to Step 2 of the screening process. This part of the process is similar to that for permanent changes, and is summarized in the preceding Table 2.1.

2.3.4 Calculation of Importance Measures for Human Actions (Step 2)

In Step 2 of the screening process for RI requests, importance measures are calculated to assess the importance of HAs involved in the LB change request. The PRA analyst performs this step. The results provide inputs to the integrated decision-making in Step 4. RG 1.174 (especially Appendix A thereto) and SRP Chapter 19 provide guidance on the use of importance measures. While this guidance is primarily written for structures, systems, and components (SSCs), the same principles apply to, and should be used for, calculations involving HAs.[1]

Identification of Human Actions

Licensing submittals often clearly identify HAs that are changed and need NRC review and approval. In other cases, they are part of a larger modification submittal and may not be clearly specified. Then reviewer should ensure that the screening evaluation considers all of the HAs that have been changed in connection with the modification or submittal. Technical reviewers with operations and PRA expertise should review the submittal to ensure that all affected HAs have been clearly identified so that they can be screened. This section discusses some potential problem areas.

---

[1] SRP Chapter 19 uses standard PRA terminology. Consequently, it defines the basic event used to represent human actions as a "human failure event" representing failures of functions, systems, or equipment that are caused by human actions (or lack of action). As such, human failure events may represent more than one action. However, this report uses the terms "human action," "HA," or "operator action."

There may be HAs that involve both equipment failures and HAs (e.g., failure to recover offsite power after a LOOP event) that are not typically called HAs in PRAs. There may also be many HAs affected by a large modification (e.g., a power uprate application) that generally reduce the time available for operators to perform actions after a reactor trip. These HAs should be screened.

There may be cases where an HA is in the emergency operating procedures (EOPs) for post-accident performance, but the HEP may be high and the licensee might choose to set the HEP in the PRA to 1.0 for screening purposes. Thus, the PRA takes no credit for the action. In these cases, the action should not automatically be assigned to Level III or receive no review. The action may still be important and may have a high risk-importance measure in the PRA. Such an action should be screened using the guidance in this report.

There may be cases where, following a modification, the circumstances for HA performance can be expected to improve. This might occur, for example, if the time available for an action has increased, a new alarm signals the need for the HA, or a new procedure is developed to guide HA performance. These types of changes should lower the HA's HEP. However, even with such improvements and the lower HEP, the HEP may still be risk-important (as defined by risk-importance measures). Step 3 provides a methodology to identify these types of circumstances and reduce the level of review.

Multiple Human Actions

A particular plant modification may encompass one or more HAs, some of which may be dependent. The determination of dependency is important and can affect the screening level. When there are many HAs affected by a change in the plant (e.g., more than five), the likelihood of dependency increases. The PRA modeling techniques for HAs vary, so that a given activity may be modeled as one HA or several. Any dependent HAs should be aggregated together. That is, when computing the RAW and FV, include all HAs or all aspects of the one HA. This will give the full importance of the HAs associated with the particular modification. Any HAs that are not dependent can be screened separately.

The RAW and FV Importance Measures

Step 2 addresses how to evaluate the importance of the HA, using two different, but complementary, risk-importance measures: the Risk Achievement Worth (RAW), and the Fussell-Vesely (FV) importance measure. Both of these risk-importance measures are first evaluated relative to the plant's new baseline CDF, assuming the proposed modification is in place. Next, if necessary, they are evaluated relative to the plant's new baseline LERF with the proposed modification assumed to be in place.

The "new baseline CDF" is a shortened term for the "new CDF (with modification in-place)" used in the previous section when defining the ΔCDF_{mod}. Similarly, the new baseline LERF is a shortened term for the "new LERF (with modification in-place)." The RAW measures importance by computing the increase in CDF when the HA fails. That is, the HEP of the HA is increased from its base case value to1.0 and the overall CDF is re-computed. Then, to compute the RAW, a ratio or difference of the new higher CDF to the baseline CDF is taken. The more common ratio method of expressing RAW is used here. The RAW importance

measure was defined and discussed in NUREG/CR-3385 (Vesely, et al., 1983) and (Lambert, 1975). One equation for the ratio value of the RAW for HA "x" is as follows:

$$RAW\ (x) = (CDF \text{ with } x \text{ set to } 1.0) / CDF_{new\ BL}$$

A high RAW value means that failure of the HA results in a risk-significant situation. Thus, the HA's reliability should be verified by a thorough HFE review.

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:

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

If FV is high, the HA contributes to a relatively large portion of risk. Thus, for defense-in-depth purposes, the HA's reliability should be ensured by a thorough HFE review.

Figures 2.5, 2.6, 2.7, and 2.8, respectively, provide guidance on the level of HFE review based on the calculated RAW and FV for the new baseline CDF and LERF. The curves delineating the boundaries between the different levels of review are related to the definitions for Regions I, II, and II given in RG 1.174, as discussed below [and in more detail in Higgins, et al. (2002) and the response to public comments on NUREG-1764].

CDF Importance Measure Evaluation of HA

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 attributable to the HA and the corresponding amount of HFE review. The Levels are distinct and different from the RG 1.174 Regions or PRA levels (i.e., Level I, II, III PRA). 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 HFE review is assigned by Step 2. If an HA falls very close to the dividing line between two levels, the reviewer may want to use the qualitative criteria in Step 3 to finally determine the level of review for the HA (in Steps 3 and 4 of the screening process).

The curve delineating the split between Level I and Level II in Figure 2.5 is roughly based on a CDF of 1E-4 core damage events per reactor-year. Performance deficiencies associated with actions in Level I would generally be colored Red in the NRC's Reactor Oversight Program. Similarly, the curve delineating the split between Level II and Level III in Figure 2.5 are roughly based on a CDF of 1E-5 core damage events per reactor-year. Performance deficiencies associated with actions in Level II would generally be colored Yellow, while those in Level III would be colored White or Green. Figure 2.6 and the FV importance measure were added to provide 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., as a result of a relatively high base-case HEP), but contribute notably to the base-case CDF.

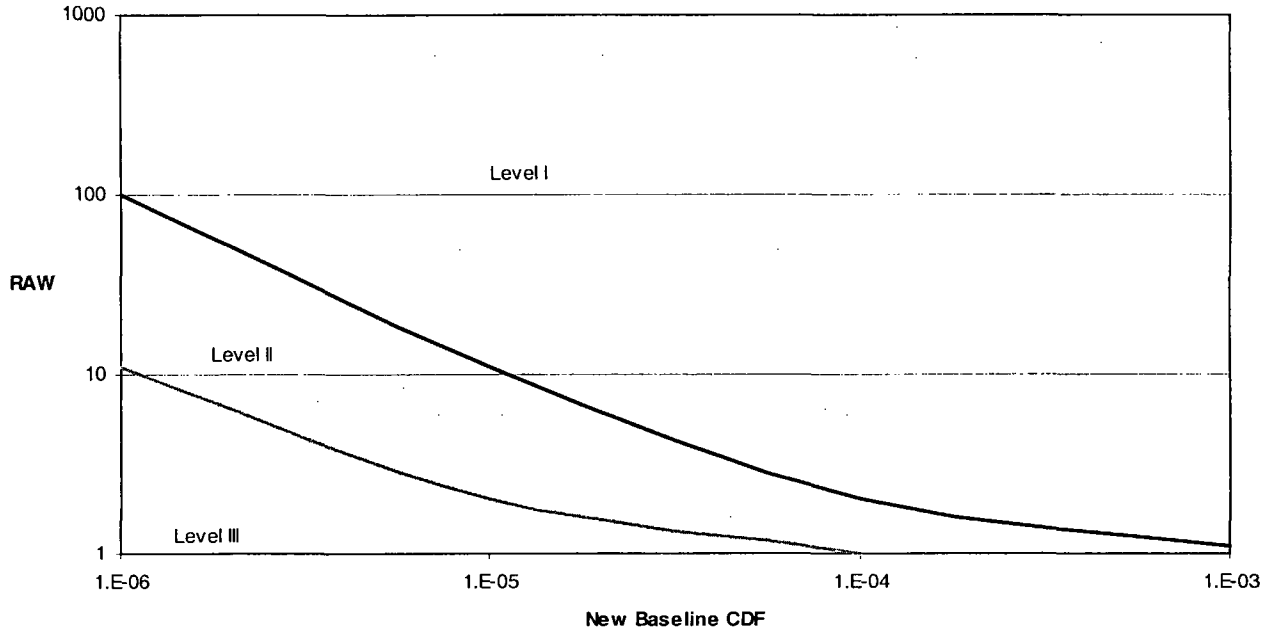


Figure 2.5 RAW vs. New Baseline CDF

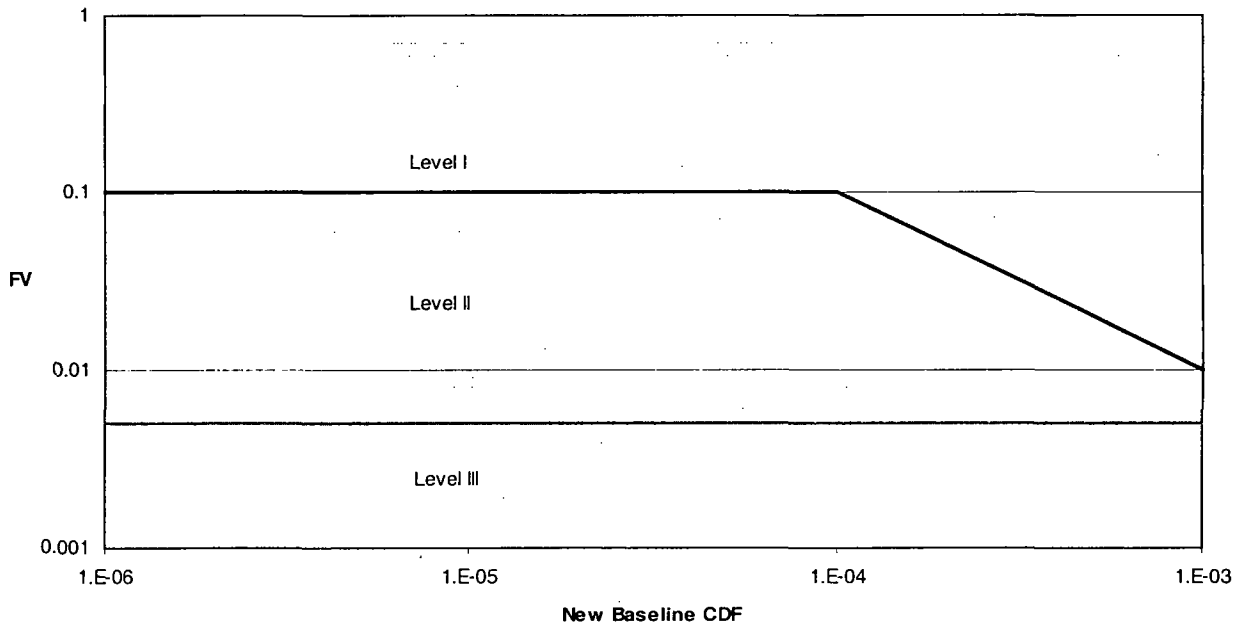


Figure 2.6 FV vs. New Baseline CDF

LERF Importance Measure Evaluation of HA

Next, the PRA analyst determines if the HA requires a separate LERF importance measure evaluation. In general, the default would be that it is not necessary to separately perform a LERF evaluation in this step for the following reasons:

- Most HAs affect primarily CDF, and the LERF evaluation would not yield a different risk-review level.
- LERF importance measures are not routinely calculated, while the CDF importance measures are.
- Some PSA Level II models are not structured to support such calculations of LERF importance measures.

If the PRA analyst judges that an HA does affect the LERF calculations differently than the CDF calculations and a LERF evaluation should be done, the following approach can be used.

The method will use a LERF RAW importance measure, designated as RAW (L) and a LERF FV importance measure designated as FV (L). These measures are analogous to the RAW and FV for the CDF calculations used above and are being applied by the industry for other regulatory purposes. They are defined as follows:

$$\text{RAW (L) (x)} = \text{(LERF with x set to 1.0) / LERF}_{\text{new BL}}$$

$$\text{FV (L) (x)} = \sum \text{all LERF cutsets (or sequences) containing x / LERF}_{\text{new BL}}$$

Figures 2.7 and 2.8 show the levels of review for RAW (L) and FV (L). They were adapted from Figures 2.5 and 2.6 by adjusting the values of the baseline LERF on the x-axis by 1 order of magnitude to account for the fact that LERF values and acceptance criteria generally are 1 order of magnitude less than CDF values and acceptance criteria.

The reviewer computes the RAW (L) and FV (L) values for the HA. Together with the new baseline LERF, this will determine the appropriate level of HFE review. If an HA falls very close to the dividing line between levels of review, the reviewer may want to use the qualitative criteria in Step 3 to finally determine the level of review for the HA (in Steps 3 and 4 of this screening process).

After the regions for both RAW and FV are determined from Figures 2.5 and 2.6, (plus 2.7 and 2.8, if used), the HAs should be placed in the most conservative or highest risk region of the two figures (or four figures, if all were used). If the licensee undertook the calculation and placement in the figures, the results should be submitted to the NRC for verification and use.

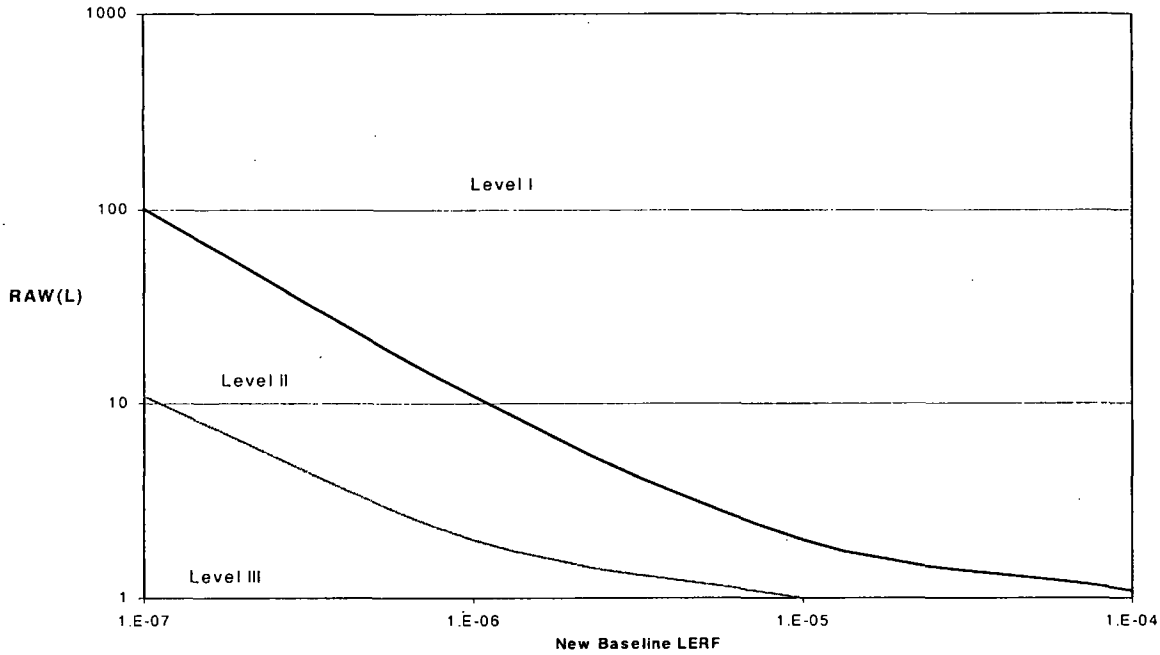


Figure 2.7 RAW (L) vs. New Baseline LERF

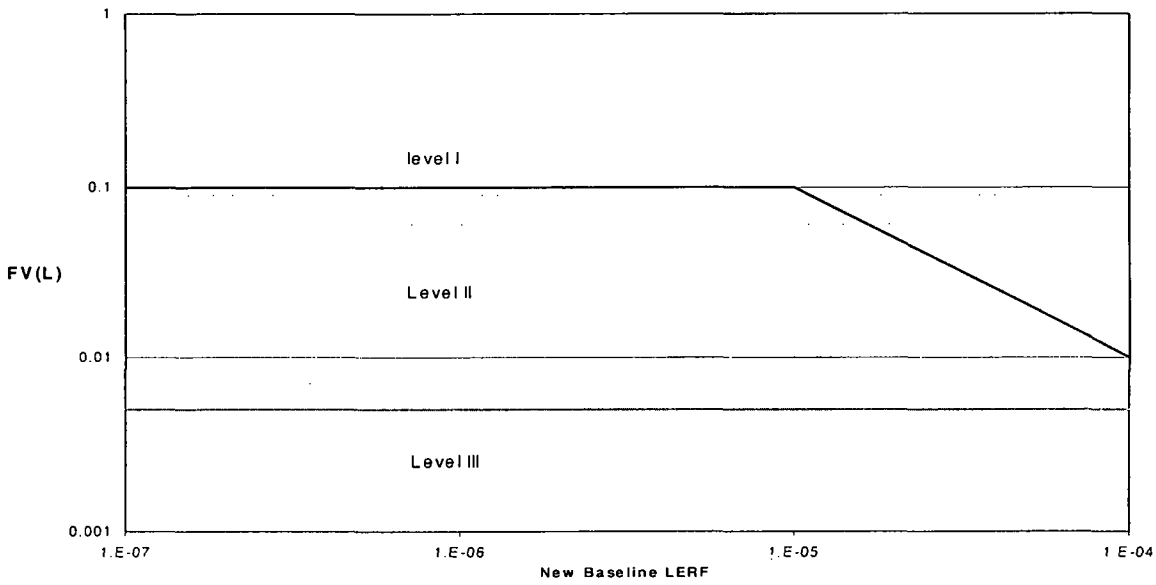


Figure 2.8 FV (L) vs. New Baseline LERF

Consideration of Quality of PRA and Uncertainties

When evaluating the importance measures associated with the HA, careful attention should be paid to the quality of the PRA and the uncertainties that can affect the allocation. Its quality should be adequate to support the assessment of the risk-significance of the HA. Regulatory Guide 1.200 (NRC 2007) provides guidance in the area of PRA quality. Also, the PRA analyst should make a judgment about the uncertainty of the HEP. If the HEP is too high or too low as a result of uncertainty or poor modeling, this will affect both the RAW and FV measures, but in opposite directions. Thus, Step 2 tends to be robust in preventing uncertainty in the HEP from affecting the assignment of the level of review. As discussed in Appendix A to RG 1.174, uncertainties in parameter values can impact the assessment of the risk-significance of a basic event when using importance measures. The assessment of the risk-significance of an HA may be tested against the uncertainties in the assessment of the corresponding HEPs by performing appropriate sensitivity studies, varying the HEP through its range of uncertainty, as, for example, characterized by the 90% confidence interval. The final assessment should be conservative.

Further, if there are judged to be dependent HAs that were not properly modeled in the human reliability analysis (HRA) *and* if the reviewer was unable to adequately address them (as discussed above), increasing the level of HFE review of the set of dependent HAs should be considered.

There may also be cases when a lessening of defense-in-depth or safety margin is associated only with an HA. Then, an increase to the level of HFE review would be appropriate. (Those cases associated with equipment should have been addressed already by reviews made with SRP Chapter 19.)

2.3.5 Qualitative Assessment of Human Action Safety-Significance (Step 3)

In Step 3 of the screening process for RI requests, a qualitative assessment is made of the HAs associated with the change request. This step will likely involve input from both PRA and HFE analysts. The purpose of this qualitative assessment is to assess factors that cannot or may not have been addressed quantitatively in Steps 1 and 2. The results of Step 3 may be recommendations for adjusting the level of HFE review determined previously. These recommendations will be inputs to the integrated decision-making process (Step 4) for determining the appropriate level of HFE review described in Section 2.3.6.

The results produced in the qualitative assessment vary, depending on the specific factors involved and the analyst's assessment. In all cases, the results consist of recommendations for modifying the assignment of level of review based upon the risk calculations of Steps 1 and 2, as follows:

No change	No recommendations for changing the level of HFE review from Steps 1 and 2 result from the qualitative assessments.
Elevate One Level	The results from the qualitative assessment indicate that the initially determined level of HFE review in Steps 1 and 2 should be adjusted to a higher level.

Reduce One Level The results from the qualitative assessment indicate that the initially determined level of HFE review in Steps 1 and 2 should be adjusted to a lower level.

In performing the qualitative assessment in Step 3, reviewers should ensure that they fully understand the HA, including its purpose for various system operational modes and its function for various accident scenarios or sequences. This information may be provided in the licensee's submittal or obtained from documents such as the plant SAR and PRA. If necessary, reviewers may also obtain added insights on the HA by applying Method 1 or 2 of the non-RI process described in Section 2.4.

Section 2.3.5.1 identifies the qualitative factors to consider when making the assessment. Section 2.3.5.2 discusses the way those factors are used to adjust the level of HFE review.

2.3.5.1 Factors Used in the Qualitative Assessments

Three types of qualitative assessment are used:

(1) personnel functions and tasks
(2) design support for task performance
(3) performance shaping factors

The reviewer should determine whether the factors associated with each type of assessment are applicable to the particular HA.

Personnel Functions and Tasks

This type of qualitative assessment examines the potential effects of the request for change on operator tasks and the functions that they perform, under five major categories:

- **Operating Experience**: Does the requested change adversely affect the performance of an action that was previously identified as problematic based on experience/events at that plant or plants of similar design?
- **New Actions**: Does the requested change introduce new HAs? Are the new HAs associated with new responsibilities for the success of safety functions (or additional actions associated with existing responsibilities)?
- **Change in Automation**: Has the requested change given personnel a new functional responsibility that they previously did not have and which differs from their normal responsibilities? For example, are operators now required to take an action in place of a previously automated one? Consider the example of simply being required to open a valve that previously was automatically operated, and where the action required to do so is similar to other valve-opening operations with which operators are familiar. This would not be a sufficient change (in and of itself) to warrant a "yes" to this question when considering task complexity. However, there may be increased workload if the aggregate of added actions is judged to be excessive, this may warrant a "yes."

- **Change in Tasks**: Has the requested change significantly modified the way in which personnel perform their tasks (e.g., making them more complex, significantly reducing the time available to perform the action, increasing the operator workload, changing the operator role from primarily "verifier" to primarily "actor")? In this case, operators do not have a new functional responsibility; instead, the way that they perform their current functional responsibilities has significantly changed and is different from what they usually do.
- **Change in Performance Context**: Has the requested changed created, in some way, a new context[2] for task performance? Or, does the change identify a previously unrecognized context? Or, does the request address a context previously not modeled or considered? If so, what are the important differences in context (e.g., different plant mode, plant behavior, timing of plant symptoms)?

Design Support for Task Performance

This type of qualitative assessment addresses how well the performance of the HAs is supported (e.g., with job aids):

- **Change in Human-System Interfaces (HSIs)**: Has the requested change significantly changed the HSIs used by personnel to perform the task? For example, are personnel now performing their tasks at a computer terminal where previously they were performed at a control board with analog displays and controls?
- **Change in Procedures**: Has the requested change significantly changed the procedures that personnel use to perform the task, or is the task not supported by procedures?
- **Change in Training**: Has the requested change significantly modified the training, or is the task not addressed in training?

Performance Shaping Factors

This type of qualitative assessment addresses four performance shaping factors:

- **Changes in Teamwork**: Has the requested change significantly changed the team aspects of performing an action. For example, (1) is one operator now performing the tasks accomplished by two or more operators in the past? (2) is it now more difficult to coordinate the actions of individual crew members? or, (3) is task performance more difficult to supervise after the modification?
- **Changes in Skill Level of Individuals Performing the Action**: Has the requested change kept the same HA but made it necessary for an individual who is less trained and has lower qualifications to take the action than was the case before the modification?

---

[2] Here, context is defined as the overall performance environment, including plant conditions and behavior that, for example, affect the time available for the operator response and the effectiveness of job aids under these conditions that lead to the assessment of performance shaping factors.

- **Change in Communication Demands**: Has the requested change significantly increased the level of communication needed to perform the task? For example, must an operator now communicate with other personnel to perform actions that previously could be taken at a local panel containing all necessary HSIs?
- **Change in Environmental Conditions**: Has the requested change significantly increased the environmental challenges (such as radiation, or noise) that could negatively affect task performance?

2.3.5.2 Adjusting the Level of Human Factors Engineering Review

Reduce the Level of HFE Review

After considering the factors identified in Section 2.3.5.1, the analyst should consider reducing the level of HFE review if the HA has the following characteristics:

- The answers are "no" to most of the questions posed by the analysis of the qualitative factors. This indicates that there is no change or very little change to the HA from the modification. One "yes" answer should not necessarily preclude a reduction in the level of the review, unless it is a "yes" to a significant question.
- The action is well-defined and, based on the information presented in the submittal, the analyst is confident that it can be easily performed (e.g., it is clear when to perform the action, there are clear procedures, there is sufficient time and staff available, and the action is similar to those routinely taken).

If all the answers are "no" and there is truly no change to the HA, the level of review can be reduced one level, regardless of other considerations. In certain cases, the licensee's ability to correctly perform an HA may be improved, say due to more time available. If this is the only change in the HA, the level of review can also be reduced one level, regardless of other considerations.

The level of review should be reduced by only one level (e.g., Level II to Level III). Additionally, when the review is reduced to Level III, the following criteria from SRP Chapter 19 (Appendix C.2) should be used to verify that the SSCs or human (operator) actions are of low safety-significance:

- The human (operator) action does not relate to the performance of a safety function or a support function to a safety function, or does not complement a safety function. The human (operator) action does not support (e.g., involve dependencies with) other operator actions that are credited in PRAs for either procedural or recovery actions.
- The failure of the human (operator) action will not result in the eventual occurrence of a PRA initiating event.
- The human (operator) action is not required in maintaining barriers to the release of fission products during severe accidents.
- The failure of the human (operator) action will not unintentionally release radioactive material, even in the absence of severe accident conditions.

If any of the above criteria are not satisfied, then re-elevation to a Level II HFE review is recommended.

Elevate the Level of HFE Review

The level of HFE review *may* be increased if "yes" was the answer to any of the questions posed by the qualitative factors. The analyst should consider the potential effect of the "yes" factors. If such responses are obtained for many factors, the level of review of the HA should probably be increased. If a "yes" response is received for only one or two factors, the analyst should consider whether the information leading to the "yes" response is sufficient to warrant elevating the level of review.

No Adjustment to the Level of HFE Review

If the consideration of qualitative factors (above) did not decrease or increase the level of review, no change should be recommended on the basis of Step 3.

Result of Step 3

The final result of Step 3, Qualitative Assessment of HA Safety Significance, is one of three decisions that are used in Table 2.2 of Step 4: No Change, Elevate One Level, and Reduce One Level.

2.3.6 Integrated Assessment of Human Action Safety-Significance for Risk-Informed Requests (Step 4)

This section provides guidance on how to integrate the results from Steps 1 through 3 of the screening process for risk-informed, licensing basis, change requests. This step is performed by both PRA and HFE analysts. This guidance is intended to be conservative while still providing an opportunity to reduce the level of resources spent on HFE review on the basis of risk and risk insights. This guidance is not intended to be prescriptive.

Table 2.2 summarizes the guidance for determining the appropriate level of HFE review. This table shows the various possible results from Steps 1, 2, and 3, along with recommendations for the appropriate level of HFE review. The results from Step 1 are the acceptability regions from RG 1.174. The results from Step 2 are the initial determination of the level of HFE review from calculating importance measures. Results from Step 3, as discussed in Section 2.3.5, are either "No Change," "Elevate One Level," or "Reduce One Level." The final column of Table 2.2 gives the results from this Step 4 evaluation. The recommendations for HFE review are given as Level I, II, or III.

Table 2.2 Integrated Assessment with RI Screening

Results of Step 1 *RG 1.174 (see Note 1)*	**Results of Step 2** *Importance Measures*	**Results of Step 3** *Qualitative Assessment*	**Results of Step 4** *Recommended Level of HFE Review*
Region I (HA only)	—	—	Level I
Region I (Equipment & HA)	Level I	No Change or Elevate	Level I
		Reduce	Level II
	Level II	Elevate	Level I
		No change	Level II
		Reduce	Level III
	Level III	Elevate	Level II (see Note 2)
		No Change or Reduce	Level III
Region II	Level I	No Change or Elevate	LevelI
		Reduce	Level II
	Level II	Elevate	Level I
		No change	Level II
		Reduce	Level III
	Level III	Elevate	Level II
		No Change or Reduce	Level III
Region III	Level I	No Change or Elevate	Level I (see Note 3)
		Reduce	Level II
	Level II	Elevate	Level I (see Note 3)
		No Change	Level II
		Reduce	Level III
	Level III	Elevate	Level II (see Note 4)
		No Change or Reduce	Level III

Notes:

1. If the modification being reviewed is a temporary one, then the entries of the first Column of this table will be Region I^{intgr}, II^{intgr}, & III^{intgr}.
2. This is a Region I modification where the HA was determined not to be risk significant. However, qualitative factors suggested there was sufficient cause to recommend elevating the level of HFE review. In such a situation, the analyst may determine that a further increase to a Level I review is justified, for example, where the qualitative analysis resulted in many "yes" responses or where the action is associated with a critical safety function.
3. This is a Region III modification where the HA was found to be risk significant. An example of how this may occur is in a boiling-water reactor (BWR) with a modification to the RHR Suppression Pool Cooling to improve its reliability. This would then decrease risk, placing the overall modification in Region III. But in most BWRs, the operator action of SPC, has a very high RAW value (giving it a Level I from Step 2). Such a modification may or may not involve changes to the operator action. The final level of review will be based heavily on judgement, using the qualitative factors. As an example, Step 3 may have led to a recommendation to keep the review Level I or to increase it from Level II to Level I. The analyst then may determine that a Level I review is not warranted and prefer instead to conduct a Level II review. This may occur where the HA is simple, well defined, not time-pressured, yet well supported by training, procedures, and the HSI design.
4. Caution should be used in elevating a Region III/Level III HA to avoid excess effort on an HA that has little risk significance.

2.4 Screening Process for Non-Risk-Informed Change Requests

If the licensee's change request is not risk-informed, the NRC will still want to screen the human actions in order to determine the proper level of HFE review. This can be done with the methods presented in this section or possibly with the methods in Section 2.3 if sufficient current risk information is available. That is, licensees sometimes make risk information available even though they do not request a risk-informed review.

If the licensee's risk information is not available, a different process than that described in Section 2.3 is used. The safety-significance of the HAs involved in the change request still is determined, but without benefit of the risk inputs provided by the licensee. In this case, the NRC staff performs a scoping type risk evaluation to estimate the risk-importance of the change to the HA.

2.4.1 Overall Screening Approach for Non-Risk-Informed Requests

The overall screening approach for non-RI requests parallels that for RI requests, with two modifications. First, the reviewer verifies whether a non-RI request is appropriate, following the general guidance of SRP Chapter 19. Second, the reviewer determines the safety-significance of the HAs using general risk and human reliability concepts, rather plant-specific risk information. The non-RI screening process consists of the following steps:

(1) Verify that a non-RI change request is appropriate.

(2) Assess safety-significance of the HAs by either of two methods (Section 2.4.2 and 2.4.3).

(3) Qualitatively assess the safety-significance of HAs involved in the change request (Section 2.4.4).

(4) Make an integrated assessment of HA safety-significance to determine the appropriate level of HF review (i.e., Level I, 2, or 3) (Section 2.4.5).

2.4.2 Assessment of Appropriateness of Non-Risk-Informed Submittals with Human Actions

Appendix D to SRP Chapter 19 addresses the use of risk information in reviewing requests non-RI license amendments. This assessment is performed by a risk analyst. In particular, the guidance in SRP Chapter 19 identifies the following considerations:

(1) when the risk implications of a non-RI submittal would be discussed with a risk analyst

(2) examples of the potential impacts of "special circumstances" for which NRC staff can request risk information from the licensee, or reject the application

Because non-RI licensee change requests may contain only HAs, this section only briefly discusses the guidance that SRP Chapter 19 provides on these two topics. This discussion is not intended to replace the guidance given in SRP Chapter 19. Rather, it is intended to make reviewers aware of that guidance.

If the licensee decides to provide risk information to support the change request, as a result of the guidance immediately below, the NRC uses the screening process for RI applications. If not, the NRC employs the following screening process for non-RI applications.

2.4.2.1 Risk Implications of a Non-Risk-Informed Submittal

This section replicates the guidance in SRP Chapter 19, regarding when to discuss a submittal with a risk analyst. As stated in Appendix D to SRP Chapter 19, "...the risk implications of a non-risk-informed submittal would be discussed with a risk analyst if the submittal —

- significantly changes the allowed outage time (e.g., outside the range previously approved at similar plants), the probability of the initiating event, the probability of successful mitigative action, the functional recovery time, or the operator action requirement;[3]
- significantly changes functional requirements or redundancy;
- significantly changes operations that affect the likelihoods of undiscovered failures;
- significantly affects the basis for successful safety function; or
- could create 'special circumstances' under which compliance with existing regulations may not produce the intended or expected level of safety and plant operation may pose an undue risk to public health and safety."

---

[3] The term "requirements" as used here and elsewhere in this document, refers to requirements that are established as part of the design process. The term requirements is used in this context as a term-of-art. These are not "regulatory" requirements. There are no regulatory requirements in this document, only review guidance.

In this guidance, "operator action" is explicitly stated in one of the above criteria. However, HAs can be involved in many other changes listed above (e.g., probability of successful mitigative action, functional recovery time, likelihood of undiscovered failures). If the licensee change request is judged to involve any of the above concerns, then, as stated in SRP Chapter 19, "...[the licensee request] would be referred for a more detailed risk evaluation as part of the license amendment review."

2.4.2.2 Potential Effects of "Special Circumstances"

SRP Chapter 19 provides a description of "special circumstances" and examples of their potential impacts. This section does not repeat the guidance provided in SRP Chapter 19. Reviewers of non-RI licensee change requests that involve HAs should determine if a special circumstance may exist.

Beyond these examples, SRP Chapter 19 states that if the reviewers "...believe that approval of the licensee change request would compromise the safety principles described in Regulatory Guide 1.174 and substantially increase risk relative to the risk acceptance guidelines contained in the regulatory guide, the reviewers should inform NRC management of the risk concerns and the need to further evaluate the risk associated with the request." The full guidance in the SRP should be consulted when reviewers believe that there might be such a concern associated with a non-RI submittal.

2.4.3 Assessment of Human Action Safety-Significance

As noted in Section 2.1, the reviewer must ensure that all appropriate HAs, affected by a modification or change, are properly identified and screened. Considerations for identifying all HAs are provided in Section 2.3.4.

Two different methods are discussed for determining the safety-significance of HAs. The first method, the Estimated Importance Method, requires a risk analyst to estimate the risk-importance of the HA. The second, the Generic HA Method, is available if resources do not include a PRA analyst. This second method can be performed by the HFE analyst. Either of the two methods can be used. Outputs from this step are then used in Section 2.4.5, Integrated Assessment of HA Significance for Non-RI Requests.

2.4.3.1 Estimated Importance Method (Method 1)

In this method, a PRA analyst makes an assessment, using general PRA knowledge together with an understanding of the specific plant design. The result of this assessment is a surrogate for the RAW importance measure for the HA. The following basic steps are used in making this assessment:

- Based on the definition of the HA, identify the impact of its failure on the key safety function(s) it supports:
  - Would this result in a loss of redundancy for a single system supporting a key safety function?
  - Would it result in the loss of a complete system?

- Would it result in total failure of a key safety function?

- Determine the remaining capability to perform the key safety function, given the failure of the HA.
- Identify the accident sequences affected by the HA's failure. For most PRA modeling styles, this identification should be relatively straightforward based on understanding the event trees at the functional or systemic level, and the relationship between systems and key safety functions.

Assess the significance of the failure of the HA based on an understanding of how much margin was eroded by its failure.

Using the steps above, together with generic values for initiating event frequencies, and reasonable estimates for system unavailabilities, a PRA analyst can roughly estimate the RAW risk-importance measure for the HA. In order to perform this analysis, it is useful to have a copy of the plant-specific PRA. If that is not available, other options are to use the plant-specific RI inspection notebook, the summary report of the benchmarking trip to the site, or the NRC's Standardized Plant Analysis of Risk (SPAR) models. If a HA is not modeled in the plant PRA, the RI inspection notebook, or the SPAR model, this would generally imply that the action is not risk-significant and could be preliminarily screened as Level III. This is not true for the benchmarking report, since not all risk important HAs were benchmarked. One exception to be aware of involves HAs associated with post-core-damage activities that may be important for limiting LERF-type releases, but not for preventing core damage.

The steps for using the RI inspection notebook are as follows:

(1) Identify the impact of the failure of the HA on the key safety functions modeled in the notebook.

(2) Determine the remaining capability to perform the safety function.

(3) Identify the notebook sequences on the various worksheets that contain the HA (or a surrogate item if necessary).

(4) Fail HAs in those sequences and determine the changes (or ΔCDF) in sequence frequency.

(5) Add the frequency (ΔCDF) of all affected sequences to determine a total ΔCDF (HA).

(6) Compute the RAW for HA.

For these RAW calculations use the following equation:

$$\text{RAW (HA)} = [\Delta\text{CDF(HA)} + \text{CDF}_{\text{new BL}}] / \text{CDF}_{\text{new BL}}$$

If using the summary report of the benchmarking trip to the plant site, PRA RAW values may be obtained from the tables of the report, since they were obtained for many of the more important HAs in the PRA. When using the SPAR models, RAW values can be calculated if the HA is modeled.

Once such RAW risk-importance estimates have been generated, Figure 2.5, "RAW vs. New Baseline CDF," can be used to determine the initial estimate for the level of HFE review. This level is entered in Column 1 of Table 2.4.

While not required for this method, one may also estimate the FV importance measure from the above sources. For example this could be estimated with RI inspection notebook by summing the frequency of the sequences that contain the HA and dividing by the total CDF in the footnotes of Table 2. The FV measure may also be calculated using the SPAR models.

2.4.3.2 Generic HA Method (Method 2)

When a PRA analyst is not available to undertake the approach described above, the HFE analyst can use a more generic and less quantitative method to evaluate the safety-significance of the HA. The generic method for determining HA safety-significance and subsequent level of HFE review, is based upon general risk information and some plant-specific information, whenever possible. However, the analyst should be cautioned that this generic approach is limited and may not always be conservative. For example, generic HAs that are risk-important are not necessarily representative of plant-specific risk-importances. In addition, the change request may portray different performance conditions than those delineated by generic HA definitions, or even previously submitted plant-specific HA definitions. Finally, any new HAs (i.e., HAs not previously modeled generically or in a plant-specific PRA) may never have been analyzed and may be highly safety-significant, regardless of the significance of the involved system or function in generic or plant-specific PRA results that do not reflect the requested change.

To determine HA safety-significance with the generic approach, the HA involved in the change request is compared with the appropriate list of HAs for boiling water reactors (BWRs) and PWRs, given in Tables A.1 and A.2, respectively. These HAs were initially identified and grouped based on the RI assessment process (Azarm, Higgins, and Chu, 1999), and from NUREG-1560 (NRC, 1997a). The grouping was then updated in 2001/2002 based on risk information from the latest licensee PRAs obtained during the site benchmarking of the RI inspection notebooks. These were specifically developed as slightly conservative evaluation tools.

The HAs in Tables A.1 and A.2 are organized into two groups. Group 1 contains the most risk-important HAs. RAW calculations typically would place them in the Level I area of Figure 2.5. Group 2 HAs are considered to be "potentially" risk-important. That is, they would be screened into the Level I review category for some plants, but not all. Typically, they affect risk, but not as significantly as Group 1 actions. However, at some plants they may be quite risk-important. The last item in Group 2, "Actions involving the risk-important systems," is discussed in more detail below.

Table 2.3 summarizes the simple and conservative logic on how to use the Group1 and Group 2 HA assignments. HAs assigned to Group 1 are placed in the Level I review category. Those assigned to Group 2 go in either the Level I or Level II review category. If no risk submittal is made and the plant modification involves more than a minor change to a Group 2 action, then the NRC reviewer needs to decide whether it merits a Level I or Level II review. The conservative approach is to preliminarily assign it to Level I and then proceed to the Step 3,

"Qualitative Evaluation." If the reviewers have additional information about the risk of the action, either from the licensee or from the NRC's risk staff, they may choose to preliminarily assign the Group 2 HA to Risk Region II, and then proceed to the Step 3, Qualitative Evaluation. The last item in Group 2, "Actions involving the risk-important systems," can be assigned to Level I or II, as discussed below.

On a plant-specific basis, actions *not* specifically listed in Tables A.1 and A.2 *may* also be risk-significant and, thus, could require either a Level I or II review. This is not expected to be common, but could happen. Therefore, if an action is not listed on either table, it cannot be concluded that it is not important to risk.

Thus, if no risk submittal is made and the plant modification involves an action that is not in Group 1 or 2, an additional step is taken to determine whether it involves risk-important systems for the plant. This step can be used for both new and modified HAs. The risk-important systems can be obtained from the plant's individual plant examination (IPE) or latest updated PSA. This information also can be extracted from the plant-specific, risk-informed, inspection notebooks and related benchmarking reports that were completed by the NRC. For example, systems that benchmark as Red items should be considered to have high risk-importance. Those that benchmark as Yellow or White should be considered as of moderate risk-importance, while Green ones would have lower risk-importance.

All plants now have a plant-specific, risk-informed, inspection notebook based on a recent updated version of the plant's PRA and a related benchmarking report that compared the risk-importance (or coloring) of all of the major components and human actions in the notebook to the PRA. These are available to the NRC and to licensees. The risk-importance of systems, components, or human actions can be determined by using the notebook per SECY-99-007A (NRC, 1999) and NRC Inspection Manual Chapter (IMC) 0609 (NRC, 2003). These importances can also be obtained from Table 1, "Summary of Benchmarking Results," contained in each of the plant-specific benchmarking reports. Assistance in using these documents, can be obtained by contacting the NRC's senior reactor analysts (SRAs).

If the action in a non-RI submittal involves a high risk-importance system (per the above paragraphs), and more than minor changes are involved, then the HA is considered most likely in risk Region I or II of RG 1.174. The same logic as discussed for Group II HAs applies, and the reviewer should preliminarily select a Level I or II review and then proceed to Step 3, "Qualitative Evaluation." Similarly, if the HA involves a system of moderate importance, the HA should be considered in Region II of RG 1.174. If the modification involves only systems with lower risk-importance, it is initially considered to require a Level III review.

The logic applied to HAs in the generic method is conservative and may place an HA in a higher risk region than it would receive using plant-specific RAW calculations. Thus, it is beneficial for both licensees and NRC staff to use plant-specific risk information to more properly allocate review resources.

Note that the last two rows of Group 2 in Tables A.1 and A.2 are very general and written so that they will identify any new HAs that did not exist previously and are most likely not in the PRA, but which may be risk important.

The principal motivation for these assignments is to make conservative assessments. Further adjustments can be made in Step 4 (i.e., the integrated assessment) with inputs from the qualitative assessment.

Column 3 of Table 2.3 below provides the initial level assignment based on this subsection 2.4.3.2. This should be entered into the first column of Table 2.4. The review then proceeds to the Qualitative Assessment in Section 2.4.4, below.

Table 2.3 Generic Approach for Placing HAs into HFE review Levels

Generic Groups that contain the HA	Systems involving the HA	Level of HFE review for the HA
Group 1	NA	Level I
Group 2	NA	Level I or II*
Neither Group	High risk-importance (Red)	Level I or II*
Neither Group	Moderate risk-importance (Yellow or White)	Level II
Neither Group	Low risk-importance (Green)	Level III
* See discussion in text of Section 2.4.3.2 for determination of Level I or II here.		

2.4.4 Qualitative Assessment of Human Action Safety-Significance

The qualitative assessment of HAs involved in non-RI change requests is identical to that for RI change requests. Consequently, the guidance in Section 2.3.5 is appropriate and should be used here. Outputs from this step are used in Section 2.4.5 below, "Integrated Assessment for HA Safety-Significance."

2.4.5 Integrated Assessment of Human Action Safety-Significance for Non-Risk-Informed Requests

The integrated assessment of HA safety-significance for non-RI applications is similar to that for RI applications, but simpler because there are fewer inputs to integrate. This integrated assessment is performed by the HFE analyst. Table 2.4 lists the recommendations for the level of HFE review using the non-RI screening process.

The bases for the recommended HFE review levels shown in Table 2.4 are similar to those shown in Table 2.2 for the RI screening process. However, because the HA importance estimates produced in Step 1 of the non-RI screening process are expected to be much less sophisticated than those for the RI screening process, the qualitative assessment results may be given more weight in the non-RI screening process. The goal in these recommendations, as for those in the RI screening process, is to produce reasonable but conservative assessments. In line with this goal, consider the Table entries for a Level I or Level II HA, where the qualitative assessment results recommend "Reduce." The table maintains the option to not reduce the

level of review as part of this Section 2.4.5, Integrated Assessment. This is done to add caution in reducing the level of review for a HA that was evaluated using the generic method.

Table 2.4 Integrated Assessment with Non-RI Screening

Results of Section 2.4.3 HA Safety Significance	**Results of Section 2.4.4** Qualitative Assessment Results	**Results of Section 2.4.5** Integrated Assessment
Level I	No Change OR Elevate	Level I
	Reduce	Level I or II
Level II	Elevate	Level I
	No Change	Level II
	Reduce	Level II or III
Level III	Elevate	Level II*
	No Change OR Reduce	Level III

* This is a modification where the HA was determined not to be risk-significant based on generic analyses only. However, based on qualitative factors, there was sufficient cause to recommend elevating the level of HFE review. In such a situation, the analyst may determine that a further increase to a Level I review is justified, for example, where the qualitative analysis resulted in many "yes" responses and where there is uncertainty about the importance of the action.

2.5 Level of Human Factors Engineering Review for Human Actions

Once the appropriate screening process for either RI or non-RI requests is performed, the level of HFE review to be performed is determined in Section 2.4, above. Three levels of review are provided, with the most important HAs (i.e., Level I HAs) receiving the most thorough review. Less important HAs (Level II) receive a more efficient review that is appropriate to their level of importance. HAs that are assigned to Level III receive no (or minimal) review. The principal focus of a Level III HA review is to verify that these HAs have been properly classified in Level III using the screening process. Briefly, the three levels of HFE reviews are as follows:

Level I This level corresponds with the most extensive and detailed HFE review, as described in Section 3. This level includes review of planning, analyses, design, and verification and validation activities, and a performance monitoring strategy.

Level II This level corresponds with a less detailed review. Section 4 provides guidance for performing a Level II review. In special circumstances, as identified in Step 3, the NRC may choose to add selected Level I review criteria to a Level II review, rather than elevating the HA to a full Level I review.

Level III This level corresponds with a minimal review, which includes the following aspects:

- Licensee documentation and NRC verification of the appropriateness of the Level III assignment for the HA(s) associated with the change request
- NRC verification that current regulations are still being met with the change in place (using Criterion 1 in Section 3.1, "General Deterministic Review Criteria").

In addition, licensees are encouraged to utilize the Level II guidance contained in Section 4 to ensure that the HAs can be accomplished as assumed. Also, in some cases, there may be one or two aspects, identified during the Step 3 qualitative assessment, that the NRC chooses to review. The NRC would use the appropriate portions of the Level II review guidance for these identified aspects.

3. LEVEL I REVIEW GUIDANCE

The guidance in this section is a tailoring of NUREG-0711 (NRC, 2004b) to plant modifications affecting HAs of high risk-importance. NUREG-0711, Section 1.4, "Graded Approach to Review," indicates that the level of staff review of an applicant's HFE design should reflect the unique circumstances of the review, and the guidance should be selectively applied to address the demands of each specific review.

The tailoring was accomplished by selecting the NUREG-0711 criteria that were appropriate to review of Level I HAs that are risk-important, and then modifying these criteria to better reflect the context of the types of plant modifications involved. This guidance has been developed to "stand alone." That is, aspects of the review criteria that were not changed are repeated in this section, rather than referring the reviewer back to NUREG-0711. This makes the guidance easier to use.

Even with the tailored guidance provided in this section, the reviewer can further adapt the guidance to meet the unique demands of particular reviews.

3.1 General Deterministic Review Criteria

Objective

The objective of this review is to verify that deterministic aspects of design, as discussed in RG 1.174, have been appropriately considered by the licensee. Deterministic aspects include ensuring that the change meets current regulations and does not compromise defense-in-depth.

Scope

The deterministic review criteria apply to all modifications associated with Level I HAs.

Criteria

(1) The licensee should provide adequate assurance that the change meets current regulations, except where specific exemptions are requested under 10 CFR 50.12 or 10 CFR 2.802. For example, a change might be identified as risk-significant when using a standard PRA to screen for risk. However, the NRC might grant an exemption under one or more of the following regulations: 10 CFR Part 20, Criterion 19 in Appendix A to 10 CFR Part 50, or Appendices C through R to 10 CFR Part 50.

(2) The licensee should provide adequate assurance that the change does not compromise defense-in-depth. Defense-in-depth is one of the fundamental principles upon which the plant was designed and built. Defense-in-depth uses multiple means to accomplish safety functions and to prevent the release of radioactive materials. Defense-in-depth is important in accounting for uncertainties in equipment and human performance, and for ensuring some protection remains even in the face of significant breakdowns in

particular areas. Defense-in-depth may be changed but overall should be maintained. Defense-in-depth includes the following important aspects:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- There is no over-reliance on programmatic activities to compensate for weaknesses in plant design. This may be pertinent to changes in credited operator actions.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved, and the potential for introduction of new common cause failure mechanisms is assessed. Caution should be exercised in crediting new operator actions to provide adequate assurance that the possibility of significant common cause operator error is not created.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved. One way to help ensure this for HAs that are risk-important is to establish procedures for a second check or independent verification that such important actions have been properly executed.
- Safety margins often used in deterministic analyses to account for uncertainty and provide an added margin to provide adequate assurance that the various limits or criteria important to safety are not violated. Such safety margins are typically not related to HAs, but the reviewer should take note to see if there are any that may apply to the particular case under review. It is also possible to add a safety margin (if desired) to the HA by requiring a demonstration that the action can be performed within some time interval (or margin) that is less than the time required by the analysis.

3.2 Operating Experience Review

Objective

The objective of this review is to verify that the licensee has identified and analyzed HFE-related problems and issues encountered previously in designs and human tasks that are similar to the planned modification so that issues that could potentially hinder human performance can be addressed.

Scope

The operating experience review (OER) encompasses all proposed changes to HAs and addresses the operating histories of plant systems, HAs, procedures, and HSI technologies. The scope of the HSI technology review can be graded as follows:

(1) If existing HSI components are to be used without modification and if they are currently used for safety-related functions within the plant, then a review of the operating experience with those HSI components is not necessary.

(2) If existing HSI components are to be used without modification but they are not currently used for safety-related functions then the operating experience with those HSI components should be reviewed.

(3) If new HSI components are to be installed or the existing HSI is to be modified using HSI technologies that have not been previously used in the plant for safety-related functions then the operating experience with those HSI components should be reviewed.

Criteria

(1) *Plant Systems*: The licensee's review should include information pertaining to the operation and maintenance of the plant system prior to the change in the HAs.

(2) *Human Actions*: The licensee's review should identify performance issues associated with procedural guidance, training, and HAs for the system prior to the proposed change to the actions, including the types of actions performed, the procedures available for those actions, and the adequacy of those procedures.

(3) *HSI Technologies*: The licensee's review should identify human performance issues associated with HSI technologies for the proposed changes in the HAs, if they are different from those used successfully at their plant.

(4) *Issues Identified by Plant Personnel*: Interviews and surveys with personnel should be conducted to determine operating experience related to the plant system before the change in the HAs. Discussions of plant operations and HFE/HSI design should be limited to topics relevant to the change in the HAs.

(5) *Use for Design Input*: Issues identified by the OER should be used for input to the design of modifications to the HSI, procedures, and training, and tracked to provide assurance that they are addressed.

3.3 Functional Requirements Analysis and Function Allocation[4]

Objective

The objective of this review is to verify that the licensee has done the following:

(1) defined any changes in the plant's safety functions (functional requirements analysis)

(2) provided evidence that the allocation of functions between humans and automatic systems provides an acceptable role for plant personnel [i.e., the allocations take advantage of human strengths and avoid functions that would be negatively affected by human limitations (functional allocation)]

---

[4] If there are no changes in functional requirements or functional allocation from the current plant design, this review element is not needed.

Scope

This review addresses all plant functions affected by the change in operator actions including changes to the functions and to their allocation between personnel and automatic systems. The level of detail in the functional requirements and allocation analyses may be graded by the reviewer based on: (1) the degree of difference between the HAs before and after the change, (2) the extent to which difficulties occurred in prior operations, as identified through the OER, and (3) the risk level associated with the change. The following additional considerations apply:

(1) If new safety functions are introduced or existing ones changed, then reviews of both the functional requirements analysis and function allocation analysis should be conducted. (This situation is not likely to occur since it would involve a significant deviation from the design basis that was originally approved by the NRC.)

(2) If the function allocation is changed, or if the risk level is well into Level I (as determined by the PRA/HRA review criteria) then a review of the function allocation should be conducted. (Many cases will have changed function allocations. An example may be the reallocation of responsibility from an automatic system to personnel for the initiation, on-going control, or termination of a function.)

(3) If the function allocation is not changed then no function allocation analysis is needed and the licensee should proceed with task analysis. (An example may be a manual action performed for a safety-related function that is now required under a new scenario. That is, the function is the same but the initiating circumstances are different.)

Criteria

Functional Requirements

(1) New or changed safety functions should be described, including comparisons before and after the proposed change. The set of plant system configurations or success paths that are responsible for or capable of carrying out the safety function should be clearly defined and the ones affected by the proposed changes in the HAs should be identified. This functional decomposition should address:

- high-level functions [e.g., maintain reactor coolant system (RCS) integrity] and critical safety functions (e.g., maintain RCS pressure control)
- specific plant systems and components
- technical basis for changes to functions

Functional Allocation

(1) For the functional allocation analysis, a description should be provided for each of the high-level functions allocated to the human as a result of the proposed change. The description should include the following:

- purpose of the high-level function
- conditions under which the high-level function is required
- parameters that indicate that the high-level function is available

- parameters that indicate the high-level function is operating (e.g., flow indication)
- parameters that indicate the high-level function is achieving its purpose (e.g., reactor vessel level returning to normal)
- parameters that indicate that operation of the high-level function can or should be terminated

Note that parameters may be described qualitatively (e.g., high or low), rather than as specific numerical values or setpoints.

(2) The technical basis for all relevant functional allocations should be documented. The basis for function allocations can be successful operating experience. This analysis should reflect (a) sensitivity, precision, time, and safety-related requirements; (b) required reliability; and (c) the number and level of skills of personnel required to operate and maintain the system.

(3) The allocation analysis should consider not only the personnel role of initiating manual actions but also responsibilities concerning automatic functions, including monitoring the status of automatic functions to detect system failures. The demands associated with the proposed allocation of functions should be considered in terms of all other human functions that may impose concurrent demands upon the personnel. The overall level of workload should be considered when allocating functions to the personnel.

3.4 Task Analysis

Objective

The objective of this review is to verify that the licensee's task analysis (TA) identifies the behavioral requirements of the tasks personnel are required to perform. The task analysis should form the basis for specifying the requirements for the HSI, procedures, and training based on the tasks personnel will perform. The results are also used as basic information for developing staffing and communication requirements of the plant. For a change to an existing action, a new TA may not be necessary.

Scope

The task analysis addresses HAs in their entirety, including all pertinent plant conditions, situational factors, and performance shaping factors.

Criteria

(1) The licensee should identify the information that is required to inform personnel that each HA is necessary, that the HA has been correctly performed, and that the HA can be terminated.

(2) Plant personnel who are affected by the HAs should be identified, including licensed control room operators as defined in 10 CFR Part 55 and the following categories of personnel defined by 10 CFR 50.120: non-licensed operators, shift supervisor, shift technical advisor, instrument & control technician, electrical maintenance personnel,

mechanical maintenance personnel, radiological protection technician, chemistry technician, and engineering support personnel.

(3) Task analyses should provide detailed descriptions of what the personnel must do. The licensee should identify how human tasks or performance requirements are being changed.

(4) The task analysis should address the full range of plant conditions and situational factors, and performance shaping factors anticipated to influence human performance. The range of plant operating modes relevant to the HAs (e.g., abnormal and emergency operations, transient conditions, and low-power and shutdown conditions) should be included in the task analysis.

(5) The human task requirements that result from the changes in the actions should be assessed to determine whether they are compatible with each individual's responsibilities (i.e., will not interfere with or be disrupted by the cognitive and physical demands of other tasks and responsibilities).

(6) The task analysis should identify reasonable or credible, potential errors.

3.5 Staffing

Objective

The objective of this review is to verify that the licensee has analyzed the proposed change in HAs to determine the number and qualifications of personnel based on task requirements and applicable regulatory requirements. Adding additional manual actions or shifting tasks to periods of high workload may increase staffing needs. An example is a local manual action to temporarily replace an automatic action.

Scope

The staffing analysis addresses personnel needs for all conditions in which the HA may be performed.

Criteria

(1) Staffing levels should be evaluated to determine their adequacy with respect to any additional burden that may be imposed by the plant or HA modifications. The staffing levels should be adjusted if necessary. The evaluation should be based on an analysis of the following aspects:

- current nominal (typical shift complement of personnel) and minimal staffing levels (as identified in administrative procedures)
- required actions determined from the task analysis, if performed
- the physical configuration of the work environment (e.g., control room and control consoles configurations that may affect the ability of personnel to work together)

- the availability of plant information from individual workstations from individual and group view components of the HSI
- availability of personnel considering other activities that may be ongoing and for other possible responsibilities outside the control room (e.g., fire brigade)

3.6 Probabilistic Risk and Human Reliability Analysis

Objective

The objectives of this review are to verify that the licensee has (1) updated the PRA model to reflect system, component, and HA changes that may be necessary based on the proposed modification or HAs, (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, and (3) verified that the risk insights derived from the results are addressed in the selection of HAs, development of procedures, HSI components, and training in order to limit risk and the likelihood of personnel error and to provide for error detection and recovery capability.

Scope

This review addresses PRAs and HRAs conducted by the licensee to evaluate changes in systems, components, and human tasks that result from the proposed changes in HAs. Some of these items may have been addressed and reviewed as part of the screening process for the risk-informed submittal. In addition, the NRC human factors engineering reviewers may consult with the NRC risk analysis specialist on this review.

Criteria

(1) The PRA and HRA should be modified to reflect the changes in systems, components, and human tasks. Human interactions with plant systems and components should be analyzed at least at the level modeled in the plant's current PRA. Alternatively, justification should be provided as to why the change to the PRA is minimal and not necessary.

(2) The HRA should follow a structured, systematic, and auditable process that can be audited to provide adequate assurance that the reliability of each HA is accurately estimated so that its effect on plant safety using the PRA can be assessed.

(3) The PRA/HRA should address any human interactions that may be involved with the modified plant systems and components at the level currently modeled in the plant PRA, for example —

- errors of omission and commission
- miscalibration and component restoration errors
- recovery actions

(4) The analysis of HAs should include the identification of performance shaping factors (PSFs); that is, factors that influence human reliability through their effects on performance. PSFs include factors such as environmental conditions, HSI design, procedures, training, and supervision.

(5) HAs, associated with the modification, that are risk-important 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 events and external events (if available). 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.

(6) The licensee should use the information from the modified PRA/HRA to calculate changes in CDF, LERF, and integrated risk (if a temporary change is involved).

3.7 Human-System Interface Design

Objective

The objective of this review is to evaluate the HSI design, for those changes in HAs that require changes to the HSI, to verify that the licensee has appropriately translated function and task requirements into the detailed design of the HSI through the systematic application of HFE principles and criteria.

Scope

This review addresses the design of temporary and permanent modifications to the HSI, including new HSI components and the modification of existing ones, related to the proposed changes in the HAs, to verify that the existing HSI are appropriate for the modified human action. The review addresses aspects of the HSI and the work environment that affect the ability of the personnel to perform the HAs.

Criteria

(1) The HSI should be designed consistent with HFE guidelines and the existing HSI to the degree practical.

(2) The design should seek to minimize the probability that errors will occur and maximize the probability that errors will be detected and personnel will be able to recover from them.

(3) When developing HSI components for actions performed either in the control room or locally in the plant, the following factors should be considered:

- communication, coordination, and workload
- feedback
- local environment
- inspection, test, and maintenance

(4) The layout of HSI components within consoles, panels, and workstations should be based upon (1) analyses of human roles (job analysis) and (2) systematic strategies for organization such as arrangement by importance, frequency of use, and sequence of use.

(5) HSI characteristics for the changed action should support human performance under the full range of environmental conditions.

(6) Certain human tasks will need qualified instrumentation in accordance with RG 1.97 (NRC, 1983). The task analysis should identify the necessary safety grade of the control and display equipment used for human tasks. The RG defines Type A variables as "those variables to be monitored that provide the primary information required to permit the control room operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design-basis accident events" (NRC, 1983, p. 4). Primary information is further defined in the RG as information that is essential for the direct accomplishment of the specified safety functions, but does not include those variables that are associated with contingency actions that may also be identified in written procedures. Table 1 of RG 1.97 provides detailed Category 1 criteria that Type A variables should meet. In general, these Category 1 criteria provide for environmental and seismic qualification, redundancy, quality assurance, continuous display, good human factors design, and an emergency power supply. Therefore, HAs, which are required for safety systems to accomplish their safety function for design basis accident events and for which no automatic control is provided, will need control and display instrumentation in accordance with RG 1.97. (This RG allows for consideration of alternative approaches that are adequately justified and include consideration of the risk-significance of the actions involved.) Thus, credit should only be given for these types of HAs if they can be completed using control and display instrumentation that is consistent with RG 1.97. Information in Regulatory Issue Summary (RIS) 2002-22 (NRC 2002a) may also be useful.

3.8 Procedure Design

Objective

The objective of this review is to verify that applicable plant procedures have been appropriately modified, where needed, to provide adequate guidance for the successful completion of the HAs, and that the procedures adequately reflect changes in plant equipment and HAs. In the procedure development process, HFE principles and criteria should be applied along with all other design requirements to develop procedure modifications that are technically accurate, comprehensive, explicit, easy to use, and validated.

Scope

This review addresses all plant procedures that provide guidance to personnel for the affected actions, including the following types:

- emergency operating procedures (EOPs)
- plant and system operations (including startup, power, and shutdown operations)
- abnormal and emergency operations
- alarm response

The scope includes both temporary and permanent modifications to these procedures.

Criteria

(1) Where applicable, plant procedures should be modified to provide new instructions for the proposed changes in the HAs. Exceptions may be made where the adequacy of the existing procedures can be justified. Such a justification should indicate how the existing procedures provide necessary and sufficient guidance for the changed HAs and do not contain information that is inaccurate or no longer relevant.

(2) Where appropriate, procedures should identify how the operating crew should independently verify that the HAs have been successfully performed.

(3) All procedures should be verified and validated to provide adequate assurance that they are correct and can be carried out. Their final validation should be performed as part of the validation activities described in Section 3.10.

(4) Any changes in the HSI should be reflected in the modifications of the procedures.

(5) Procedural modifications should be integrated across the full set of procedures; alterations in particular parts of the procedures should not conflict nor be inconsistent with other parts. For example, an HSI component that is modified for a HA may also affect other actions that have not been modified. Therefore, procedure changes should not be limited to only the changed HAs.

3.9 Training Program Design

Objective

The objective of this review is to verify that the licensee's training program results in adequate training for the HAs. The review should verify that appropriate training has been developed and conducted for the HAs, including any changes in qualifications, as described in NRC Information Notice 97-78 (NRC, 1997).

Scope

This review addresses the licensee's training programs for all licensed and non-licensed personnel who perform the changed HAs. The scope includes both temporary and permanent modifications to training programs.

Criteria

(1) The licensee's training program should be modified to address the knowledge and skill requirements for all changes in HAs for the licensed and non-licensed personnel.

(2) Learning objectives should be derived from an analysis that describes desired performance for the HAs after training has been completed.

3.10 Human Factors Verification and Validation

Objective

The objective of this review is to verify that the licensee's verification and validation (V&V) program:

- Provides adequate assurance that the HFE/HSI design contains all necessary alarms, displays, and controls to support plant personnel tasks (HSI Task Support Verification).
- Provides adequate assurance that the HFE/HSI design conforms to HFE principles, guidelines, and standards (HFE design verification).
- Provides adequate assurance that the HFE/HSI design can be effectively operated by personnel within all performance requirements applicable to the HAs (integrated system validation), including the following:
  - All pertinent staffing considerations are acceptable for nominal and minimal shift levels, such as shift staffing, assignment of tasks to crew members, and crew coordination within the control room and between the control room and local control stations and support centers.
  - The HAs can be accomplished within time and performance criteria
  - The integrated system performance is consistent with all functional requirements, including tolerance of failures of individual HSI features

Scope

(1) The general scope of V&V includes the following factors as applicable to the proposed changes to the HAs:

- HSI hardware and software
- procedures
- workstation and console configurations

(2) The typical order of V&V activities is:

- HSI task support verification
- HFE design verification
- integrated system validation

(3) All V&V activities are applicable regardless of whether the change in HAs involve changes in the HSI.

Criteria

HSI Task Support Verification

(1) All aspects of the HSI (e.g., controls, displays, procedures, and data processing) that are required to accomplish the HAs should be verified as available through the HSI. For HAs associated with qualified instrumentation in accordance with RG 1.97, it should be verified that the HSI provides such qualified instrumentation.

HFE Design Verification

(1) All aspects of the HSI (e.g., controls, displays, procedures, and data processing) used for the HAs should be verified as consistent with accepted HFE guidelines, standards, and principles.

(2) Deviations from accepted HFE guidelines, standards, and principles should be acceptably justified on the basis of a documented rationale such as trade study results, literature-based evaluations, demonstrated operational experience, or tests and experiments.

Integrated System Validation

Validation Testbeds

(1) For HAs performed in the main control room, the plant training simulator should be used as the testbed when conducting the validation tests.

(2) For HAs performed at locations outside of the main control room, the use of a simulation or mockup can be used or drills conducted in the plant. The conduct of these drills should not interfere with plant operations (e.g., drills may be conducted when the plant is shutdown or the affected systems are removed from service).

Plant Personnel

(1) Participants in the validation tests should be the plant personnel who will perform the changed actions. Actions that will be performed by licensed personnel should be validated using licensed personnel rather than training or engineering personnel. Similarly, actions allocated to non-licensed personnel should be validated using non-licensed personnel.

(2) To properly account for human variability, more than one crew should participate in the validation tests. This will help provide adequate assurance that variation along most of the significant dimensions that influence human performance are included in the validation tests. Participation is not necessary for personnel who do not normally operate or maintain the plant (e.g., administrative personnel who hold operating licenses).

(3) In selection of personnel, consideration should be given to the assembly of nominal and minimum crew configurations, including shift supervisors, reactor operators, shift technical advisors, etc., that will participate in the validation tests. The composition of operations personnel need only include categories of personnel that are relevant to the HAs.

Operational Conditions

(1) Integrated system validation should consider the operational conditions for which each HA is required.

(2) The operational conditions should be developed into scenarios. The following information should be defined to provide adequate assurance that important performance dimensions are addressed and to allow scenarios to be accurately presented for repeated trials:

- description of the scenario mission and any pertinent "prior history"
- specific initial conditions
- events (e.g., failures) to occur and their initiating conditions (e.g., time, parameter values, or events)
- data to be collected and the specification of what, when and how data are to be obtained and stored
- specific criteria for terminating the scenario

(3) Scenarios should have appropriate task fidelity so that realistic task performance will be observed in the validation tests and so that results can be generalized to actual operation in the real plant.

(4) When evaluating performance associated with the use of HSI components located remote from the main control room, the effects on crew performance due to potentially harsh environments (i.e., high radiation) should be simulated (i.e., additional time to don protective clothing and access radiologically controlled areas).

Plant Performance Measurement

(1) The variables used in the performance measures should include performance of the plant and personnel, as described below.

(2) Measures that assess personnel task performance should be used, including the following:

- For each specific scenario, the tasks that personnel *are required to* perform should be identified and assessed. Such tasks can include necessary primary

(e.g., start a pump) and secondary (e.g., access the pump status display) tasks. This analysis should be used for the identification of errors of omission by identifying tasks which should be performed. The proper completion of required tasks should be verified.

- The tasks that are *actually* performed by personnel during simulated scenarios should be identified and quantified. The variable(s) used to quantify tasks should be chosen to reflect the important aspects of the task with respect to system performance, such as:
  - task success or failure
  - task completion time
  - errors (omission and commission)
  - subjective reports of participants

(3) Performance criteria for the measures used in the evaluations should be established.

Data Analysis and Interpretation

(1) Validation test data, time and errors, should be analyzed through a combination of quantitative and qualitative methods. For example, task time can be statistically compared with time available to perform the task and subjective reports of participants can be qualitatively evaluated to identify potential obstacles to performance.

3.11 Human Performance Monitoring Strategy

Objective

The objective of this review is to verify that the licensee has prepared a human performance monitoring strategy for ensuring that no adverse safety degradation occurs because of the changes that are made and to provide adequate assurance that the conclusions that have been drawn from the evaluation remain valid over time. A human performance monitoring strategy by the licensee will help to ensure that the confidence developed by the completion of the integrated system validation is maintained over time. There is no intent to periodically repeat the full integrated system validation, however, there should be sufficient evidence to provide reasonable confidence that operators have maintained the skills necessary to accomplish the assumed actions.

The results of the monitoring need not be reported to the NRC, but should be retained onsite for inspection.

Scope

The scope of the performance monitoring strategy should provide adequate assurance of the following:

- HFE/HSI design can be effectively operated by personnel, both within the control room and between the control room and local control stations and support centers.
- HAs can be accomplished within time and performance criteria.
- Integrated system performance is maintained within the performance established by the integrated system validation.

Criteria

(1) A human performance monitoring strategy should be developed and documented by the licensee. The strategy should be capable of tracking human performance after the changes have been implemented to demonstrate that performance is consistent with that assumed in the various analyses that were conducted to justify the change. Licensees may integrate, or coordinate, their performance monitoring for risk-informed changes with existing programs for monitoring operator performance, such as the licensed operator training program. If a plant change requires monitoring of actions that are not included in existing training programs, it may be advantageous for a licensee to adjust the existing training program rather than to develop additional monitoring programs for risk-informed purposes.

(2) The program should be structured such that (a) HAs are monitored commensurate with their safety importance, (b) feedback of information and corrective actions are accomplished in a timely manner, and (c) degradation in performance can be detected and corrected before plant safety is compromised (e.g., by use of the plant simulator during periodic training exercises).

4. LEVEL II REVIEW GUIDANCE

The guidance in this section is also a tailoring of NUREG-0711 (2004b) to plant modifications affecting HAs of medium risk significance. The guidance in this section reflects a further reduction of the criteria to reflect the level of risk imposed by the modification in Level II. Even with the tailored guidance provided in this section, the reviewer can further adapt the guidance to meet the unique demands of particular reviews.

4.1 General Deterministic Review Criteria

Objective

The objective of this section is to verify that deterministic aspects of design, as discussed in RG 1.174, have been appropriately considered by the licensee. Deterministic aspects include verifying that the change meets current regulations and does not compromise defense-in-depth.

Scope

The deterministic review criteria are applicable to all modifications associated with Level II HAs.

Criteria

(1) The licensee should provide adequate assurance that the change meets current regulations, except where specific exemptions are requested under 10 CFR 50.12 or 10 CFR 2.802. Examples of regulations that may be affected by a change, but that may be identified as risk-significant when using a standard PRA to screen for risk include the following: 10 CFR Part 20, Criterion 19 of Appendix A to 10 CFR Part 50, and Appendices C through R to 10 CFR Part 50.

(2) The licensee should provide adequate assurance that the change does not compromise defense-in-depth. Defense-in-depth is one of the fundamental principles upon which the plant was designed and built. Defense-in-depth uses multiple means to accomplish safety functions and to prevent the release of radioactive materials. It is important in accounting for uncertainties in equipment and human performance, and for ensuring some protection remains even in the face of significant breakdowns in particular areas. Defense-in-depth may be changed but overall should be maintained. Defense-in-depth includes the following important aspects:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- There is no over-reliance on programmatic activities to compensate for weaknesses in plant design.
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.

4.2 Analysis

Objective

The objective of the review is to verify that the licensee has analyzed the changes to HAs and identified HFE inputs for any modifications to the HSI, procedures, and training that may be necessary.

Scope

The review criteria are applicable to all modifications associated with Level II HAs.

Criteria

(1) *Functional and Task Analysis*

- The licensee should identify how the personnel will know when the HA is necessary, that is performed correctly, and when it can be terminated.
- Task analyses should provide a description of what the personnel must do. The licensee should identify how human tasks or performance requirements are being changed.
- The task analysis should identify reasonable or credible, potential errors and their consequences.

(2) *Staffing*: The effects of the changes in HAs upon the number and qualifications of current staffing levels of operations personnel for normal and minimal staffing conditions.

4.3 Design of Human System-interfaces, Procedures, and Training

Objective

The objective of the review is to verify that the licensee has supported the HAs by appropriate modifications to the HSI, procedures, and training.

Scope

The review criteria are applicable to all modifications associated with Level II HAs.

Criteria

(1) *HSIs*: Temporary and permanent modifications to the HSI should be identified and described. The modifications should be based on task requirements, HFE guidelines, and resolution of any known operating experience issues.

(2) *Procedures*: Temporary and permanent modifications to plant procedures should be identified and described. The modifications should be based on task requirements and resolution of any known operating experience issues. Justification should be provided when the plant procedures are not modified for changes in operator tasks.

(3) *Training*: Temporary and permanent modifications to the operator training program should be identified and described. The modifications should be based on task requirements and resolution of operating experience issues. Justification should be provided when the training program is not modified for changes in operator tasks.

4.4 Human Action Verification

Objective

The objective of this review is to verify that the licensee has demonstrated that the HAs can be successfully accomplished with the modified HSI, procedures, and training.

Scope

The review criteria are applicable to all modifications associated with Level II HAs.

Criteria

(1) An evaluation should be conducted at the actual HSI to determine that all required HSI components, as identified by the task analysis, are available and accessible.

(2) A walkthrough of the HAs under realistic conditions should be performed to determine that —

- The procedures are complete, technically accurate, and usable.
- The training program appropriately addressed the changes in plant systems and HAs.
- The HAs can be completed within the time criterion for each scenario that is applicable to the HAs.

The scenario used should include any complicating factors that are expected to affect the crews' ability to perform the HAs.

(3) The walkthroughs should include at least one crew of actual operators.

5. FINAL DECISION ON ACCEPTANCE OF HUMAN ACTIONS

Once the NRC review of a proposed licensee change (either risk informed or non-risk informed) in HAs is completed, a final decision regarding the acceptability of the human action aspect of the modification must be made. This decision will provide input to the NRC's overall decision whether to accept or reject the licensee's submittal. The final results of the HFE review will provide input to the Integrated Decision-making (see discussion in RG 1.174, Section 2.2.6) and may be documented in a safety evaluation report provided to the NRC project manager.

At this point a significant amount of information has been gathered, reviewed, and evaluated that can be used to assist in the final decision-making. This includes the following information:

- risk values related to the change or modification, including their location on the acceptance guideline figures
- time associated with the change
- results of the Level I or Level II review, which includes both human factors engineering information relating to the ability of operators to reliably perform the actions in question and deterministic aspects of the proposed change
- answers to requests for additional information (RAIs) that NRC has developed providing additional information or commitments
- other factors related to the plant in question that may bear on the decision

This information needs to be considered in an integrated fashion, that considers risk, but does not base the final decision on risk alone. RG 1.174 notes that the use of PRA technology should be increased in all regulatory matters, but it should be done in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy. RG 1.174 also notes that decisions concerning proposed changes are expected to be reached in an integrated fashion, considering traditional engineering and risk information, and may be based on qualitative factors as well as quantitative analyses and information. The review guidance in this document takes these concepts into consideration.

RG 1.174 notes that HAs in the high risk area of Region I are generally not desired, but there are examples of such actions in plants today (e.g., the PWR ECCS switch over situation described in Generic Issue B-17). Also, there may be extenuating circumstances in which the licensee can adequately justify a modification to add a Region I HA (e.g., if the change is temporary or if there are other changes that lower the CDF).

Another important consideration is whether and how well the licensee has addressed the HFE aspects of the modification. The results of the HFE analyses discussed in Sections 2, 3, and 4 must be considered in an integrated manner. No individual analysis is sufficient in and of itself. Thus, the decision will not be driven solely by the numerical results of the PRA. Each type of information helps in building an overall picture of the implications of the proposed change on risk. The PRA has an important role in putting the change into its proper context as it impacts the plant as a whole. As the discussions in the previous section indicate, both quantitative and qualitative arguments may be brought to bear. The different pieces of evidence that are used to make a final decision may not be combined in a formal way, but they do need to be clearly

documented. The proposed change should be given increased NRC management attention when the calculated values of the changes in the risk metrics approach the criterion levels of current, accepted guidelines.

The main factors in the decision process are discussed here first and then supplementary decision factors are listed that may assist when the decision is difficult to make.

Main Decision Factors

- **Change in CDF**: One consideration is the value of ΔCDF_{mod} or the increase in CDF attributable to the modification. The placement of this value into the regions of Figure 2.1 can also be considered. The confidence one has in the PRA HEP value and, hence, that the change in CDF is at the value shown by ΔCDF_{mod} is partially determined by the results of the HFE review noted below.
- **Change in LERF**: Another consideration is ΔLERF, similar to the change in CDF, above.
- **Risk-Importance Measures for the HA**: The values of RAW and FV give a measure of the risk-importance of the HA in question. (Chapter 2 discusses the specific meanings of these measures.) These provide insight on the contribution of the HA to risk.
- **Time and Integrated Risk**: A further consideration is the length of time that the change will be in place, if the modification temporary. The integrated risk over time (or ICCDP and ICLERP) can be considered, as discussed in Chapter 2.
- **Human Factors**: A most important consideration is the degree of confidence that operators can perform the actions required for the modification in question. This is determined by the aggregate evaluation in Sections 3.2 through 3.12 of the Level I review guidance and Sections 4.2 through 4.4 of the Level II review guidance.
- **Deterministic Criteria**: Another consideration is the more traditional deterministic review guidance provided in Section 3.1 of the Level I review guidance and Section 4.1 of the Level II review guidance.

Supplemental Decision Factors

Additional factors may also be used, as appropriate, to determine the acceptability of a change. These were adapted from RG 1.174 Section 2.2.6, "Integrated Decision-making":

- the cumulative effect of previous changes and the trend in CDF (the licensee's risk management approach)
- the cumulative effect of previous changes and the trend in LERF (the licensee's risk management approach)
- the effect of the proposed change on operational complexity, burden on the operating staff, and overall safety practices
- plant-specific performance and other factors (for example, siting factors, inspection findings, performance indicators, and operational events), and Level 3 (offsite consequence) PRA information, if available

- the benefit of the change in relation to its CDF/LERF increase
- the practicality of accomplishing the change with a smaller CDF/LERF impact
- the practicality of reducing CDF/LERF when there is reason to believe that the baseline CDF/LERF are above the guideline values (i.e., 1E-4 and 1E-5 per reactor year)

6. REFERENCES

10 CFR Part 50. "Domestic Licensing of Production and Utilization Facilities." Washington, DC: U.S. Government Printing Office.

ANSI/ANS-58.8-1994 (1994). "Time Response Design Criteria for Safety-Related Operator Actions." La Grange Park, Illinois: American Nuclear Society.

Azarm, A., J. Higgins, and L. Chu (1999). "Development of a Risk-Informed Baseline Inspection Program" (BNL Report No. JCN W6234). Upton, New York: Brookhaven National Laboratory.

Higgins, J., and J. O'Hara (2004). "The Development of Guidance for the Review of Changes to Human Actions" (BNL Letter Report Y6022-T8-8/2004). Upton, New York: Brookhaven National Laboratory.

Higgins, J., J. O'Hara, W. Stubler, and R. Deem (1999). "Summary of Credit of Past Operator Action Cases" (BNL Report No. W6022-T1-1-7/99). Upton, New York: Brookhaven National Laboratory.

Higgins, J., J. O'Hara, P. Lewis, and J. Persensky (2002). "Development of a Risk Screening Methodology for Credited Operator Actions," in *Proceedings for the 2002 IEEE 7th Conference on Human Factors and Power Plants.* Phoenix, Arizona: Institute of Electrical and Electronics Engineers.

Lambert, H.E. (1975). *Measures of Importance of Events in Reliability and Fault Tree Analysis*, ed. R.E. Barlow, J.B. Fussell, and N.D. Singpurwalla. Philadelphia, PA: SIAM Press.

NEI-00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance." Washington, DC: Nuclear Energy Institute.

NRC (1983). "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environmental Conditions During and Following an Accident" (Regulatory Guide 1.97). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (1991). Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," Enclosure 2, NRC Inspection Manual, Part 9900, Technical Guidance, STS10P, Change No. 90-015, dated October 31, 1991, "Operable/Operability: Ensuring the Functional Capability of a System or Component, Section 6.7, *Use of Manual Action in Place of Automatic Action.*" Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (1997). "Crediting of Operator Action in Place of Automatic Actions and Modification of Operator Actions, Including Response Times" (Information Notice 97-78). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (1997a). "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" (NUREG-1560). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (1998). "An Approach for Plant-Specific, Risk-Informed Decision-Making: Technical Specifications" (Regulatory Guide 1.177), August, 1998. Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (1999). SECY-99-007A, "Recommendations for Reactor Oversight Process Improvements" (Followup to SECY-99-007). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2001). "NRC Regulatory Issue Summary 2001-02: Guidance on Risk Informed Decision-Making in License Amendment Reviews" (RIS 2001-02). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2002a). "NRC Regulatory Issue Summary 2002-22: Use of EPRI/NEI Joint Task Force Report, 'Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule'" (RIS 2002-22). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2002b). "Human-System Interface Design Review Guidelines" (NUREG-0700, Rev. 2). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2002c). NRC Inspection Manual, Chapter,0609, "Significance Determination Process, Appendix A, Attachment 2, Site Specific Risk-Informed Inspection Notebook Usage Rules." Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2002d). "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800), Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance." Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2002e). "An Approach to Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Regulatory Guide 1.174, Rev. 1). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2003). NRC Inspection Manual, Manual Chapter 0609, "Significance Determination Process." Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2004). "Guidance for the Review of Changes to Human Actions: Final Report" (NUREG 1764). Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2004a). "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800), Chapter 18, "Human Factors Engineering." Washington, DC: U.S. Nuclear Regulatory Commission.

NRC (2004b). "Human Factors Engineering Program Review Model" (NUREG-0711, Rev. 2). Washington, DC: U.S. Nuclear Regulatory Commission

NRC (2007). "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results For Risk-informed Activities" Regulatory Guide 1.200, January 2007, Washington, DC: U.S. Nuclear Regulatory Commission.

Vesely, W., T. Davis, R. Denning, and N. Saltos (1983). "Measures of Risk Importance and Their Applications" (NUREG/CR-3385). Washington, DC: U.S. Nuclear Regulatory Commission.

7. GLOSSARY

Component: An individual piece of equipment such as a pump, valve, or vessel, usually part of a plant system.

Function: An action that is required to achieve a desired goal. Safety functions are those functions that serve to ensure higher-level objectives and are often defined in terms of a boundary or entity that is important to plant integrity and the prevention of the release of radioactive materials. A typical safety function is "reactivity control." A high-level objective, such as preventing the release of radioactive material to the environment, is one that designers strive to achieve through the design of the plant and plant operators strive to achieve through proper operation of the plant. The function is often described without reference to specific plant systems and components or the level of human and machine intervention that is required to carry out this action. Functions are often accomplished through some combination of lower-level functions, such as "reactor trip." The process of manipulating lower-level functions to satisfy a higher-level function is defined here as a control function. During function allocation, the control function is assigned to human and machine elements.

Human-system interface (HSI): The means through which personnel interact with the plant, including the alarms, displays, controls, and job performance aids. Generically this includes maintenance, test, and inspection interfaces as well.

Human factors: A body of scientific facts about human characteristics. The term covers all biomedical, psychological, and psychosocial considerations; it includes, but is not limited to, principles and applications in the areas of human factors engineering, personnel selection, training, job performance aids, and human performance evaluation (see "Human factors engineering").

Human factors engineering (HFE): The application of knowledge about human capabilities and limitations to plant, system, and equipment design. HFE provides reasonable assurance that the design of the plant, systems, equipment, human tasks, and the work environment are compatible with the sensory, perceptual, cognitive, and physical attributes of the personnel who operate, maintain, and support the plant (see "Human factors").

Mockup: A static representation of an HSI (see "Simulator").

Performance criteria: The criteria against which measured performance is compared in order to judge its acceptability. Approaches to the establishment of performance criteria include:

> *Requirement Referenced*: This is a comparison of the performance of the integrated system with respect to an accepted, quantified, performance requirement. For many variables a requirement-referenced approach can be used; i.e., requirements for plant, system, and operator performance can be defined through engineering analysis as part of the design process. Plant parameters governed by technical specifications and time requirements for important operator actions are examples of performance measures for which a requirement-referenced criteria can be determined. For performance measures where such specific requirement referenced criteria cannot be used alternative criteria development methods must be used.

Benchmark Referenced: This is a comparison of the performance of the integrated system with that of a benchmark system which is predefined as acceptable under the same conditions or equivalent conditions. Such an approach is typically employed when no accepted independent performance requirements can be established. Performance is evaluated through comparisons to an accepted benchmark rather that through an absolute measurement. For example, the evaluation may test whether the plant under review can be operated to stay within a level of operator workload not exceeding that associated with Plant X. Plant X is identified as acceptable for reasons such as its acceptable operating history and operators report their workload levels to be acceptable. In this case the performance measure must be obtained for Plant X and the new system, under similar operational conditions, and then compared. In the establishment of benchmark-referenced criteria, similar test conditions should be established for the benchmark system and system under evaluation.

Normative Referenced: Normative-referenced comparison is similar to a benchmark reference comparison, however, the performance criterion is not based upon a single comparison system, it is based upon norms established for the performance measure through its use in many system evaluations. The new system performs as compared to the norms established under the same conditions or equivalent conditions. This approach can be used when no accepted independent performance requirements can be established, but repeated use of the same performance measure enables the development of performance norms for acceptable and unacceptable systems.

Expert-Judgement Referenced: This is a comparison of the performance of the integrated system with criteria established through the judgement of SMEs.

Performance shaping factors (PSFs): Factors that influence human reliability through their effects on performance. PSFs include factors such as environmental conditions, HSI design, procedures, training, and supervision.

Primary tasks: Those tasks performed by the operator to supervise the plant, i.e., monitoring, detection, situation assessment, response planning, and response implementation.

Risk-important human action: An action that must be performed successfully by operators to ensure plant safety. There are both absolute and relative criteria for defining these risk important actions. From an absolute standpoint, a risk-important action is one whose successful performance is needed to ensure that predefined risk criteria are met. From a relative standpoint, the risk-important actions constitute the most risk-significant human action identified.

Requirement: This document uses the term "requirements" in two different ways: (1) requirements that are established as part of the design process; e.g., design requirements, functional requirements, task requirements, etc.; and (2) regulatory requirements identified in 10 CFR. There are no regulatory requirements established in this document.

Safety-related operator action: A manual action required by plant emergency procedures that is necessary to cause a safety-related system to perform its safety-related function during the course of any design-basis event. The successful performance of a safety-related operator action might require that discrete manipulations be performed in a specific order.

Secondary tasks: Those tasks that the operator must perform when interfacing with the plant, but are not directed to the primary task. Secondary tasks may include navigating through and paging displays, searching for data, choosing between multiple ways of accomplishing the same task, and making decisions regarding how to configure the interface.

Simulator: A facility that physically represents the HSI configuration and that dynamically represents the operating characteristics and responses of the plant in real time (see "Mockup").

System: An integrated collection of plant components and control elements that operate alone or with other plant systems to perform a function.

Task: A group of activities that have a common purpose, often occurring in temporal proximity, and utilize the same displays and controls.

Testbed: The representation of the human-system interface and the process model used in testing.

Validation: The process by which the integrated system (consisting of hardware, software, and personnel elements) is evaluated to determine whether it acceptably supports safe operation of the plant.

Validity: The characteristics of the methods and tools used in the validation process. See the specific uses of the term: construct validity, convergent validity, performance representation validity, statistical conclusion validity, system representation validity, and test design validity.

Verification: The process by which the human-system interface design is evaluated to determine whether it acceptably reflects personnel task requirements and HFE design guidance.

Vigilance: The degree to which an operator is alert.

Workload: The physical and cognitive demands placed on plant personnel.

APPENDIX A.

GENERIC HUMAN ACTIONS THAT ARE RISK-IMPORTANT

This attachment contains two tables of generic HAs for BWRs and PWRs that are risk-important. Each table is further divided into "Group 1" HAs that are risk-important and "Group 2" HAs that are potentially risk-important. To facilitate readability of the tables, the names of common events and plant systems are given in acronyms. These acronyms are defined in the front matter of this document. These tables are used in the Generic HA Method (Method 2) described in Section 2.4.3.2.

Table A.1 Generic BWR Human Actions That Are Risk-Important

Group 1: BWR Human Actions That Are Risk-Important	
Human Actions	**Description and Reasons for Risk-Importance**
Perform manual depressurization	On selected sequences, such as station blackout (SBO), manual depressurization is required after failure of high pressure injection systems to allow for injection with low pressure systems. A complicating factor is that some procedures initially direct the operator to inhibit ADS. In some PRAs, this appears in cutsets up to 45% of CDF. Operators typically depressurize by manually operating the safety relief valves (SRVs).
Vent containment	On a transient or loss-of-coolant accident (LOCA) sequence, with failure of the PCS, containment temperature and pressure increase and must be controlled. This can be done by containment heat removal, suppression pool cooling, or containment venting. Actions are required to remove decay heat before adverse conditions are reached (e.g., high suppression pool temperature leading to loss of ECCS pumps).
Align containment or suppression pool cooling	
Actions during shutdown	Almost all actions, including actuation of various equipment, are done manually during shutdown. The operator's understanding of the plant configuration is necessary for the successful manual actions.

Group 2: BWR Potentially Risk-Important Human Actions	
Human Actions	**Description and Reasons for Risk-Importance**
Level Control in anticipated transient without scram (ATWS)	Effective reactor vessel level manual control at lower than normal levels (e.g., near the top of the active fuel) is needed during an ATWS in order to reduce core power.
Initiate standby liquid control (SLC)	Manual initiation of SLC is needed for ATWS sequences.
Inhibit automatic depressurization system (ADS)	Some IPEs conclude that core damage will occur if ADS is not manually inhibited in an ATWS event due to instabilities created at low pressures.
Miscalibrate pressure switches	Various pressure switches are important for initiating ECCS and operating ECCS permissives. Common cause mis-calibration of these switches can affect multiple trains of safety systems.
Initiate isolation condenser (IC)	For early design BWR plants, this action is important during accidents to ensure continued viability of the cooling from the IC.
Control feedwater (FW) events	The actions of operators to properly control the FW system as an injection source after loss-of-instrument air or other loss of FW events can be important in various sequences, such as transients and small LOCAs.
Recover offsite power	The actions of operators to recover offsite power after a total loss of offsite power (LOOP) is important to limit the risk due to station blackout and other LOOP core damage sequences. These are modeled with various recovery times in PRAs.
Shedding of direct current (DC) load after SBO	While often not well modeled, operator action to shed DC loads is needed to extend the battery charge in order to operate the alternating current (AC)-powered independent high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems and to keep the SRVs open (to allow low pressure vessel injection from a diesel-driven fire pump). This extends the time to core damage and the time that operators have for recovery of AC power.
Similar actions to those in Group I	Actions that are substantially similar (but not identical) to those contained in Group 1 of this table should be considered as potentially risk-important, if they involve the same systems, components, or actions.
Actions involving the risk-important systems	Each plant has a few systems that are clearly the most risk-significant in the plant. Human actions associated with these systems should be considered as potentially risk-important. When modifications associated with these risk-important systems are being considered, new human actions may be created that were not in the original PRA, but that will be risk-important. (See Section 2.4.3.2.)

Table A.2 Generic PWR Human Actions That Are Risk-Important

Group 1: PWR Human Actions That Are Risk-Important	
Human Actions	**Description and Reasons for Risk-Importance**
Establish recirculation	In LOCA scenarios, the switching of ECCS lines from the injection to the recirculation mode is done manually. Failure to do so or human error involving the valve alignment is important. Both low pressure and high pressure recirculation modes were noted to be important.
Feed and bleed	Failure of the operator to initiate and perform the feed and bleed operation of the reactor coolant system as a last resort of heat removal is important. Of particular importance is the bleed portion using the pressurizer pressure-operated relief valves (PORVs).
Provide water supply for auxiliary feedwater (AFW)	Use of water pumps to transfer water, from other sources of make up to the CST for use by AFW, is considered important in scenarios when long-term cooling through the steam generator (SG) is needed.
Reactor coolant pump (RCP) trip	On a loss of cooling to the RCP seals, it is important for operators to quickly trip the pumps to prevent an RCP seal LOCA.
Action during shutdown	Almost all actions, including actuation of various equipment, are done manually during shutdown. The operator's understanding of the plant configuration is necessary for the successful manual actions.

Group 2: PWR Potentially Risk-Important Human Actions	
Human Actions	**Description and Reasons for Risk-Importance**
Recover RCP seal cooling	In some plants there are means of alternate cooling for RCP seals that could be relied on in scenarios involving loss of CCW. However, the alignment of the system is manual and requires operator action.
Recover emergency AC or offsite power	Some losses of AC power can be recovered by either manual transfer of the source of power, or recovery of onsite normal/emergency AC power. This recovery action is considered risk-significant in many PRAs.
Actions in response to ATWS	Upon failure of RPS, the operator should perform several actions, starting with manual scram, ensuring turbine trip, and most importantly initiating emergency boron injection.
Depressurization (DEP) and equalization during SGTR event	An important strategy during a steam generator tube rupture (SGTR) event is the depressurization of primary and secondary systems and the equalization of pressures between primary and secondary. These all help to limit the leakage.
Isolate SG	During both a main steamline break (MSLB) and an SGTR event, isolation of the affected SG is important.
Shut PORV block valve	During a stuck-open PORV event, shutting the PORV block is an important action to eliminate the leak.
Isolate ISLOCA	In some plants there is a capability to isolate an interfacing systems LOCA through manual actions. Operator failure to isolate an interfacing LOCA in the low-pressure injection (LPI) system is considered risk-significant in these plants.
Similar actions to those in Group I	Actions that are substantially similar to those contained in Group 1 of this table should be considered as potentially risk-important, if they involve the same systems, components, or actions.
Actions involving the risk-important systems	Each plant has a few systems that are clearly the most risk-significant in the plant. Human actions associated with these systems should be considered as potentially risk-important. When modifications associated with these risk-important systems, are being considered new human actions may be created that were not in the original PRA, but that will be risk-important. (See Section 2.4.3.2.)

APPENDIX B.

CHANGES INCORPORATED IN REVISION 1 TO NUREG-1764

(1) The lead-in discussion in Section 2.1, "Licensee Change Requests Involving Human Actions," was clarified as it relates to the selection of HAs in a submittal for screening.

(2) A sentence was added to Section 2.3.3, "Human Factors Screening Approach for Temporary Changes (Step 1)," noting that if a temporary change to an HA will be in place for more than 1 year, it should be screened as a permanent change.

(3) A new subsection, "Identification of Human Actions," was added to Section 2.3.4, to address the following aspects of screening:

- *Treatment of HAs that are affected by a modification, but are not clearly identified in the submittal package.* An example identified in some actual submittals is the recovery of offsite power. This action can be affected by a modification but has not always been addressed in licensee submittals, since it is typically not modeled as an HA in the PRA.
- *Treatment of cases in which an HA is in the EOPs for post accident performance.* The HEP may be high and set to 1.0 in the PRA for screening purposes, thus taking no credit for the HA. In these cases, the HA should not automatically be assigned to Level III or left unreviewed. The HA may still be important and may have a high FV in the PRA.
- *Treatment of cases where the time available for the HA after a modification has increased, resulting in a lower HEP.* The HEP may still have a high RAW value so the reviewer must decide whether to screen the HA or simply conclude that there is no change. Step 3 provides a method to identify these circumstances and to reduce the level of review, if appropriate.

(4) When multiple HAs are affected by a modification, the question may arise as to which HAs to screen. Guidance explicitly addressing this situation was added to the "Multiple Human Actions" subsection of Section 2.3.4.

(5) The likelihood of dependency increases as there are more HAs; thus, special attention should be given to determining dependency when a submittal contains a large number of HAs. Guidance addressing this situation was added to the "Multiple Human Actions" subsection of Section 2.3.4.

(6) The lead in to Section 2.4, "Screening Process for Non-Risk-Informed Change Requests," was modified to indicate that if sufficient up-to-date risk information is available, but the submittal is officially non-RI, the NRC may still perform the screening using the RI method in Section 2.3 of this NUREG-series report.

(7) More descriptive names were provided for Methods 1 and 2 of the non-RI screening in Section 2.4.3.

(8) Section 2.4.3.1, "Estimated Importance Method (Method 1)," was updated with additional guidance on the use of the RI inspection notebooks, RAW values from the NRC's RI benchmarking reports, and the NRC's SPAR models.

(9) The possible use of the Fussell-Vesely importance measure with the non-RI Method 1 to 2 was added to Section 2.4.3.1, "Estimated Importance Method (Method 1)."

(10) Section 2.4.3.2, "Generic HA Method (Method 2)," was revised to coordinate the last paragraph on p. 27 of NUREG-1764, Rev. 0, with the last line of the Group 2 tables in Appendix A.

(11) In addition to the changes listed above, minor editorial changes were made throughout.

NRC FORM 335 (9-2004) NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1764, Revision 1

2. TITLE AND SUBTITLE

Guidance for the Review of Changes to Human Actions

3. DATE REPORT PUBLISHED

MONTH	YEAR
09	2007

4. FIN OR GRANT NUMBER

Y6022

5. AUTHOR(S)

P. Lewis, J. Persensky, J. Bongarra, S. Cooper, G. Parry/NRC

6. TYPE OF REPORT

Regulatory

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

A. Szabo, NRC Project Manager; J. Bongarra, Technical Monitor; J. Persensky, Senior Technical Advisor

11. ABSTRACT *(200 words or less)*

The U.S. Nuclear Regulatory Commission (NRC) reviews changes in human actions, such as those that are credited in nuclear power plant safety analyses. This document provides guidance for reviewing those changes. The first phase is a screening analysis of the licensee's proposed modification and the affected human actions to assess their risk-importance. A graded, risk-informed approach is used to determine the appropriate level of human factors engineering review. This approach can be accomplished for licensee submittals that are either risk-informed or non-risk-informed. For risk-informed submittals, the first phase has four steps: (1) use of Regulatory Guide (RG) 1.174 to determine the risk-importance of the entire plant change or modification that involves the human action, (2) quantification of the risk-importance of the human action itself, (3) qualitative evaluation of the human action, and (4) integrated assessment to determine the appropriate level of human factors engineering review. For non-risk informed submittals, a similar process is used which includes the use of generic risk information to determine the safety significance of the HA in place of the first two steps used in a risk-informed submittal.

The NRC's review of licensee submittals that involve changes to human actions is an iterative process. The final results of the human factors engineering review provide input to integrated decision-making and a safety evaluation report. This revision (Rev. 1) contains relatively minor changes and clarifications in the text (virtually all of them in Chapter 2). Appendix B lists the most significant changes.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Operator Actions, Workload, Workarounds, Human Reliability, Guidance for Human Actions, Human Actions, Guidance for Operator Actions, Risk-informed, Graded Approach to Review, Human Factors, Regulatory Guide 1.174, Risk-Informed Decision-making, Licensing, License change requests, Graded approach, quanitification of risk importance, human factors review, Licensing Basis, Human Factors in Safety Evaluation Reports, NUREG-0800 - Human Factors Engineering, Standard Review Plan

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS