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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

August 13, 2001

MEMORANDUM TO: Dr. Dana A. Powers, Chairman
Human Factors Subcommittee

Dr. Mario V. Bonaca, Chairman
Safety Research Program Subcommittee

FROM: Michael T. Markley, Senior Staff Engineer

SUBJECT: FUTURE ACTIVITIES DURING 485th ACRS MEETING: CONCERNING NRC HUMAN RELIABILITY ANALYSIS RESEARCH PLAN: FISCAL YEARS 2001-2005

The purpose of this memorandum is to forward the staff document entitled, NRC Human Reliability Analysis Research Plan: Fiscal Years 2001-2005, for consideration by the Committee during discussion of future activities at the 485th ACRS meeting, September 5-8, 2001.

Background

In late July 2001, the staff forwarded its human reliability analysis (HRA) research plan for fiscal years 2001-2005 in support of the NRC Risk-Informed Regulation Implementation Plan (RIRIP). The plan describes the proposed tasks the staff plans to examine lessons learned from current research activities, development activities to improve HRA methods and tools in selected areas, and application to support ongoing risk-informed decisionmaking. This initiative includes data collection and analysis activities associated with the NRC Program on Human Performance in Nuclear Power Plant Safety. The HRA Research Plan considers international research in this area and is intended to address ACRS advice in its report on NRC Safety Research (NUREG-1635).

Expected Committee Action

Please advise me of any additional materials that you feel are necessary for you to make a recommendation to the Committee during discussion of future activities during the September 5-8, 2001 ACRS meeting.

The ACRS has had a long-standing interest in HRA and the NRC's Human Performance Plan. It is likely that the Committee will be very interested in hearing a briefing on this matter and may consider this a focused topical area for the 2002 ACRS report on NRC Safety Research.

Attachments: As Stated

cc w/o attach: ACRS Staff and Fellows

Draft Report for Information

NRC Human Reliability Analysis Research Plan: Fiscal Years 2001-2005

May 2001

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U.S. Nuclear Regulatory Commission
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Acronyms

ACRS	Advisory Committee on Reactor Safeguards (NRC)
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ATHEANA	A Technique for Human Event Analysis
CDF	Core Damage Frequency
COOPRA	International Cooperative PRA Research Program
CSNI	Committee on the Safety of Nuclear Installations
EFC	Error Forcing Context
EPRI	Electric Power Research Institute
FY	Fiscal Year
HFE	Human Failure Event
HRA	Human Reliability Analysis
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronic Engineers
INEEL	Idaho National Engineering and Environmental Laboratory
IPEEE	Individual Plant Examination of External Events
LP&SD	Low Power and Shutdown
MOU	Memorandum of Understanding
NFPA	National Fire Protection Association
NMSS	Office of Nuclear Material Safety and Safeguards (NRC)
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
OECD	Organization for Economic and Cooperation and Development
PHP	Program on Human Performance in Nuclear Power Plant Safety
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
R&D	Research and Development
RES	Office of Nuclear Regulatory Research (NRC)
SDP	Significance Determination Process
SNL	Sandia National Laboratories
UA	Unsafe Action

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Executive Summary

It is widely recognized that human actions that depart from or fail to achieve what should be done, can be important contributors to the risk associated with the operation of nuclear power plants. This recognition is based upon actual operating experience and the results of probabilistic risk assessment (PRA) studies. Human reliability analysis (HRA), the process for identifying potentially important human failure events and assessing their likelihood, is therefore an important part of PRA and an essential tool for risk-informed regulation.

HRA is currently being used (as part of PRA) to support a variety of decision making activities. However, depending on the situation being analyzed, there can be significant uncertainties in the results of currently available methods. Numerous reviews have identified areas for improvement that would extend the range of the methods, improve their ability to identify correct decisions, or increase the confidence of decision makers in the results of these methods. These areas include the adequacy of the data being used, the treatment of the scenario-specific context for operator actions, and the role of organizational influences.

Recent HRA research and development work conducted by the U.S. Nuclear Regulatory Commission (NRC) has focused on the development of A Technique for Human Event Analysis (ATHEANA). ATHEANA is an HRA method aimed at addressing the issue of scenario-specific context and a particularly challenging topic in HRA: the treatment of errors of commission. ATHEANA's underlying premise is that significant human errors occur as a result of a combination of influences associated with plant conditions and specific human-centered factors that trigger error mechanisms in the plant personnel. This premise requires the identification of these combinations of influences, called the "error-forcing contexts" (EFCs), and the assessment of their influence.

Recognizing that the ATHEANA development process has made sufficient progress to allow the application of ATHEANA (as one tool in the HRA toolbox) in actual regulatory applications (e.g., the analysis of pressurized thermal shock scenarios in support of a potential change to 10 CFR 50.61), and that the NRC has a broad range of HRA research and application needs which need to be addressed, the NRC staff has developed an HRA Research Program Plan to guide its efforts in the next few years.

This report documents the HRA Research Program Plan. This plan includes research and development tasks (to develop improved HRA methods and tools in selected areas) and HRA applications tasks (to support ongoing risk-informed decision making activities). It builds on the results of past HRA research (both conducted at NRC and other organizations), and includes joint data collection and analysis activities with the NRC Program on Human Performance in Nuclear Power Plant Safety, and collaboration activities with various international research groups. It also includes tasks aimed at ensuring that the HRA Research Program effectively communicates its results to users and other interested stakeholders.

This report contains: an overall description of the research program (including the program objectives, background issues, related regulatory activities, and related NRC and non-NRC research activities); the program's specific technical objectives for Fiscal Years 2001-2005; the overall technical approach and tasks; project management information; and a discussion of the

plan for effectively communicating the program results. Table E1 lists the tasks included in the research program; Table E2 lists key milestones. Note that the task list includes a number of program support activities (indicated by letters) as well as technical activities (indicated by numbers). The report shows that the research program supports the NRC's Risk-Informed Regulation Implementation Plan (RIRIP), and that the program is appropriately coordinated with other relevant programs within and outside the NRC.

Table E1 - Human Reliability Analysis Research Program Tasks, FY 2001-2005

Task	Title
1	HRA Data Collection and Analysis
2	HRA Guidance Development
3	HRA Quantification and Uncertainty
4	Pressurized Thermal Shock HRA
5	Fire HRA
6	Steam Generator Tube Rupture HRA
7	HRA for Aging Cable Systems
8	HRA for Materials and Waste Applications
9	Reactor System Synergisms Effects and HRA
10	HRA for Upgraded and Advanced Control Rooms
11	Latent Errors in HRA
12	HRA Extended Applications
13	Formalized Methods: Screening, Individual and Crew Modeling
A	HRA Research Planning
B	HRA Results Communication
C	General NRC HRA Technical Support
D	Industry and International HRA Activities

Table E2 - Human Reliability Analysis Research Program Key Milestones, FY 2001-2002

Key Milestone (FY 2001-2002)	Date
Complete draft of initial research program plan for review	December 2000
Organize and host WGRISK workshop on errors of commission	May 2001
Complete framework for HRA quantification	June 2001
Develop HRA research lessons to support risk-informed regulatory applications	September 2001
Complete initial evaluation of HRA data needs	September 2001
Present proposed quantification process at WGRISK workshop	October 2001
Hold first workshop/seminar on key research results	December 2001
Provide HRA input to PTS technical basis analysis report	July 2002
Complete review and evaluation of observed latent errors	September 2002
Complete evaluation of symptom-based procedures data	September 2002
Develop improved methods and tools for LP&SD HRA	September 2002

1. Introduction

This report documents the human reliability analysis (HRA) research program plan for Fiscal Years (FY) 2001-2005 currently being executed by the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES). The report describes the program objectives, the tasks being taken to accomplish these objectives, the intended uses of the program outputs, and the relationship between the research program and other related NRC programs, including the NRC Program on Human Performance in Nuclear Power Plant Safety (PHP) documented in SECY-00-0053 [1].

This section provides background information and an overview of the HRA program plan. The details of the plan are provided in the remainder of the report.

1.1 Program Purpose and Objectives

The overall purpose of the HRA research program is to support the NRC's Risk-Informed Regulation Implementation Plan (RIRIP) [2], which has been developed to implement the NRC's strategic plan, especially with respect to a number of the performance goals in the Nuclear Reactor Safety and Nuclear Materials Safety strategic arenas [3]. The general objectives of the program are as follows.

- Develop improved human reliability analysis (HRA) methods, tools (including guidance), and data needed to support NRC regulatory activities, including the broad implementation of risk-informed regulation.
- Develop HRA results and insights to support the development of technical bases for addressing identified or potential safety issues.
- Provide HRA support for the planning and execution of NRC programs and activities (e.g., the PHP) outside the immediate scope of the RIRIP.
- Ensure effective communication of research results to end users.
- Ensure effective use of resources in satisfying the preceding objectives.

The specific technical objectives for FY 2001-2005 are listed in Section 2 of this report. These specific objectives are based on a consideration of current and anticipated reactor safety and materials safety staff user needs, on HRA research results developed by NRC and others (especially with respect to the treatment of context in accident scenario analysis - see Section 1.2.3 below), and on the recognition that NRC's risk-informed regulatory needs are likely to require a variety of HRA tools (including guidance as well as analysis methods). Potential longer term activities (post-FY 2005) are discussed in Section 5.

1.2 Background

This section provides background material on the role of human reliability analysis (HRA) in probabilistic risk assessment (PRA), the regulatory needs for HRA, the current state of the HRA technology and discipline, and past NRC HRA research and development efforts.

1.2.1 HRA in Probabilistic Risk Assessment

Human reliability analysis (HRA) is a process for identifying potentially important human failure events [1] and assessing their likelihood. For probabilistic risk assessment (PRA) applications, HRA provides a means for systematically incorporating the current understanding of human behavior into the PRA. In typical applications, the role of HRA can be conceptualized as follows. (For a more complete description, see [4].)

Consider the general event tree/fault tree model for a generic engineered system shown in Figure 1. The effects of possible human actions (both positive and negative) are incorporated through the definition of appropriate human failure events (HFEs). These HFEs can appear as top events in the event tree or as basic events in the fault trees for the event tree top events. They can be used to represent failures of appropriate actions (including the failure to take action) before the initiating event (i.e., "latent errors") or after the initiating event (i.e., "dynamic errors"). They can also be used to represent actions causing the initiating event.

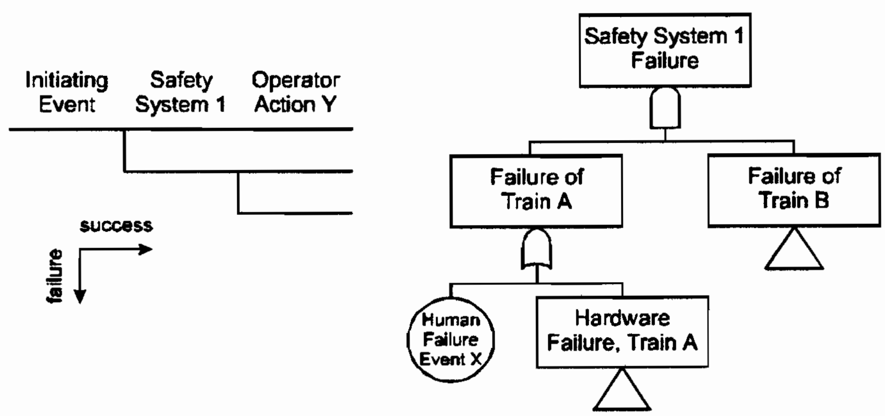

Figure 1. Schematic Diagram - Role of HRA in PRA

[1] In this report, the term "human failure event" is used instead of the more generic "human error" to avoid an implication of blame (e.g., for situations where operators follow their training, but the training is inappropriate for the situation) and to provide an explicit tie with the probabilistic risk assessment of which the HRA is a part.

Previous analyses and past experience from operational events show that credible HFEs can usually be identified for most (if not all) of the safety functions treated in PRAs. It is therefore not surprising that HRA generally plays an important role in PRA (see, for example, a review of the results of Individual Plant Examinations [5]), and that uncertainties in HRA results are often an important driver in the uncertainty in the overall results of the PRA. Consequently, it is important to have HRA methods, tools, and data that can, for a given risk-informed decision, adequately assess the human contribution to risk. These methods, tools, and data need to support the identification of potentially significant HFEs and the quantification of the likelihood of these HFEs. The quantification process should appropriately address dependencies of the HFEs on the scenario (including other HFEs in the scenario) and the uncertainties in the probabilities of the HFEs. Depending on the needs of the particular decision, the analysis may need to be performed at a sufficient level of detail to support the identification of key causes of error and their potential fixes. Section 1.2.3 provides a summary discussion on the current HRA state of the art and the current state of the HRA discipline. Section 1.2.4 discusses past NRC efforts in HRA research and development.

1.2.2 Risk-Informed Regulatory Initiatives and HRA Needs

As stated in the NRC's policy statement on the use of PRA [6], the NRC intends to increase the use of PRA technology in "all regulatory matters to the extent supported by the state of the art in PRA methods and data." Recent general activities include efforts to make Part 50 of the Code of Federal Regulations more risk-informed [7];[2] the updating of the general risk-informed framework for supporting licensee requests for changes to a plant's licensing basis, described in a revised version of Regulatory Guide (RG) 1.174 [8]; the revision of the reactor oversight process to incorporate risk information [9]; and the increasing use of PRA in evaluating the significance of operational events [10]. A potentially key supporting element is the development of industry-consensus standards on the performance of a PRA. Regarding such standards, NRC is supporting activities by the American Society of Mechanical Engineers (ASME) [11], the American Nuclear Society (ANS) [12], and the National Fire Protection Association (NFPA) [13], as indicated in the RIRIP.

It can be seen from the discussion in Section 1.2.1 on the relationship between HRA and PRA that HRA plays an important role in all of the above activities, and can be expected to play a major role in any specific NRC initiative that employs PRA results and insights. Two examples showing how HRA is currently playing a major role are as follows.

- Pressurized Thermal Shock (PTS) rule revision. As described in SECY-00-0140 [14], the NRC is currently developing the technical basis for modifying the Pressurized Thermal Shock (PTS) screening criteria specified in title 10 Part 50.61 of the U. S. Code of Federal Regulations. As part of this effort, a PTS PRA is being performed, and an HRA is being performed as part of this PRA. A key issue in the PRA is the identification and

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[2]As stated in Ref. 2, risk-informed regulation is "an approach to regulatory decision making that uses risk insights as well as traditional considerations to focus regulatory and licensee attention on design and operational issues commensurate with their importance to health and safety."

quantification of HFEs that will lead to overcooling of the reactor vessel. (Note that most PRAs are aimed at undercooling scenarios and therefore typically assume that actions to cool the reactor coolant system will lead to successful termination of the accident.)

- Spent fuel pool risk assessment for decommissioned plants. The NRC staff has recently completed a scoping PRA for spent fuel pools at decommissioned plants [15]. The intent of this evaluation is to provide a technical basis for decision making regarding emergency planning requirements. A key part of the PRA is the treatment of plant personnel recovery actions as credit for these actions can significantly reduce the estimated level of risk. The HRA challenge is to assess the likelihood of unsuccessful actions over the very long (e.g., several day) accident durations modeled.

In the future, it is anticipated that additional demands on HRA will arise, due to changes in the nuclear industry. These changes may involve changes to operating plants (through plant aging, the influences of deregulation, and plant upgrades using digital technology) and the possibility of radically new plant designs. These changes will likely affect human performance, and will therefore affect HRA. In fact, these changes may require a new set of HRA methods, tools, and data to appropriately identify and quantify risk-significant HFEs.

Table 1 provides a list of RIRIP implementation activities with potential HRA needs. The list includes activities which may have needs for improved methods, tools, and data to enable the staff to review licensee analyses and to perform independent analyses. It also includes activities which may have needs for HRA results and insights as part of the technical bases for regulatory decisions. Note that the need for HRA is not limited to regulatory decisions involving human performance issues; some form of HRA is required for most risk-informed decisions because the results of the HRA can and typically do affect the risk significance of even non-human performance related issues.

1.2.3 HRA: The State-of-the-Art and The Discipline

As indicated in Section 1.2.2, and as discussed in a 1998 OECD/CSNI review of HRA modeling and data issues [16], HRA is currently being used (as part of PRA) to support a variety of decision making activities. This does not mean that there are no weaknesses in the HRA state of the art. However, it is important to point out that, in many practical situations, current HRAs can be good enough to: a) support the identification of the correct decision,[3] and b) provide confidence to the decision maker that the identified decision is indeed the correct one. Therefore, any improvements in HRA need to be aimed at extending the range of HRA (to support a broader range of decisions), improving the ability of HRA to help identify the correct decision, or improving the confidence of the decision maker in the results of the HRA.

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[3]Note that, lacking the advantage of hindsight for the types of events typically addressed by PRA, the "correctness" of a decision can be viewed as a function of the degree to which available relevant information is used. To best support decision making, therefore, the HRA method used needs to appropriately reflect the current state of knowledge about human behavior under the conditions of interest to the PRA.

Table 1. RIRIP Implementation Activities with Potential HRA Needs*

- Regulation
  - risk-informing special treatment requirements (RSMS8-1)
  - risk-informing particular rules in 10 CFR Part 50, e.g., combustible gas control, ECCS, fire protection, PTS (RSMS8-2, RSMS8-4, RSMS8-6)
  - assessing regulatory effectiveness (RSEER1-6)

- Licensing
  - establishing guidance for risk-informed licensing basis changes (RSMS5-1)
  - review advanced reactor designs (RSEER1-1)
  - perform PRA for spent fuel dry storage cask (WSMS1-2)

- Oversight
  - developing baseline inspection programs (RSMS1-2)
  - assessing the significance of inspection findings (RSMS1-4)
  - developing performance indicators (RSMS1-3, RSMS3-1)

- Event analysis
  - analyzing accident sequence precursors (RSMS3-4)

- Research and development
  - developing standards for application of risk-informed, performance-based regulation in conjunction with national standards committees (RSEER1-2)
  - developing improved methods for calculating risk (RSEER1-3)

*The codes in parentheses are the specific RIRIP implementation activity identifiers

The HRA methods widely used in current PRA applications include task-oriented decomposition methods (e.g., the Technique for Human Error Rate Prediction - THERP [17]), time-reliability correlations (e.g., the Human Cognitive Reliability/Operator Reliability Experiments - HCR/ORE - method [18]), and judgment-based methods (e.g., the Success Likelihood Index Method-Multi Attribute Utility Decomposition Method - SLIM-MAUD [19]). These methods have an important common feature: they all recognize that the context of the human failure event (HFE) being analyzed affects the likelihood of that HFE. Furthermore, they attempt to make direct use of available data in quantifying the probability of the HFE. (SLIM-MAUD also addresses another important issue: the possibility of biases introduced when using expert judgment.)

However, it is widely recognized that there are significant uncertainties in the results of these methods. Numerous reports and papers have been written on this subject (see, for example, [4, 16, 20]). Key sources of uncertainty raised in these reviews, as well as in much earlier reviews (e.g., the Lewis Commission's review of WASH-1400 [21]) include the adequacy of the data used to support HRA, and the adequacy of the current understanding of human behavior under accident conditions. Regarding the latter, one concern raised over the years by researchers is the lack of explicit treatment of one particular aspect of HFE context: post-initiator dynamic plant behavior (i.e., the evolution of plant conditions over time). The coupling of this behavior with the operators' training and procedures was a significant aspect of the Three Mile Island accident in 1979. Other scenario-specific complicating factors, including multiple equipment failures and faulty instrumentation readings, have been significant contributors in actual operational events yet are, at best, treated as operator workload issues. Another issue raised with widely used HRA methods involves the role of organizational processes and factors (e.g., work processes and safety culture) as another contextual factor for multiple HFEs within an accident sequence and across multiple sequences [22]. Because of their pervasive influence (which affects the degree of dependency between HFEs and therefore the joint probability of their occurrence), these factors may have a significant impact on risk. A related issue, recently raised based on a review of core damage accident precursors [23], is the adequacy of current HRA methods for dealing with "latent errors," i.e., pre-initiator HFEs.

Numerous research efforts have been initiated over the last decade to provide improved HRA methods that address these issues. (Key NRC activities are summarized in Section 1.2.4.) In particular, methods aimed at providing improved treatment of post-initiator accidents (including the NRC-supported ATHEANA method [24]) have been developed and are starting to be used in trial applications. Many of these methods are reviewed in a recent CSNI study looking at "errors of commission" [4] [25], and some of these methods are starting to be used in decision-support applications (see, for example, [26, 27]). Regarding the treatment of organizational factors, the state of the art is less advanced. A number of methods (e.g., see [28, 29]) have been proposed and undergone some degree of testing, but it appears that further work is needed before these methods can be used in a decision support role. More generally, progress to date on improved HRA methods has generally not been sufficient to allow widespread use of these new methods. One reason for the slow progress is the difficulty of the problem being addressed. Another is a lack of coordination within the field.

The HRA discipline has, over the years, been characterized as being in need of improvement. In 1990, Dougherty stated that HRA needed a "second coming," i.e., the development of models representing "a synthesis of the various irrefutable points of all of the contrary views" [30]. In 1997, Lydell argued that, as HRA researchers strove to develop these second generation models, large amounts of energy were being wasted because these researchers were not paying sufficient attention to (and thereby benefitting from) each others' work, were not talking with analysts performing HRA in PRA applications, and did not understand the needs of these PRA applications [31]. In a summary of a 1994 workshop on HRA challenges, Blackman stated that HRA "is a fractured discipline lacking in consensus" [32]. His summary identifies needs in the areas of HRA model development, data, and communication between researchers.

These calls for improvements in cooperation have not been ignored. In recent years, efforts have been made to develop a common understanding of the relationships among the different methods. For example, international cooperative activities have led to comparisons of

[4]Refs. 16 and 25 define an error of commission as an inappropriate action, particularly one that might occur during the response to a transient or an accident, that places the plant in a situation of higher risk. Other definitions found in the literature are similar but not identical.

current and developing HRA approaches, and to joint recommendations for future activities [16, 25]. (Table 2 shows that these recommendations are consistent with those of the 1994 HRA workshop summarized by Blackman [33].) An informal working group (called MOSAIC) has been formed to further interactions between HRA researchers.

Based upon the results of these activities, and upon recent presentations of HRA research activities, it appears that the HRA research community may be moving towards a reasonable degree of consensus on a number of issues. For example, there appears to be fairly broad agreement that more causally-based methods are needed to better identify, model, and quantify potentially risk significant post-initiator HFEs to support current PRA structures, and that these methods should explicitly consider potential "error forcing contexts" (EFCs), i.e., combinations of plant conditions and other influences that make operator errors more likely. It is further generally recognized that treatment of plant conditions requires some consideration of the plant's dynamic behavior during the accident. There also appears to be general agreement that, although specific actions directly affecting a plant are taken by individuals, organizational and environmental factors (including safety culture) strongly affect those actions, and therefore are also an important part of the EFC.

However, there is disagreement over how some of these issues should be addressed. Apparent areas of disagreement include:

- Terminology and definitions
- Appropriate levels of model decomposition
- Precise definition and treatment of the EFC for a given HFE (e.g., which contextual factors should be treated deterministically, which should be treated through the use of HFE probability modifiers, and which factors need not be explicitly addressed)
- The role of generic data (and generic HFE estimates) in HFE quantification
- The adequacy of the current (static) PRA framework for defining scenario context

Nevertheless, it appears that there is a reasonable degree of consensus within a good portion of the HRA research community as to: a) the basic nature of the problem to be solved (e.g., the treatment of HFEs as part of a PRA - see [4]), b) important areas for research and development (e.g., the appropriate definition of scenario-specific context), and c) the need for increased cooperation to accelerate progress.

In summary, improved methods for dealing with recognized issues in HRA (e.g., the importance of addressing the scenario-specific context for an HFE) are being developed and some of these are starting to see practical applications. All of these methods appear to require additional testing, improvements, and stronger data bases before they are ready for routine application in day-to-day problem solving. Many (if not most) HRA methods developers have recognized the need for improved cooperation, in order to more quickly provide necessary tools to support PRA. As seen in Section 3 of this report, the HRA research program plan includes activities intended to actively encourage such cooperation.

Table 2. Summaries of Recommendations From a Sample of HRA Workshops and Collaborative Projects*

1994 Workshop on Human Reliability Models (recommendations for evolutionary models) [33]

1. Develop shared definitions, common language, common system of analysis
2. Assess the reliability and validity of proposed tools, data, and information.
3. Improve task analysis methods so that they can systematically identify the context for a given action.
4. Address what the operator thinks.
5. Address the influence of teamwork and organizational factors.
6. Find better ways to use existing information (e.g., simulator experiment results).
7. Focus data collection programs on two levels: errors during normal operations and errors in near misses and actual accidents.

1998 OECD/CSNI Report on Human Reliability Modeling and Data Issues [16]

1. Validate HRA methods.
2. Avoid standardizing HRA methods to the point that stagnation results.
3. Intensify data collection efforts.
4. For potential future use, pay attention to work on emerging HRA techniques.
5. For potential future use, pay attention to work on emerging dynamic PSA methods.
6. Increase cooperation.

2000 OECD/CSNI Report on Errors of Commission [25]

1. Work to better understand how operators work. Move towards "human-centered analysis" rather than "equipment-centered analysis".
2. Establish a database that includes qualitative aspects of errors of commission (EOCs).
3. Improve knowledge to support quantification.
4. Develop guidance for standardizing documentation of error opportunities/error forcing contexts.
5. Develop guidance on the role of traditional task analysis.
6. Improve the incorporation of knowledge from behavioral sciences to support human error identification, analysis, and quantification.
7. Improve guidance and procedures for identifying likely EOC opportunities and for screening important EOCs.

*Note: the recommendations listed above are paraphrased from the source reports.

1.2.4 NRC HRA Research Activities

The previous section has pointed out that the current HRA methods and data need improvement in a number of areas. To ensure efficient use of resources in developing these improvements, the research program needs to build upon the results of previous and parallel efforts. This section briefly summarizes key NRC-supported HRA research and development activities. [5] Section 1.4 discusses a number of other ongoing research activities in human factors and HRA.

In 1972, NRC (then the Atomic Energy Commission) initiated the landmark Reactor Safety Study (WASH-1400) [36] in order to better understand the risks associated with the operation of commercial reactors in the U.S. As part of this study, it was recognized that methods were needed to assess the likelihood of human errors. To address this need, WASH-1400 used the THERP methodology developed at Sandia National Laboratories in the early 1960's (see the discussion by Swain [37]). Somewhat later, NRC supported the formalization of the lessons learned from the WASH-1400 application in an HRA Handbook, NUREG/CR-1278 [38]; this report was updated in 1983 [17]. In the mid-1980's, as part of its Accident Sequence Evaluation Program (ASEP), NRC supported the development of a simplified version of THERP [39], the SLIM-MAUD method [19], and a time-reliability approach (based on plant simulator data) [40] that was used in the Risk Methods Integration and Evaluation Program (RMIEP) [41]. It is important to recognize that THERP, ASEP, and SLIM-MAUD are still widely used in practical PRA applications. It has also been suggested that THERP is still an adequate method for treating pre-initiator HFEs, although improvements are needed for the treatment of dependent failures, e.g., those due to organizational influences [42].

In the mid-1980's, NRC also started a number of research projects looking at the effect of nuclear power plant organization on safety. As part of this effort in the early 1990's, a preliminary method for relating measurable organizational data and PRA parameters was developed [43]. The data collected characterized, among other things, the function and behaviors of different parts of the organization, the interactions between these different parts, and the organizational cultures of these different parts. A SLIM-MAUD technique for converting the observations into changes in human error probabilities was proposed and demonstrated. During this time, the NRC also supported: a) work exploring dynamic event tree and discrete event simulation methods for addressing the post-initiator dynamic interactions between the control room operating crew and the plant, and the interactions within the crew [44, 45], and b) the development of a process for performing HRA [46].

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[5]There is, of course, a substantial body of HRA literature developed by industry and international research and development programs. Examples include an industry-developed procedure for performing HRA as part of a PRA [34], an HRA method incorporating the results of simulator experiments [18], a procedure for treating errors of commission which has been applied (at a screening level) in a non-U.S. nuclear power plant (NPP) PRA [35], and the current and advanced HRA methods covered by the OECD/CSNI reviews [16, 25]. The purpose of this section is not to provide a general HRA literature review, but to describe key NRC contributions to the field.

In the early 1990's, following some unexpected predictions from a French study, NRC started investigating low power and shutdown risk. Based on a review of human errors during a number of operational events (see NUREG/CR-6093 [47]), it was determined that errors of commission would need to be explicitly treated, and that an improved HRA approach was needed to do this. Initial work (see NUREG/CR-6265 [48]) showed that the ideas being developed were transferable to HRA for all operational modes. Consequently, it was decided to attack the broader problem and address all operational modes through the development of an improved HRA method that has come to be called ATHEANA (A Technique for Human Event Analysis).

The basic premise of ATHEANA, following the work of earlier pioneers (including Reason [49] and Woods [50, 51]) and substantiated by reviews of a number of significant accidents both within and without the nuclear industry, is that significant human errors occur as a result of a combination of influences associated with plant conditions and specific human-centered factors that trigger error mechanisms in the plant personnel. This premise requires the identification of these combinations of influences, called the "error-forcing contexts" (EFCs), and the assessment of their influence.

Since the project's inception, a number of activities have been performed. These include the performance of retrospective analyses of actual events (to better understand the characteristics of the EFCs observed in real life), the performance of limited scope studies (including a test application to a Three Mile Island-like scenario) to demonstrate ATHEANA's capability in prospective analyses, two independent peer reviews to identify areas for improvement in the method, and the development of improved methods to systematically search for EFCs. The results of these activities are documented in NUREG/CR-6093 [47], NUREG/CR-6265 [48], NUREG/CR-6350 [52], and NUREG-1624, Rev. 1 [24].

A draft version of NUREG-1624, Rev. 1 has been reviewed by the NRC's Advisory Committee on Reactor Safeguards (ACRS) [53]. Among the issues raised in that review were the need for a screening method (as part of ATHEANA), and the need for an improved quantification method. In a more recent review of the NRC reactor safety research activities, the ACRS recommended that "the ATHEANA effort be terminated and a new plan for the quantification of the probability of human unsafe acts be developed" [54]. The full set of ACRS recommendations and their resolution are provided in Appendix A.

Currently, ATHEANA is being applied in support of an activity developing the technical basis for modifying the Pressurized Thermal Shock (PTS) screening criteria specified in title 10 Part 50.61 of the U. S. Code of Federal Regulations (the Pressurized Thermal Shock Rule). The reasons for this application are two-fold. First, as noted in an Advisory Committee on Reactor Safeguards (ACRS) review of an earlier NRC analysis, PTS scenarios provide potential operational problems, including "the conflicting need to maintain adequate pressure for core cooling purposes while avoiding PTS," and "control of feedwater and auxiliary feedwater to provide adequate core cooling while avoiding overcooling" [55]. Thus an HRA method capable of dealing with these issues, including potential errors of commission, is needed. Second, the application of ATHEANA as part of an actual regulatory analysis is expected to lead to valuable feedback that will support the identification of necessary process-related and technical improvements for HRA.

Although most of the recent NRC HRA methods development activity has focused on ATHEANA, a second NRC-supported HRA methods development activity is worth noting. In 1994, the then Office for the Analysis of Evaluation of Operational Data (AEOD) initiated the development of an improved method for addressing human contributions to core damage accident sequence precursors (ASP). The purpose of the project was to replace a highly simplified HRA method then in use by the ASP program with an improved method which would account for key aspects of human behavior, yet would still be simple enough to be used by non-HRA specialists. The results of the work, which used THERP and ASEP as a starting point, are documented in Ref. 56. The method is currently being used in a variety of NRC applications (not just ASP analyses) largely as a screening tool.

In summary, NRC HRA research activities have led to methods, tools, and results that are currently being used in a number of applications. Further, the work has provided useful results in two important areas where HRA improvements are needed: the treatment of post-initiator HFEs (including errors of commission), and the treatment of organizational influences on HRA. The research tasks described in Section 3 of this report are designed to build on these results, as well as key results of non-NRC HRA research and development activities.

1.3 Program Outputs and Regulatory Uses

Early NRC HRA research work has provided many of the "first generation" HRA tools (e.g., the HRA Handbook, ASEP, SLIM-MAUD) that have enabled NRC and licensees to pursue risk-informed regulation. Recent NRC HRA research activities have also provided a number of products for immediate use in regulatory applications. These include:

- ATHEANA support of the PTS work developing the technical basis for modifying the PTS Rule (see the discussion in the preceding section).
- Provision of a simple HRA tool (the ASP HRA method discussed in the preceding section) that has been used in regulatory analyses (e.g., the analysis of risk associated with spent fuel pools at decommissioned facilities [15]).
- Support of the NRC's new Significance Determination Process through the development of a simple tool for incorporating remote shutdown operations issues into the evaluation of fire protection inspection findings [57].

In addition, recent research efforts have resulted in the publication of a number of reports on the ATHEANA method. These reports are expected to provide the basis for HRA developments which will be used in supporting future risk-informed regulation activities.

It is recognized that the HRA Research Program needs to ensure that its results are useable in regulatory applications, and that the NRC needs to make broader use of the results when these are produced. For example, as indicated in discussions with the ACRS, it is recognized that the HRA research results need to play a greater role in the refinement of the NRC's human factors research under the Program for Human Performance (PHP). It also appears that there is a greater potential role for NRC's HRA research results in the ongoing development of HRA standards by the ASME and the IEEE. The HRA Research Program plan addresses the issue of results communication in Section 4.

1.4 Relationship with Other Ongoing Research Programs and Activities

This section briefly identifies other relevant research programs, and discusses the relationship between the HRA Research Program and these other programs. The listing of programs is not intended to be exhaustive; particular attention is paid to programs with which the HRA Research Program will interact with, or is planning to interact with.

1.4.1 NRC Research Programs and Activities

- NRC Program on Human Performance in Nuclear Power Plant Safety (PHP)

As stated in SECY-00-0053 [1], the mission of the PHP is "to ensure that reactor safety is maintained through effective regulation and oversight of human performance in the design, operation, maintenance, and decommissioning of nuclear reactor facilities." The mission "will be accomplished by: (1) identifying human performance issues important to public health and safety, (2) increasing understanding of the causes and safety implications of these human performance issues, and then (3) implementing the appropriate regulatory response to human performance issues."

The primary interface between the HRA Research Program and the PHP identified in SECY-00-0053 concerns data: the PHP is tasked with providing data needed for HRA (where it is understood that "data" can come in a wide variety of forms, including qualitative information relevant to HRA models being developed). For example, the PHP has recently completed a study which suggests that latent errors may have more impact on plant risk than previously recognized, and that they may require improved treatment in HRAs [23]. Task 11 of the HRA Research Program (see Section 3 of this report) is a follow-up to the PHP study.

More generally, it is recognized that the specification of data needs, and of the appropriate means to meet these needs, are not trivial tasks. The former issue is linked to the question of quantification, i.e., how the data will be used in determining the probability of an HFE. Addressing the second issue requires experimental design considerations. Therefore, it is anticipated that the HRA data "interface" between the two programs will be redefined as a joint activity. This joint activity is covered by Task 1 of the HRA Research Program, as described in Section 3 of this report.

It is also anticipated that there will be other areas of interaction between the PHP and the HRA Research Program in activities specified in the PHP plan, including: the development of a Human Performance Evaluation Protocol, the development of a technical basis for guidance to review plant changes which credit operator actions, the development of Commission policies regarding human performance issues (e.g., fatigue), and the efforts preparing for the future (e.g., regarding control station design review guidance and deregulation). In particular, it is anticipated that results and insights from the HRA Research Program will provide feedback to the PHP to support the planning and execution of PHP activities, and that the PHP will identify cases for which HRA improvements are needed.

- Pressurized Thermal Shock (PTS) - Development of a Technical Basis to Support Revision of 10 CFR 50.61

As discussed in Sections 1.2.2 and 1.2.4, the NRC is in the process of developing the technical basis for modifying the Pressurized Thermal Shock Rule. The HRA Research Program is providing support to this effort through: a) the application of ATHEANA in PTS PRA analyses for two plants, and b) the reviews of PTS HRAs performed for two additional plants. Noting the challenges posed by PTS scenarios discussed in Section 1.2.4, it is expected that this work will lead to improved HRAs for all four plants. It is also expected that the lessons learned from the work will help identify needed HRA improvements.

This support activity is covered by Task 4 of the HRA Research Program, as described in Section 3 of this report.

- NRC Fire Risk Research Program

One of the upcoming tasks in NRC's Fire Risk Research Program is the performance of a "fire risk requantification study," i.e., a re-analysis of fire risk at a selected number of plants, to determine if the application of the new fire risk assessment (FRA) methods, tools, and data developed under the program will have a significant impact on the FRA results and insights [58]. It is expected that ATHEANA will be used to perform the HRA portion of the requantification study. Similar to the ATHEANA application to PTS, it is expected that ATHEANA will be useful because of the potentially challenging error forcing contexts (including instrumentation failures) associated with fires. It is also expected that the application (which is not aimed at resolving an immediate regulatory question, but will likely result in tools supporting future risk-informed regulatory applications) will help identify needed HRA improvements.

This support activity is covered by Task 5 of the HRA Research Program, as described in Section 3 of this report.

- Steam Generator Tube Rupture (SGTR) PRA Modeling Improvements

In FY 2000, the Office of Nuclear Reactor Regulation (NRR) requested that the Office of Nuclear Regulatory Research (RES) develop improved "probabilistic safety assessment modeling of these [SGTR] scenarios, including the effects of operator actions" [59]. It is anticipated that, similar to the PTS and fire risk efforts, ATHEANA will provide the basis for the HRA tools developed.

This support activity is covered by Task 6 of the HRA Research Program, as described in Section 3 of this report.

- NRC Cable Aging Research Program

As part of the NRC's Environmental Qualification Task Action Plan, RES is conducting a cable research program addressing, among other things, the performance of aged low-voltage I&C cables [60]. The staff currently plans to assess the risk significance of the failures of these cables. Due to the potential impact of these cable failures on operator performance (e.g., through the possibility of spurious indications and signals, as well as loss of function), it is expected that HRA will have a major role in the risk assessment effort, and that ATHEANA will be a useful tool for performing the HRA. This activity is covered by Task 7 of the HRA Research Program, as described in Section 3 of this report.

- Dry Cask PRA

In support of the Office of Nuclear Materials Safety and Safeguards (NMSS), RES is concurrently performing a screening-level PRA for a dry cask spent fuel storage system [61]. This screening PRA will identify areas that may need further analysis. It is anticipated that the HRA performed as part of the PRA will employ currently available methods, tools, and data. This activity is covered by Task 8 of the HRA Research Program, as described in Section 3 of this report. Task 8 also covers HRA activities addressing a broader range of nuclear materials and waste PRA applications.

- NRC Digital I&C Research Program

The NRC Digital Instrumentation and Control (I&C) Research Program is concerned with, among other things, the risk implications of the introduction of digital I&C technology into existing plants and the use of such technology in new plants (including advanced reactors) [62]. Given that the evaluation of these implications will require an integrated treatment of hardware, software, and humans, and that the Program on Human Performance also has activities associated with these issues, it is expected that the analysis will require a coordinated effort by the Digital I&C Research Program, the PHP, and the HRA Research Program. This activity is covered by Task 10 of the HRA Research Program, as described in Section 3 of this report.

- Rulemaking Plan to address Fatigue of Workers at Nuclear Power Plants

The NRC staff is in the process of finalizing a SECY paper requesting Commission approval for the staff to proceed with developing a rulemaking addressing the issue of worker fatigue at nuclear power plants. HRA research may be needed to support assessments of the effect of fatigue on plant risk.

- PRA Standards

As mentioned in Section 1.2.2, the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) are in the process of developing standards for PRA. These standards will include requirements for the performance for HRA. Although the results of the HRA Research Program have not, to date, been formally used in the standards development process, NRC staff and contractors contributing to the HRA Research Program have provided input to that process. Furthermore, as indicated in Section 4 of this report, it is intended that greater efforts will be made to ensure that future HRA Research Program results (including a description of lessons learned to date, as discussed in Section 3.3. of this report) are used to support revisions of these standards, and the development of an upcoming Institute for Electrical and Electronics Engineers (IEEE) HRA standard.

- Dynamic PRA

Under a cooperative research agreement with RES, the University of Maryland is currently exploring the use of "dynamic PRA methods" [63], i.e., methods designed to explicitly address plant dynamics (including the interaction between the plant and the operators) during accident scenarios. The modeling tool developed will be based on earlier work performed at the University of Maryland (e.g., see [64]); it will integrate a thermal hydraulic simulation code (RELAP5) with a dynamic event tree model for the plant and a model for the operating crew that addresses team interactions and individual cognition. This work may lead to an improved definition of context for such HRA methods as ATHEANA (which are designed to be used in a conventional event tree/fault tree structure). It may also eventually provide a practical means to address human actions in a more natural framework than that currently used in PRAs.

The test case chosen for the project is pressurized thermal shock. The project will therefore use information developed during the PTS PRA work, including the ATHEANA model and results (see Task 4). It is also expected that there will be technical interactions between the project team and the HRA Research Program members to exchange information on their approaches to the modeling of human actions.

1.4.2 Other Programs and Activities

As can be seen from a review of the HRA literature, there is a considerable amount of international activity concerned with the development of improved HRA methods for nuclear power plant applications. Through the NRC's participation on the OECD/CSNI risk working group (WGRISK) and in the International Cooperative PRA Research Program (COOPRA), the HRA Research Program has access to the research (and researchers) of many of the countries with active programs (e.g., Finland, France, Germany, Hungary, Japan, Korea, Spain, Switzerland, United Kingdom). As described under Support Task D of this research plan (see Section 3), significant efforts will be made to pursue cooperative activities that will accelerate the progress of all involved. Task D will also address activities supporting interactions with Electric Power Research Institute (EPRI) HRA efforts, which were described at a May 2001 OECD/CSNI workshop on errors of commission held at NRC Headquarters.

In addition to pursuing cooperative activities with other HRA research programs, the NRC will work with a number of existing human factors research programs to develop data needed to support HRA. The list of programs will include the ongoing Halden program and the recently initiated program at the Korea Atomic Energy Research Institute (with whom NRC has recently signed a memorandum of understanding). It is may also include work being done by the National Aeronautics and Space Administration (NASA) and at the Nuclear Power Engineering Corporation (NUPEC) of Japan.

1.5 How To Use This Report

The remainder of this report covers the HRA research program's specific technical objectives (Section 2); project leadership, schedule, and milestone information (also Section 2); the overall technical approach and tasks (Section 3); a discussion of the plan for effectively communicating the program results (Section 4); and a discussion of potential activities following

FY 2005 (Section 5). The list of references is provided following Section 5. Appendix A lists recommendations received from ACRS and shows how these recommendations will be addressed.

Readers interested in a program overview should consult Section 2, Section 3.1, Section 4, and Section 5. Readers interested in the details of a specific technical task should consult Sections 3.2 through 3.14; these sections provide the technical objectives and approach for each task, as well as limited background notes concerning the motivation for the tasks. Similarly, Sections 3.15 through 3.18 discuss necessary support tasks (e.g., collecting feedback information and revising the program plan) for the research program.

2. Technical Objectives and Project Management, FY 2001-2005

2.1 Technical Objectives

The overall objectives of the HRA Research Program are presented in Section 1.1. The technical objectives for FY 2001-2005 are listed below under each overall objective. [6]

- Develop improved human reliability analysis (HRA) methods, tools (including guidance), and data needed to support NRC regulatory activities, including the broad implementation of risk-informed regulation.

  Define the qualitative and quantitative data needs of HRA.
  Collect and analyze data to support HRA model development and quantification.
  Develop lessons learned from past HRA research activities to support risk-informed regulatory applications.
  Develop guidance for HRA analysts and reviewers to support risk-informed regulatory applications.
  Develop and test a formal approach to HRA quantification.
  Develop an improved HRA approach for post-severe accident steam generator tube rupture scenarios.
  Develop, as needed, improved HRA methods and tools to support risk-informed materials and waste applications.
  Develop improved HRA methods and tools to support PRAs for reactors with upgraded or advanced control rooms.
  Develop improved HRA methods to identify, model, and quantify latent errors.
  Develop improved HRA methods and tools to address post-initiator ex-control room actions, low power and shutdown operation, long term recovery actions, and severe accidents.
  Develop a screening method for use in the application of context-focused HRA methods (e.g., ATHEANA).
  Develop and test models addressing cognition and team issues for use in HRA.

- Develop HRA results and insights to support the development of technical bases for addressing identified or potential safety issues.

  Provide HRA support for Pressurized Thermal Shock PRA [14].
  Provide HRA support for fire risk assessment [58].
  Provide HRA support for Steam Generator Tube Rupture PRA [59].
  Provide HRA support for the risk evaluation of aged I&C cables [60].
  Provide HRA support for dry cask storage PRA [61].
  Provide HRA support for the evaluation of the risk significance of reactor systems synergisms.

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[6] The technical objectives are listed in the order of the technical tasks which they support (see Section 3). The listing is not intended to reflect relative priorities among the tasks.

2.2 Project Management

2.2.1 Key Personnel [7]

Technical Oversight:	N. Siu (NRC/RES)
Project Management:	E. Thornsbury (NRC/RES)
Principal Investigator, Tasks 2-6, 10, 12-13:	J. Forester (SNL)
Principal Investigator, Tasks 1, 7, 11:	D. Gertman (INEEL)
Principal Investigator, Task 8:	B. Hallbert (INEEL)

2.2.2 Tasks and Milestones

The tasks to be performed during FY 2001-2005 are listed in Table 3. (Details on the tasks, including the full set of objectives for each task, are provided in Section 3.) Table 3 also identifies those tasks either formally requested through a user need letter, or identified in formal correspondence as being required to resolve an existing HRA issue.

Figure 2 provides a summary schedule for these tasks and Table 4 lists the associated milestones for FY 2001-2002. Milestones for FY 2003-2005 will be documented in an update of this plan. Post-FY 2005 anticipated activities are not addressed in these tables, but are discussed in Section 5.

[7] The principal investigator for Task 9, "Reactor System Synergisms and HRA," has not yet been established.

Table 3. Human Reliability Analysis Research Program Tasks, FY 2001-2005

Task	Title	User Need [a]
1	HRA Data Collection and Analysis	RIRIP RSEER1-3
2	HRA Guidance Development	RIRIP RSMS5-1
3	HRA Quantification and Uncertainty	RIRIP RSEER1-3
4	Pressurized Thermal Shock HRA	RIRIP RSMS8-6 , NRR[b]
5	Fire HRA	RIRIP RSMS8-4
6	Steam Generator Tube Rupture HRA	NRR[c]
7	HRA for Aging Cable Systems	NRR[d]
8	HRA for Materials and Waste Applications	RIRIP WSMS1-2, NMSS [e]
9	Reactor Systems Synergisms and HRA	RIRIP RSEER 1-3
10	HRA for Upgraded and Advanced Control Rooms	RIRIP RSEER1-1
11	Latent Errors in HRA	RIRIP RSEER 1-3
12	HRA Extended Applications	RIRIP RSEER 1-3
13	Formalized Methods: Screening, Individual and Crew Modeling	RIRIP RSEER 1-3
A	HRA Research Planning	
B	HRA Results Communication	
C	General NRC HRA Technical Support	
D	Industry and International HRA Activities	

[a] See Table 1 for RIRIP implementation activity identifiers.

[b]Task 4 (Section 3.5) supports the development of a technical basis for a proposed rule change regarding pressurized thermal shock [14]. The overall effort responds to a user need letter [65].

[c]Task 6 (Section 3.7) supports an NRR request for an "Improvement of probabilistic safety assessment modeling of [severe accident-induced steam generator tube rupture (SGTR)] scenarios, including the effects of operator actions" [59].

[d]Task 7 (Section 3.8) supports an RES cable research program which is covered by the NRC's Task Action Plan on Environmental Qualification [60].

[e]Task 8 (Section 3.9) supports an NMSS request for the development and implementation of a dry cask storage probabilistic risk assessment [66, 67].

Table 4. HRA Research Program Milestones, FY 2001-2002 (Page 1 of 2)

Task	FY 2001-2002 Milestone	Date
1	Complete initial evaluation of HRA data needs	September 2001
1	Complete evaluation of a selected set of simulator data	September 2002
1	Complete evaluation of symptom-based procedures data	September 2002
2	Develop HRA research lessons to support risk-informed regulatory applications	September 2001
2	Develop initial guidance for HRA performance and review to support risk-informed regulatory applications	September 2002
3	Complete framework for HRA quantification	June 2001
3	Present proposed quantification process at WGRISK workshop	October 2001
3	Issue draft report on HRA quantification process	December 2001
3	Issue final report on quantification process	March 2002
4	Complete Oconee 1 PTS HRA	March 2001
4	Complete Beaver Valley 1 PTS HRA	July 2001
4	Complete Palisades PTS HRA review	September 2001
4	Complete Calvert Cliffs 1 PTS HRA review	October 2001
4	Issue report on PTS HRA approach and analyses	February 2002
4	Provide HRA input to PTS technical basis analysis report	July 2002
5	Complete HRA preparation for fire risk requantification study	December 2001
8	Complete HRA analysis for dry cask screening assessment	November 2001
8	Complete characterization of HRA needs for NMSS applications	September 2002
9	Complete identification and characterization of HRA issues for synergistic effects PRA	September 2002
10	Complete identification of key HRA issues for upgraded and advanced control rooms	September 2002
11	Complete review and evaluation of observed latent errors	September 2002
12	Develop improved methods and tools for ex-control room actions	September 2002
12	Develop improved methods and tools for LP&SD HRA	September 2002

Table 4. HRA Research Program Milestones, FY 2001-2002 (Page 2 of 2)

Task	FY 2001-2002 Milestone	Date
A	Complete draft of initial research program plan for review	December 2000
A	Finalize initial plan	June 2001
A	Issue report on HRA lessons learned (FY 2001-2002)	September 2002
B	Hold first workshop/seminar on key research results	December 2001
B	Complete assessment of information base need and feasibility	September 2002
D	Organize and host WGRISK workshop on errors of commission	May 2001
D	Participate in WGRISK workshop on HRA data	October 2001

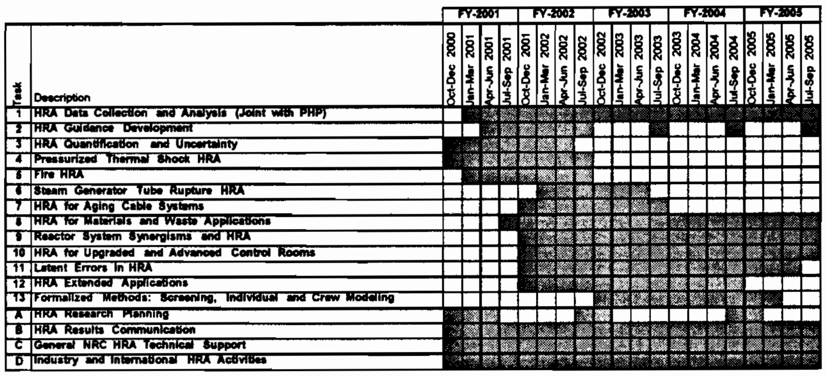

Figure 2. Human Reliability Analysis Research Program Summary Schedule

3. Technical Approach

3.1 Task Overview

The human reliability analysis research program technical tasks for FY 2001-2005 are listed in Table 3. These tasks are intended to support the achievement of the overall program objectives listed in Section 1.1; they have been selected based upon a consideration of NRC's HRA needs (discussed in Section 1.2.2), the current HRA state of the art (as discussed in Section 1.2.3), and parallel research programs in HRA and human factors (as discussed in Section 1.4). It should be noted that the tasks represent a significant broadening of activities beyond the NRC's recent HRA research work (which concentrated on developing ATHEANA).

The listed tasks can be organized according to their periodicity or the overall program objective supported.

Regarding periodicity, it can be seen that some of the tasks listed in Table 3 need to be repeated (or performed nearly continuously), while others will only need to be performed once. In particular, Support Tasks A through D are continuous or periodic activities. (For example, Task A - HRA Research Planning - will be performed every two years.) On the other hand, Tasks 1 through 13 are expected to performed only once (though they may be revisited as necessary).

Regarding the program objectives supported, Table 5 provides a mapping between the tasks and the program objectives. Note that most of the tasks support multiple objectives.

Sections 3.2 through 3.18 provide, for each task, the task objectives, a brief description of the technical approach, and key milestones for FY 2001-2002. (Additional milestones will be developed when the research program plan is updated, as discussed below.)

Table 5. HRA Research Program Objectives and Supporting Tasks

Objective	Supporting Tasks
Develop improved human reliability analysis (HRA) methods, tools (including guidance), and data needed to support NRC regulatory activities, including the broad implementation of risk-informed regulation	1-13, D
Develop HRA results and insights to support the development of technical bases for addressing identified or potential safety issues	4-10
Provide HRA support for the planning and execution of NRC programs and activities (e.g., the PHP) outside the immediate scope of the RIRIP	1, 11, C
Ensure effective communication of research results to end users	B, D
Ensure effective use of resources in satisfying the preceding objectives	A

3.2 Task 1 - HRA Data Collection and Analysis

One of the common criticisms of the current HRA state of the art (see Table 2) concerns the strength of the available data. Data are needed not only for the quantification of human failure event (HFE) probabilities, but also to support the HRA models (which, for example, postulate that certain factors are part of the error forcing context - EFC, and that there are specific relationships between the EFC elements and the HFE probability).

Regarding HFE quantification, additional discussion is needed because of the sparsity of actuarial data for direct estimation.

ATHEANA, like other advanced HRA methods, distinguishes between the occurrence of a particular EFC, and the occurrence of an unsafe act (UA), given the EFC. Thus, data are needed to assess both the likelihood of an EFC (given the PRA scenario in which the HFE is embedded), and the conditional likelihood of a UA (or set of UAs) leading to the HFE, given the EFC. However, in general, the information available from reports on operational events are not of sufficient quality to directly address these two issues. Furthermore, the risk significant HFEs for which probabilities are desired are typically events in accident scenarios which have not yet been observed. Thus, classical statistical methods cannot be used to develop the HFE probabilities. It can be seen that the use of subjective judgment is unavoidable, and that the Bayesian framework for estimation, which directly addresses situations where data are sparse, provides an appropriate way to proceed.

This is an important point from the standpoint of a data collection program, because the Bayesian framework accommodates different forms of "evidence," including indirect observations, model predictions, and expert judgment, as well as actuarial data. The precise formalisms for employing these data in an HRA context are not yet developed (see Task 3). However, regardless of how the formalisms are developed, it is important to recognize that a wide variety of information can be used in a Bayesian analysis. Therefore, the data collection activity need not focus on one particular source of information (e.g., operational events, simulator experiments, operator requalification tests) over another.

It is also important to recognize that the quantity and quality of HRA data has been a concern since the time of WASH-1400 (see, for example [21]), and that this concern is not expected to be resolved quickly (given the issues mentioned above). Therefore, it can be expected that substantial improvements will require a sustained, long-term, and potentially resource-intensive effort.

3.2.1 Task 1 - Objectives

The objectives of this task are to

- Define the qualitative and quantitative data needs of HRA
- Develop long-term working relationships with key HRA and human factors research programs capable of generating new data
- Collect and analyze data to support HRA model development and quantification

3.2.2 Task 1 - Description of approach

This task is to be performed as a joint activity with the NRC Program on Human Performance in Nuclear Power Plant Safety [1]. This will increase the degree of human factors input on the phenomenological issues being addressed and will support the development of a strong connection with ongoing experimental research programs.

The first step in the task is to define the data needs for HRA, and especially HRA quantification. This requires that reasonable progress be made on Task 3 ("HRA Quantification and Uncertainty") before substantial work can be started. Subsequently, activities will be performed to: identify potential cooperating programs in addition to those with which NRC already has cooperative agreements (e.g., Halden), communicate these needs to cooperating programs, analyze existing data from these programs, specify additional data collection/generation activities, and analyze the data from these additional activities.

In parallel with these activities, which are largely focused on experimental programs, efforts will be made to collect information from programs compiling human-related operational experience data and related sources. Regarding the latter, efforts will be made to collect information on the technical bases supporting the industry's current symptom-based emergency operating procedures. This information is likely to include the scenario variations considered by the procedure developers, and the bases for addressing or not addressing these variations in the procedures.

3.2.3 Task 1 - FY 2001-2002 Milestones

• Complete initial evaluation of HRA data needs	September 2001
• Complete evaluation of a selected set of simulator data	September 2002
• Complete evaluation of symptom-based procedures data	September 2002

3.3 Task 2 - HRA Guidance Development

As indicated in Section 1.2.4, most of the recent NRC HRA research work has focused on the development of improved HRA methods (especially ATHEANA). While these improved methods are proving useful in some risk-informed applications, it is recognized that not all risk informed applications require such detailed analyses. A key question that must be faced by both HRA analysts and HRA reviewers is under what conditions are the improved methods needed, and under what conditions are the older (and widely used) methods sufficient. Another question is how can the selected method best be applied to the problem at hand.

3.3.1 Task 2 - Objective

The objectives of this task are to:

- Develop lessons learned from past HRA research activities to support risk-informed regulatory applications.
- Develop guidance for HRA analysts and reviewers to support risk-informed regulatory applications.

3.3.2 Task 2 - Description of approach

The first step under this task will be to develop lessons learned from recent HRA research activities (including the ATHEANA development process) to support ongoing and anticipated risk-informed regulatory applications. This is expected to involve: a) an assessment of the type, quality, and characteristics of HRA information needed to support such regulatory applications; b) a review and evaluation of a selected number of "first generation" HRA methods (e.g., THERP, ASP HRA, HCR/ORE, SLIM-MAUD) from the perspective of these information needs; c) the characterization of the information provided by recent HRA research relative to the needs and gaps identified by the preceding two activities; and d) the identification of gaps remaining to be addressed.

The second step will be to develop guidance for performing and reviewing HRA analyses in support of risk-informed regulatory applications. This work will extend the results of the lessons learned activity above. It will also employ the results of ATHEANA applications completed under other tasks in the HRA research program (e.g., see Tasks 4-7). An initial set of guidance will be developed in FY 2002. It is anticipated that this guidance will be periodically updated as additional results from the HRA research program are developed.

3.3.3 Task 2 - FY 2001-2002 Milestones

• Develop HRA research lessons to support risk-informed regulatory applications	September 2001
• Develop initial guidance for HRA performance and review to support risk-informed regulatory applications	September 2002

3.4 Task 3 - HRA Quantification and Uncertainty

In its 1999 review of ATHEANA as documented in NUREG-1624 Rev. 1 [24], the ACRS commented that the quantitative portion of the ATHEANA methodology "still needs significant development" [53]. One of the issues raised by the ACRS concerned whether the ATHEANA process for using expert judgment builds upon the body of work that has been developed on expert elicitation and the utilization of expert opinions (e.g., see NUREG/CR-6372 [68]).

A second, and somewhat related issue has been identified during the course of the PTS HRA. In the PTS analysis, as indicated in Section 3.5, efforts are being made to distinguish between aleatory and epistemic uncertainties. To make this distinction, the meaning of the model parameters has to be clear. For example, in the case of HRA, the question is if the HFE probability, which is taken to be a measure of aleatory uncertainty, includes such things as variations in time of day at which the accident initiator occurs, or if these variations are to be included in the uncertainty about the HFE probability. It is not clear that the issue has been seriously addressed in the HRA literature. This is an important point because, when eliciting expert judgments, it is necessary to be clear and consistent about the quantity being estimated.

A third quantification issue follows from the discussion in Section 3.2: the information available to support quantification may be in a variety of forms (e.g., operational events, model predictions, results of simulator experiments, expert judgments, tabulated generic error probabilities). It is widely recognized that Bayes' Theorem provides an appropriate formalism for dealing with these different forms of evidence. However, the specific implementations of Bayes' Theorem to address certain forms of evidence have not been developed.

Based upon these three observations, it is apparent that work is needed on the fundamental issue of HFE quantification (which includes the treatment of uncertainties). This task is aimed at establishing an approach for dealing with the problem; it will lay the groundwork for the data collection activities pursued under Task 1.

3.4.1 Task 3 - Objective

The objective of this task is to develop and perform some preliminary tests of a formal approach to HFE quantification which:

- addresses uncertainties in a manner consistent with the PTS PRA philosophy [69],
- makes appropriate use of the various forms of available information, and
- appropriately accounts for potential biases in situations involving expert elicitation.

3.4.2 Task 3 - Description of approach

The first step in the task will be to develop an updated framework for HRA quantification. It is anticipated that this framework will draw explicit relationships between EFCs, UAs, and HFEs; will explicitly identify the contextual elements that need to be addressed as part of a given EFC; will explicitly categorize uncertainties in the various contextual elements as being aleatory or epistemic; will identify and categorize uncertainties in the estimation of the conditional probability of a UA (or set of UAs), given a particular EFC; and will indicate the general quantification process to be followed.

If a formal Bayesian approach is to be used, the second step will be to characterize the forms of evidence likely to be available to support quantification, and to develop appropriate likelihood functions for use in a Bayesian estimation process.

The third step, which can be conducted in parallel with the second step, will be to review relevant literature on elicitation processes, considering the quantification needs identified by the quantification framework, and then to adopt (or adapt) a process suitable for use in future HRA analyses.

The fourth step will be to combine the results of the preceding steps into a unified quantification process, to test this process through a demonstration problem, and to develop recommendations for additional research.

3.4.3 Task 3 - FY 2001-2002 Milestones

This task was initiated in FY 2000. It is scheduled for completion in FY 2002.

• Complete framework for HRA quantification	June 2001
• Present proposed quantification process at WGRISK workshop	October 2001

- Issue draft report on HRA quantification process — December 2001
- Issue final report on quantification process — March 2002

3.5 Task 4 - Pressurized Thermal Shock HRA

The ATHEANA application to pressurized thermal shock (PTS) has been mentioned in Sections 1.2.2 and 1.2.4. This work was initiated in FY 2000 and is scheduled for completion in early FY 2002.

3.5.1 Task 4 - Objective

The objective of this task is to provide HRA support to the PTS PRA effort.

3.5.2 Task 4 - Description of approach

This task involves the application of ATHEANA towards the analysis of two pressurized water reactors (PWRs) - Oconee 1 and Beaver Valley 1, and the review of licensee-performed PRAs for two additional PWRs - Palisades and Calvert Cliffs 1. The ATHEANA application includes the development of a generic PTS functional event tree, the identification of potential HFEs, visits to the plant to talk with plant operators and trainers, modeling of the HFEs, and quantification of the HFEs. Consistent with the rest of the PTS PRA, the quantification process accounts for aleatory (sometimes called "random") and epistemic (sometimes called "state of knowledge") uncertainties. (See Apostolakis [70, 71] for the treatment of uncertainty in PRAs; Ref. 69 is a white paper on the treatment of uncertainty in the PTS effort.)

3.5.3 Task 4 - FY 2001-2002 Milestones

This task was initiated in FY 2000. It is scheduled for completion in FY 2002.

- Complete Oconee 1 PTS HRA — March 2001
- Complete Beaver Valley 1 PTS HRA — July 2001
- Complete Palisades PTS HRA review — September 2001
- Complete Calvert Cliffs 1 PTS HRA review — October 2001
- Issue report on PTS HRA approach and analyses — February 2002
- Provide HRA input to PTS technical basis analysis report — July 2002

3.6 Task 5 - Fire HRA

Current fire risk assessment (FRA) treatment of the response of plant operations staff to fire events is relatively crude. Some FRAs increase human error probabilities to account for the additional "stress" induced by the fire and some do not take credit for ex-main control room actions in the affected fire area (due to heat and smoke). However, these adjustments may not adequately address such plant-specific issues as the role of fire brigade members in accident response or the complexity of fire response procedures, nor are they universally agreed upon. Moreover, they are quite judgmental; there currently is no strong technical basis for the magnitude (or even direction) of the adjustments.

Another concern is that certain elements of context that may arise due to the effects of fire (e.g., fire-induced faulty instrumentation readings, spurious equipment actuations, progressive loss of equipment over time) on operator situation assessment and decision making, nor do they address incorrect operator actions stemming from incorrect decisions.

In principle, ATHEANA provides an appropriate approach for addressing these issues of task allocation, procedure complexity, and fire-induced EFCs. This task involves an application of ATHEANA to a number of plants. This application will support the "fire risk requantification study" to be performed under the fire risk research program [58]. It is expected that this application will be valuable to the area of FRA, as well as a useful and demanding test of ATHEANA. It should be noted that a highly preliminary application of ATHEANA was presented to the ACRS late in 1999. The work performed under this task will represent a significant extension of that earlier work.

3.6.1 Task 5 - Objective

The objective of this task is to support the fire risk requantification study through:

- investigating the possibility of developing an improved technical basis for incorporating fire-induced environmental effects in HRA;
- developing any necessary HRA methods for addressing EFCs associated with fire effects (e.g., environmental effects, loss of instrumentation, spurious actuations, time-dependent equipment losses);
- applying the fire HRA approach towards the analysis of the plants included in the requantification study;
- developing insights regarding the risk associated with the impact of fires and fire-induced failures on operator situation assessment, decision making, and associated actions; and
- developing insights regarding fire HRA methods.

3.6.2 Task 5 - Description of approach

This task will be performed in two steps. The first step represents a preparation for the requantification study. The preparations will include a review of the fire safety literature for information on environmental effects, a review of the need for improved HRA methods (if any) to account for other fire-induced EFCs, and the development and implementation of these modifications. This step will build upon the work underlying the preliminary ATHEANA application to fire reviewed by the ACRS, and upon the results of a review being performed for the Individual Plant Examination of External Events (IPEEE) program [72].

The second step involves the application of the HRA approach to plants selected for analysis as part of the requantification study. As indicated in the Fire Risk Research Program Plan [58], the requantification study will require close cooperation between NRC and industry. It is hoped that the FRAs to be updated will represent a range of plant, plant ages, and FRA types (e.g., vulnerability analyses vs. detailed FRAs). The potential for applying the updated FRAs to evaluate specific issues at a plant will also be a consideration in the selection of plants to be analyzed. The precise plants to be analyzed will be determined following ongoing discussions with the industry regarding the extent and form of cooperation.

3.6.3 Task 5 - FY 2001-2002 Milestones

Preliminary work on this task was initiated in FY 2000. The task is scheduled for completion in early FY 2002. Note that the full set of milestones are not yet developed, as their definition involves upcoming interactions with the Electric Power Research Institute (EPRI), which, under current plans, will be coordinating the industry effort in the requantification study.

- Complete HRA preparation for fire risk requantification study December 2001

3.7 Task 6 - Steam Generator Tube Rupture HRA

In FY 2000, the Office of Nuclear Reactor Regulation (NRR) requested that the Office of Nuclear Regulatory Research (RES) perform a number of confirmatory research activities addressing steam generator tube integrity during postulated severe accidents in pressurized water reactors [59]. One of the desired outcomes is an "Improvement of probabilistic safety assessment modeling of [severe accident-induced steam generator tube rupture (SGTR)] scenarios, including the effects of operator actions." This task will support a broader PRA effort addressing this user need.

3.7.1 Task 6 - Objective

The objective of this task is to develop an improved HRA approach for post-severe accident SGTR scenarios.

3.7.2 Task 6 - Description of approach

The overall approach used will be similar to that used for the PTS project being supported by Task 4; it will involve an integrated engineering analysis of the scenario with inputs from PRA, thermal hydraulics, and structural analysis teams. The PRA portion of the analysis will build upon the accident progression event trees (APETs) developed in an earlier study [73]. The HRA analysis is expected to employ the ATHEANA method to determine if all potentially significant HFEs have been identified, to identify significant EFCs, and to quantify the likelihood of the HFEs.

3.7.3 Task 6 - FY 2001-2002 Milestones

This task will be initiated in FY 2002. It is currently anticipated that the work will be completed in FY 2003. A detailed schedule is being developed.

3.8 Task 7 - HRA for Aging Cable Systems

Recent RES-sponsored environmental qualification tests involving the exposure of thermally aged I&C cables to harsh environments (e.g., those caused by large loss of coolant accidents) have shown that certain cable types can fail catastrophically and others can experience performance anomalies under design basis accident conditions [74]. It can be inferred that the conditional failure probability of these cables (given the environment) may be sufficiently high to warrant their explicit treatment in PRAs. (Note that current PRAs assume that the cables are sufficiently reliable that they need not be modeled, except in the case of fire risk assessments.)

Such a treatment would need to address, among other things, the possibility of spurious indications and actuations (as well as loss of function) and the consequent effect on the plant operators.

In FY-2002, RES will initiate an activity to evaluate the risk associated with cable system aging and failure [75]. This activity is currently expected to address the frequency-magnitude relationship for the post-accident environment for various initiators, variability in the aging of actual cables, cable fragilities, cable function and separation, and operator response to cable failures. Task 7 will provide HRA support to this activity. Note that Task 9 will address more general aspects of aging.

3.8.1 Task 7 - Objectives

The objective of this task is to provide HRA support to the NRC's aged cable risk assessment activity.

3.8.2 Task 7 - Description of approach

The plan for the aged cable risk assessment is under development. The approach used in this task will be developed to be consistent in scope and detail with the overall risk assessment. It is currently expected that the ATHEANA approach will be useful, as its focus on scenario context provides a means to address the potential confusion arising from the various cable failures that can occur. It is also expected that the HRA will require a review of events in which operators had to deal with significant losses of instrumentation (e.g., the Rancho Seco "light bulb" incident in 1978).

3.8.3 Task 7 - FY 2001-2002 Milestones

This task will be initiated in FY 2002. It is currently anticipated that the work will be completed in FY 2003. A detailed schedule is being developed.

3.9 Task 8 - HRA for Materials and Waste Applications

As indicated in the RIRIP [2], NMSS is currently developing a risk-informed regulation framework to cover applications involving the NRC's nuclear materials safety and nuclear waste safety arenas. This development activity involves, among other things, the performance of case studies on specific topics.

In 1997, NMSS requested assistance from RES in the development and implementation of a PRA of dry cask storage facilities [66]. The user need has since evolved into a request involving PRA and probabilistic fracture mechanics [67]. An important part of the PRA is the reliability of the loading, sealing, and onsite transportation of the casks. Therefore, the project requires the identification, analysis, and quantification of human error probabilities. Task 8 provides HRA support to the dry cask PRA. It will also provide HRA support to other materials and waste applications as needs arise.

3.9.1 Task 8 - Objectives

The objectives of this task are to

- Provide HRA support to the dry cask PRA
- Provide HRA support to other nuclear materials and waste risk assessment activities, as needed

3.9.2 Task 8 - Description of approach

Regarding the dry cask PRA support, the first step of this task is to develop a preliminary understanding of the problem. This will be obtained through document review, a plant visit, and interactions with the licensee.

The second step of this task is to perform a screening analysis to identify the potentially most risk significant human failure events. The HRA method to be used in this step will be selected by discussions among the NRC staff and contractors. Additional information from existing studies, plant procedures, and plant personnel will be used as available. An additional site visit will be arranged to obtain a clear understanding of dry cask storage system operations.

The third step of this task is to develop detailed HRA models (with quantification) for use in the PRA. The inputs will appropriately account for uncertainty and work will be required to integrate the results into the overall risk analysis.

Regarding more general nuclear materials and waste applications, it is recognized that the wide variety of facilities and processes of concern to NMSS will likely require the development of a variety of PRA (and HRA) methods and tools. RES plans to initiate work in FY 2002 to, in concert with NMSS, characterize the PRA methods, tools, and data needs for these facilities and processes; Task 8 will provide HRA support to this characterization effort.

3.9.3 Task 8 - FY 2001-2002 Milestones

The dry cask PRA effort supported by this task is scheduled for completion in FY 2003. The screening level assessment phase of the PRA is currently scheduled for completion in November 2001.

- Complete HRA analysis for dry cask screening assessment — November 2001
- Complete characterization of HRA needs for NMSS applications — September 2002

3.10 <u>Task 9 - Reactor Systems Synergisms and HRA</u>

As pointed out by the ACRS in its recent review of NRC's reactor safety research activities [54], a number of changes in the U.S. nuclear power industry are either underway or are being considered. These changes include plant aging, extended fuel burnups, and licensing actions to allow power uprates. Not only can each of these changes individually affect plant risk levels and profiles, they may have a collective, synergistic effect on risk. However, the risk and phenomenological models needed to assess these risk impacts are not yet well developed.

In FY 2002, RES will initiate a research activity aimed at developing needed methods, tools, and data to assess the collective risk impact of these and other major changes occurring within the industry. Task 9 will provide HRA support to this activity.

3.10.1 Task 9 - Objectives

The objective of this task is to provide HRA support for the development of risk assessment methods, tools, and data needed to assess the collective impact of major changes in current U.S. NPPs.

3.10.2 Task 9 - Description of approach

Work on developing PRA methods, tools, and data will be initiated in FY 2002. As an early part of the this work, a plan for identifying, prioritizing, and addressing key issues will be developed. It is expected that the plan development process will include a review and characterization of major changes, a review of standard NPP PRA assumptions, and an identification of areas where the PRA models may need to be significantly revised. The model review will consider model structure (e.g., success criteria, boundary conditions, cascading effects) as well as parameter values (e.g., failure probabilities, mission times). HRA issues will be considered as part of this overall review effort. The issues are likely to include consideration of potential changes to the time available to operators to perform actions, and of complicating factors that may arise during accidents (e.g., see the discussion on cable aging under Task 7). Other changes that may be considered involve changes in human-related areas (e.g., changes in plant staff size and demographics).

The results of the overall review effort will indicate if existing PRA (and HRA) methods, tools, and data require major or minor changes. It should be noted that RES has sponsored a feasibility study looking at the integration of physical models for key aging mechanisms into conventional PRA structures [76]. The methods and tools developed in that work, together with the results of past studies on PRAs for aging plants, are expected to provide a useful starting point for the treatment of synergistic effects.

3.10.3 Task 9 - FY 2001-2002 Milestones

- Complete identification and characterization of HRA issues for synergistic effects PRA — September 2002

3.11 Task 10 - HRA for Upgraded and Advanced Control Rooms

The U.S. Department of Energy has a number of studies underway developing designs for "Generation IV" reactors [77]. As noted by a number of human factors researchers (e.g., see O'Hara [78]), these advanced reactors not only have different operator/plant interfaces (e.g., involving operator navigation through multiple video displays), they also are intended to have fundamentally different roles for the operators in responding to accidents. (For example, the changes may result in the operators' role becoming more one of supervisory control.) These differing interfaces and operator roles are likely to require improvements in current HRA methods and tools in order to support risk-informed design reviews and certifications.

It should also be noted that currently operating plants are gradually upgrading their control rooms by replacing their analog I&C systems with advanced digital systems [1, 62]. These changes are also likely to require improvements in current HRA methods and tools to support risk-informed regulatory applications.

3.11.1 Task 10 - Objective

The objectives of this task are to:

- Identify key issues associated with HRA for upgraded and advanced control rooms
- Develop guidance for reviewers of HRAs involving upgraded and advanced control rooms
- Develop improved HRA methods and tools to support PRAs for upgraded and advanced control rooms

3.11.2 Task 10 - Description of approach

This task, which will be initiated in FY 2002, will involve a review of current trends in control room upgrades, of current proposals for advanced reactors, and of previous studies on the risk implications of advanced control room technology (e.g., see [79]). Based on these reviews, key HRA issues will be identified and the ability of existing methods (including ATHEANA) to address these issues (in light of the information available at a design stage) will be evaluated. Guidance for reviewers of HRAs for upgraded and advanced control rooms will be developed. It is expected that improved HRA methods will be also be deveioped and demonstrated in a limited test. It is anticipated that these methods will address: a) interactions between the operators, digital protection and control systems, and the plant; and b) any changes in the roles of operators (as compared with current approaches).

3.11.3 Task 10 - FY 2001-2002 Milestones

This task will be initiated in FY 2002. Given the time scale of advanced reactor developments, it is expected to continue beyond FY 2005.

- Complete identification of key HRA issues for upgraded and advanced control rooms — September 2002

3.12 Task 11 - Latent Errors in HRA

As indicated in Section 1.4.1, the PHP has recently completed a study which suggests that latent errors, i.e., errors which occur prior to an initiating event but which are not revealed until some later point in time due to a triggering event (e.g., an accident scenario), may have more impact on plant risk than previously recognized, and that they may require improved treatment in HRAs [23].

Current PRA treatments of latent errors are varied. Some studies address these errors explicitly (as separate contributors to component, train, or system unavailability), while others treat them implicitly (through the failure probabilities assigned to the hardware). The modeling choice is generally dependent on the form of the data used to estimate unavailabilities (e.g., whether failures due to human error are distinguished from other failures).

A number of currently available HRA methods, e.g., THERP, appear to be capable of dealing with individual latent errors and their effects [42]. However, these methods do not deal with a potentially significant issue: systematic dependencies between latent errors, e.g., due to such factors as common work processes [28]. This issue may be important because, if the dependencies are significant, their cumulative impact on multiple HFEs and multiple sequences may alter a plant's risk profile.

3.12.1 Task 11 - Objectives

The objectives of this task are to:

- Develop an improved understanding of latent errors observed during operational events
- Determine where HRA improvements are needed to improve the treatment of latent errors
- Develop improved HRA methods to identify, model, and quantify latent errors

3.12.2 Task 11 - Description of approach

The first step of this task will involve the review and evaluation of the latent errors identified in the PHP study. The evaluation shall consider the structure of current PRA component failure databases (to determine how the observed errors are addressed), and of current HRA methods (to determine the extent to which they can be used to model these errors. The evaluation is expected to result in recommendations regarding how current HRA methods can be best used, as well as regarding where improvements are needed.

The second step, which can be performed in parallel with the first, will involve an analysis of operational data for failures which were or may have been caused by latent errors, to determine if there is evidence for dependencies between these failures. This analysis will consider but will not be limited to common cause failure data, as it will consider events involving different components, different systems, and at different times.

The third step will develop improved methods for treating latent errors. The thrust of this work will naturally depend on the results of the preceding tasks. However, it is currently anticipated that the issue of dependencies will need to be addressed, and that organizational considerations (e.g., work processes) will need to be treated in order to address these dependencies. It is also anticipated that results from ongoing international research efforts in this area (e.g., including the work of the COOPRA working group on organizational influences on risk) will be needed for this step.

The final step will involve an application of the improved methods. The application will revisit the conclusions of the PHP study, and will provide insights regarding the risk significance of latent errors, as well as insights regarding the usability of the improved methods.

3.12.3 Task 11 - FY 2001-2002 Milestones

This task will be initiated in FY 2002. Due to the complexity of some of the issues that will likely need to be addressed, and the potential need to use the results of cooperative international research efforts, it is scheduled for completion in FY 2005.

- Complete review and evaluation of observed latent errors September 2002

3.13 Task 12 - HRA Extended Applications

To date, much of the emphasis of HRA methods development activities worldwide has been on the treatment of HFEs associated with control room actions taken to prevent core damage within a few hours after an initiating event. As many of these methods are based on a general understanding of human behavior and the sources of error, they should be applicable when dealing with other situations (e.g., post-initiator actions outside of the control room, long term recovery actions, actions taken during severe accidents, and actions during low power and shutdown operation). However, these other situations provide challenges (e.g., regarding the treatment of teamwork, the interactions of multiple teams, the availability and quality of indications, the impact of the use of guidelines rather than specific procedures, the extended time available for actions) whose practical treatment may require additional developments.

3.13.1 Task 12 - Objectives

The objectives of this task are to

- Evaluate existing HRA methods
- Develop, as needed, improved HRA methods and tools

for the following situations

- post-initiator actions outside of the control room
- low power and shutdown (LP&SD) operation
- long term recovery
- severe accidents

3.13.2 Task 12 - Description of approach

For each situation, this task will identify the key features that need to be addressed, and will evaluate existing HRA methods (including both widely used methods as well as recently developed methods) with respect to their ability to practically address these features. Areas for improvement will be identified and improved methods or tools (including guidance) developed, as needed.

The first two areas to be addressed are ex-control room actions and LP&SD operation. Work will be initiated on these in FY 2002. Work on severe accidents HRA will be initiated in FY 2003, and work on long term recovery actions will be initiated in FY 2004. It is expected that the work on severe accidents HRA will benefit from the (more limited) analyses performed to support severe accident-induced SGTR model development (see Task 6, Section 3.7).

3.13.3 Task 12 - FY 2001-2002 Milestones

This task will be initiated in FY 2002. It is scheduled for completion in FY 2004.

- Develop improved methods and tools for ex-control room actions September 2002
- Develop improved methods and tools for LP&SD HRA September 2002

3.14 Task 13 - Formalized Methods: Screening, Individual and Crew Modeling

The ACRS review of ATHEANA [53] and the results of previous peer reviews have identified a number of specific areas where ATHEANA (as documented in NUREG-1624, Rev. 1 [24]) can be improved. One area, the process for quantifying HFE probabilities, is being addressed by Task 3. This task addresses other areas identified, including the lack of a formal screening method, the lack of an explicit model of cognition for individual crew members (e.g., to provide more formal links between error forcing contexts, potential error mechanisms, and unsafe acts), and the lack of an explicit model for addressing interactions within a crew. Regarding the latter two issues, it is expected that the development of explicit models will improve the accuracy of HRA predictions, reduce the reliance of the analysis results on the judgment of the particular analysis team involved, and will provide an improved means for incorporating experimental data into the analysis (e.g., to test implicit hypotheses built into the analysis, to the assess the strength of specific model factors).

This task is scheduled to start in FY 2003, in order to take advantage of the ongoing tasks (including the ATHEANA applications to various situations), and of anticipated input from ongoing cooperative research activities (e.g., work being conducted by the risk working group of OECD/CSNI).

3.14.1 Task 13 - Objectives

The objectives of this task are to:

- Develop a screening method for use in context-based HRA methods
- Develop and test explicit models for addressing individual cognition and team issues for use in HRA

3.14.2 Task 13 - Description of approach

Regarding the development of a screening method, the previous ATHEANA applications for PTS, fire, and SGTR will be reviewed. The purpose of the review will be to characterize how screening was done in those previous analyses, and to identify areas for improvements in the process. Based upon the results of this review, and upon an understanding of the information available at different stages of an HRA analysis, a more formal screening method will then be developed. This method will be tested in a limited application.

Regarding the explicit modeling of cognitive and team issues, it is recognized that ATHEANA has been developed to support a conventional (static) PRA model structure, whereas a detailed treatment of operator cognition and team effects may require a modeling approach that explicitly accounts for system dynamics. It is also recognized that there are a number of research activities looking at these effects (including the dynamic PRA work being performed at the University of Maryland [64]). In this task, the results of these ongoing activities will be reviewed to determine how their results can be used within a context-based approach to HRA. The results of this review will be used to propose an improved HRA approach. This proposed approach will be tested, likely using data obtained from Task 1.

3.14.3 Task 13 - FY 2001-2002 Milestones

This task will be initiated in FY 2003. It is scheduled for completion in FY 2005.

3.15 Support Task A - HRA Research Planning

3.15.1 Task A - Objective

The objective of this task is to ensure that the HRA Research Program appropriately reflects current research results and progress, and current NRC priorities. The lessons learned report will be used to identify potential changes to the HRA Research Program Plan.

3.15.2 Task A - Description of approach

This report, which documents the HRA Research Program Plan, represents the result of the first step in this task. It is expected that the report will be updated every two years, based upon a review of current NRC needs and of current HRA research results.

3.15.3 Task A - FY 2001-2002 Milestones

- Complete draft of initial research program plan for review — December 2000
- Finalize initial plan — June 2001
- Issue report on HRA lessons learned (FY 2001-2002) — September 2002

3.16 Support Task B - HRA Results Communication

3.16.1 Task B - Objective

The objective of this task is to ensure that the HRA Research Program results are efficiently communicated to users, cooperative research partners, and to interested members of the public.

3.16.2 Task B - Description of approach

In addition to the standard mechanisms for disseminating research results (e.g., publication of reports, conference papers, and journal papers), a number of additional mechanisms will be investigated and employed if judged efficient. The additional mechanisms considered will include:

- Training. RES staff will work with NRC staff and contractors responsible for the HRA training course to ensure that the research results are incorporated into the training, as appropriate.

- Workshops/Seminars. The possibility of annual workshops (or seminars) to discuss the latest results of the HRA Research Program will be investigated. These workshop/seminars will be coordinated with other HRA meetings (e.g., professional conferences, WGRISK or COOPRA meetings).

- Information Base. Work will be initiated in FY 2002 to identify alternative methods (e.g., an information base with tailored search tools posted on the NRC web site or provided on a CD/ROM) to increase the availability of key research results. If there is sufficient demand, work on one or methods may be initiated in FY 2003.

3.16.3 Task B - FY 2001-2002 Milestones

The following milestones are associated with the communication of results of the overall HRA Research Program. Milestones associated with papers and reports for individual tasks are covered under those tasks.

- Hold first workshop/seminar on key research results — December 2001
- Complete assessment of information base need and feasibility — September 2002

3.17 Support Task C - General NRC HRA Technical Support

This task addresses requests for HRA support (e.g., in developing plans to treat an emerging issue) not included in the scope of the other tasks in the HRA Research Program.

3.17.1 Task C - Objective

The objective of this task is to provide needed HRA support for activities not covered under an existing task in the HRA Research Program.

3.17.2 Task C - Description of approach

Three principal activities are anticipated:

- Provide HRA support for addressing new issues. This covers support for the performance of initial, scoping-level assessments of the potential risk significance of these issues, and support for the development of initial project plans aimed at developing a more accurate assessment of risk significance.

- Develop responses to reviews and requests for information from oversight committees and the Commission.

- Provide initial support to the PHP in the development of Commission policies regarding human performance issues (e.g., fatigue).

3.17.3 Task C - FY 2001-2002 Milestones

This activity is ongoing; milestones will be established as issues are identified.

3.18 Support Task D - Industry and International HRA Activities

Recognizing that NRC's resources for HRA research and development are limited, and that there are a number of significant international HRA R&D efforts underway, there is a strong incentive for NRC to try to benefit from these international efforts. In order to accomplish this, NRC needs to actively participate in ongoing international cooperative activities, especially those associated with the risk working group (WGRISK) of OECD/CSNI, and with COOPRA (see Section 1.4).

WGRISK is currently finishing a task looking at errors of commission (see [25]), has held one HRA workshop, and plans to hold another in the near future. The first workshop discussed errors of commission, and was held in the Washington, DC area in May, 2001 with support provided by the HRA Research Program. The second will be held in Munich, Germany in October, 2001. This second workshop will discuss HRA data, and can be viewed as an initial step towards the development of increased collaboration.

COOPRA has a working group interested in the effect of organizational influences on risk. The results of this working group's activities are expected to provide useful information to Task 11 ("Latent Errors in HRA") and Task 13 ("Formalized Methods: Screening, Individual and Crew Modeling").

Regarding industry efforts, as discussed at the May 2001 WGRISK workshop on errors of commission, EPRI has initiated an HRA/PRA Tools Users Group aimed at: a) helping industry converge on common HRA methods, and b) enabling different analysts to obtain comparable results for similar situations. The users group is developing an "HRA Calculator," is considering the use of a 2nd generation HRA method developed by EdF (MERMOS [27]), and is considering the quantification of the impact of organizational factors on safety.

3.18.1 Task D - Objectives

The objectives of this task are to:

- Support the exchange of HRA research information
- Develop targeted cooperative HRA research activities to support NRC's HRA Research Program objectives

3.18.2 Task D - Description of approach

The primary activities under this task will be to support both WGRISK and COOPRA in their HRA activities. NRC will take the lead in organizing the WGRISK errors of commission workshop, will participate in the organization of the HRA data workshop, and will take an active role in future WGRISK HRA activities. NRC will also participate as a member of the COOPRA working group on organizational influences. Interactions with the EPRI HRA activity may be pursued under the EPRI/NRC Memorandum of Understanding on Cooperative Nuclear Safety Research [80].

In addition to these formal activities, the possibility of alternative, less formal interactions with selected HRA R&D programs on specific topics will be investigated. In particular, depending upon the results of ongoing discussions following the May, 2001 WGRISK errors of commission workshop, the development of a white paper providing a detailed review of the HRA state of the art will be considered. If performed, this review will consider currently available HRA methods from a variety of viewpoints (e.g., the phenomenology addressed, the analysis process employed, the data required), and will build upon previous HRA methods reviews. The purpose of this review will be to provide a framework for initiating discussions with other researchers on cooperative efforts. The decision to develop the paper will depend upon the need for such a paper to develop fruitful cooperative efforts.

3.18.3 Task D - FY 2001-2002 Milestones

- Organize and host WGRISK workshop on errors of commission — May 2001
- Participate in WGRISK workshop on HRA data — October 2001

4. Communication Plan

In order to achieve the overall program objectives listed in Section 1, the human reliability analysis research program includes a number of activities supplementing the technical tasks identified in Section 3. These additional activities, described in this section, will ensure that the research results are readily available to the NRC staff and industry for effective use in regulatory activities. [8] Furthermore, they will ensure that interested members of the HRA community (including international researchers) and the general public can stay abreast of the research program and access results as desired.

4.1 Communication of Research Planning

The following activities will be performed to inform stakeholders regarding the human reliability analysis research plan objectives, activities, and schedule, and to gain feedback from these stakeholders.

- The research plan for FY 2001-2005 is documented in this report. The report may be published as a NUREG document; it will be publicly available through the Public Document Room, ADAMS, and various locations on the NRC web site.

- Efforts will be made to present the plan and subsequent updates to the plan at a number of public meetings, including

  meetings of the ACRS Subcommittee on Probabilistic Risk Assessment; and
  technical conferences (e.g., the RES Nuclear Safety Research Conference, to be held in October, 2001).

4.2 Communication of Research Results

As discussed under Support Task B (see Section 3.16), the following activities have been or will be performed to inform stakeholders regarding the results of the human reliability analysis research program.

- Research reports, journal and conference papers, and other publications (e.g., key white papers) will be distributed using standard channels (e.g., mailing lists, responses to requests, placement in the Public Document Room, posting on the NRC web site, posting in ADAMS).

- Summary papers of key results will be submitted to archival journals, and summary presentations will be made at a variety of meetings, including:

  meetings of the ACRS Subcommittee on Probabilistic Risk Assessment;
  scientific conferences and workshops; and
  meetings of international cooperative working groups.

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[8] See Section 1.2.2 for a description of these activities.

- A periodically updated list of currently available products from the human reliability analysis research program will be placed on the NRC web site.

- The PRA Steering Committee will be kept abreast of the research program's status and the availability of results.

- RES staff will work with NRC staff and contractors responsible for the human reliability analysis training course to ensure that the research results are incorporated into the training, as appropriate.

- A public workshop will be held in the early part of FY 2002. The objectives of the workshop will be to summarize the current results of the research program and to discuss the implications of these results for ongoing activities.

- Ongoing HRA standards development activities will be monitored and areas where the HRA research results may be useful will be provided to the appropriate standard writing committees.

- Work will be initiated in FY 2002 to identify alternative methods (e.g., an information base with tailored search tools posted on the NRC web site or provided on a CD/ROM) to increase the availability of key research results. If there is sufficient demand, work on one or methods may be initiated in FY 2003.

5. Potential Activities Beyond FY 2005

Although considerable progress in the development and deployment of HRA methods is expected by the end of FY 2005, the challenges of predicting human performance and the needs of risk-informed regulation are considerable. Therefore, it can be expected that HRA research, development, and applications activities will be needed beyond FY 2005. In particular, on the research and development side, it can be expected that additional work on collecting and analyzing data; on validating HRA models; and on increasing the ability of HRA to deal with: dynamic plant-operator interactions, organizational influences, advanced systems, and non-reactor systems will be needed. On the applications side, it can be expected that a engineering issues will continue to arise, and that the resolution of these issues will require HRA (as part of PRA).

The specific activities to be addressed and their priorities will be discussed at appropriate times with RES management and the ACRS Subcommittee on Probabilistic Risk Assessment. The plan for post-FY 2005 human reliability analysis research will be developed as part of the process of updating the NRC's Risk-Informed Regulation Implementation Plan, and will be documented as an update to this report (as discussed under Support Task A, Section 3.15).

References

1. U.S. Nuclear Regulatory Commission, "NRC Program on Human Performance in Nuclear Power Plant Safety," SECY-00-0053, February 29, 2000.

2. U.S. Nuclear Regulatory Commission, "Risk-Informed Regulation Implementation Plan," SECY-00-0213, October 26, 2000.

3. U.S. Nuclear Regulatory Commission, "Nuclear Regulatory Commission FY 2000-2005 Strategic Plan," NUREG-1614, Vol. 2, September 2000.

4. G. Parry, "The need for, and some suggested characteristics of, an improved HRA approach for use in PSAs," in *Human Reliability Models: Theoretical and Practical Challenges* , H. Blackman, N. Siu, and A. Mosleh, eds., Center for Reliability Engineering, University of Maryland, College Park, MD, pp. 13-29, 1998.

5. U.S. Nuclear Regulatory Commission, *Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance* , NUREG-1560, 1997.

6. U.S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal Register* , Vol. 60, p. 42622 (60 FR 42622), August 16, 1995.

7. U.S. Nuclear Regulatory Commission, "Options for Risk-Informed Revisions to 10 CFR Part 50 - 'Domestic Licensing of Production and Utilization Facilities,'" SECY-98-300, December 23, 1998.

8. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.March 25, 1999.

9. U.S. Nuclear Regulatory Commission, *New NRC Reactor Inspection and Oversight Program* , NUREG-1649, Rev. 1, 2000.

10. R.J. Belles, et al, *Precursors to Potential Severe Core Damage Accidents: 1998* , U.S. Nuclear Regulatory Commission, NUREG/CR-4674, Vol. 27, 2000.

11. ASME Standard to be released May/June 2001.

12. American Nuclear Society, *External Events PRA Methodology Standard* , Draft, BSR/ANS-58.21, December 25, 2000.

13. National Fire Protection Association, *Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants, 2001 Edition* , February 2001.

14. U.S. Nuclear Regulatory Commission, "Reevaluation of the Pressurized Thermal Shock Rule (10 CFR 50.61) Screening Criterion," SECY-00-0140, June 23, 2000.

15. W.D. Travers, Executive Director for Operations, letter to Commissioners, "Final Technical Study on Spent Fuel Pool Accident Risk at Decommissioning Plants (WITS #199900132)," December 20, 2000.

16. Nuclear Energy Agency Committee on the Safety of Nuclear Installations, *Critical Operator Actions: Human Reliability Modeling and Data Issues*, NEA/CSNI/R(98)1, 1998.

17. A.D. Swain and H.E. Guttman, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications*, U.S. Nuclear Regulatory Commission, NUREG/CR-1278, 1983.

18. A.J. Spurgin, et al, *Operator Reliability Experiments Using Power Plant Simulators, Volume 2: Technical Report*, Electric Power Research Institute, EPRI NP-6937, July 1990

19. D.E. Embrey, et al, *SLIM-MAUD: An Approach to Assessing Human Error Probabilities Using Structured Expert Judgment*, U.S. Nuclear Regulatory Commission, NUREG/CR-3518, 1984.

20. E. Dougherty, Guest Editor, *Reliability Engineering and System Safety, 29*, Special Issue on Human Reliability Analysis, 1990.

21. H.W. Lewis, et al, *Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission*, NUREG/CR-0400, 1978.

22. T. Murley, Director, Office of Nuclear Reactor Regulation, memorandum to E. Beckjord, Director, Office of Nuclear Regulatory Research, "Organizational Factors Research," February 4, 1994.

23. U.S. Nuclear Regulatory Commission (2000). Letter from J. Rosenthal to J. Larkins, March 6, 2000. ADAMS Accession # ML003689518.

24. U.S. Nuclear Regulatory Commission, *Technical Basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA)*, U.S. Nuclear Regulatory Commission, NUREG-1624 Rev. 1, 2000.

25. Nuclear Energy Agency Committee on the Safety of Nuclear Installations, *Errors of Commission in Probabilistic Safety Assessment*, NEA/CSNI/R(2000)17, 2000.

26. W.J. Galyean, et al, "Integrated probabilistic safety assessment of pressurized thermal shock for U.S. pressurized water reactors," *Proceedings of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM5)*, Osaka, Japan, November 27-December 1, 2000.

27. C. Bieder, S. Vidal, P. Le Bot, "Feedback from the actual implementation of the MERMOS method," *Proceedings of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM5)*, Osaka, Japan, November 27-December 1, 2000.

28. G. Apostolakis, K. Davoudian, and J.-S. Wu, "The Work Process Analysis Model (WPAM)," *Reliability Engineering and System Safety, 45*, 107-125, 1994.

29. D.M. Murphy and M.E. Pate-Cornell, "The SAM framework: modeling the effects of management factors on human behavior in risk analysis," *Risk Analysis, 16*, 510-515, 1996.

30. E. Dougherty, 'Human reliability analysis - where shouldst thou turn?'," *Reliability Engineering and System Safety, 29*, 283-299, 1990.

31. B. Lydell, "A practitioner's view on the state of HRA methodology," Letter to the Editor, *Reliability Engineering and System Safety, 55*, 1997.

32. H. Blackman, "Introduction: a course for human reliability analysis," in *Human Reliability Models: Theoretical and Practical Challenges*, H. Blackman, N. Siu, and A. Mosleh, eds., Center for Reliability Engineering, University of Maryland, College Park, MD, pp. 1-11, 1998.

33. H. Blackman, N. Siu, and A. Mosleh, *Human Reliability Models: Theoretical and Practical Challenges*, Center for Reliability Engineering, University of Maryland, College Park, MD, 1998.

34. D.J. Wakefield, G.W. Parry, A.J. Spurgin, and P. Moieni, "Systematic Human Action Reliability Procedure (SHARP) Enhancement Project, SHARP1 Methodology Report," EPRI TR-101711, Electric Power Research Institute, Palo Alto, CA, December 1992.

35. J.A. Julius, E.J. Jorgenson, G.W. Parry, and A. Mosleh, "A procedure for the analysis of errors of commission in a probabilistic safety assessment of a nuclear power plant at full power, *Reliability Engineering and System Safety, 50*, 189-201, 1995.

36. U.S. Nuclear Regulatory Commission, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400 (NUREG-75/014), 1975.

37. A.D. Swain, "Human reliability analysis: need, status, trends and limitations," *Reliability Engineering and System Safety, 29*, 301-313, 1990.

38. A.D. Swain and H.E. Guttman, *Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications* (Draft Report for Interim Use and Comment), NUREG/CR-1278, U.S. Nuclear Regulatory Commission, Washington, DC, 1980.

39. A. D. Swain, *Accident Sequence Evaluation Program Human Reliability Analysis Procedure*, U.S. Nuclear Regulatory Commission, NUREG/CR-4772, 1987.

40. L. M. Weston, D.W. Whitehead, and N.L. Graves, *Recovery Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP): Development of the Data-Based Method*, U.S. Nuclear Regulatory Commission, NUREG/CR-4834, Vol. 1, 1987.

41. A.C. Payne, Jr., et al, *Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Internal Events Accident Sequence Quantification Main Report*, U.S. Nuclear Regulatory Commission, NUREG/CR-4832, Vol. 3, Part 1, 1992.

42. "Summary of Discussion Group I," in *Human Reliability Models: Theoretical and Practical Challenges*, H. Blackman, N. Siu, and A. Mosleh, ed., Center for Reliability Engineering, University of Maryland, College Park, MD, pp. 231-236, 1998.

43. S.B. Haber, J.N. O'Brien, D.S. Metlay, and A. Crouch, *Influence of Organizational Factors on Performance Reliability*, U.S. Nuclear Regulatory Commission, NUREG/CR-5538, 1991.

44. C. Acosta and N. Siu, "Dynamic event trees in accident sequence analysis: application to steam generator tube rupture," *Reliability Engineering and System Safety*, *41*, 135-154, 1993.

45. Y. Huang, et al, *Modeling Control Room Crews for Accident Sequence Analysis*, Massachusetts Institute of Technology, MITNE-296, 1991.

46. T.G. Ryan, "A task analysis-linked approach for integrating the human factor in reliability assessments of nuclear power plants," in *Accident Sequence Modeling: Human Actions, System Response, Intelligent Decision Support*, G.E. Apostolakis, P. Kafka and G. Mancini, eds., Elsevier Applied Science, London, 1987, pp. 219-234.

47. M.T. Barriere, W.J. Luckas, D.W. Whitehead, and A.M. Ramey-Smith, *An Analysis of Operational Experience During LP&S and Plan for Addressing Human Reliability Assessment Issues*, U.S. Nuclear Regulatory Commission, NUREG/CR-6093, 1994.

48. M.T. Barriere, et al, *Multidisciplinary Framework for Analyzing Errors of Commission and Dependencies in Human Reliability Analysis*, U.S. Nuclear Regulatory Commission, NUREG/CR-6265, 1995.

49. J. Reason, *Human Error*, Cambridge University Press, 1990.

50. D.D. Woods, et al, *Models of Cognitive Behavior in Nuclear Power Plant Personnel*, U.S. Nuclear Regulatory Commission, NUREG/CR-4532, 1986.

51. D.D. Woods, E.M. Roth and H. Pople, Jr., "Modeling human intention formation for human reliability," in *Accident Sequence Modeling: Human Actions, System Response, Intelligent Decision Support*, G.E. Apostolakis, P. Kafka and G. Mancini, eds., Elsevier Applied Science, London, 1987, pp. 169-200.

52. S.E. Cooper, et al, *A Technique for Human Error Analysis (ATHEANA)* , U.S. Nuclear Regulatory Commission, NUREG/CR-6350, 1996.

53. D.A. Powers, Chairman, Advisory Committee on Reactor Safeguards, letter to W.D. Travers, Executive Director for Operations, "NUREG-1624, Rev. 1, 'Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA)'," December 15, 1999.

54. J.T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, memorandum to the Commission, "Draft Report to the U.S. Nuclear Regulatory Commission on Reactor Safety Research from the Advisory Committee on Reactor Safeguards," March 22, 2001.

55. P. Shewmon, Chairman, Advisory Committee on Reactor Safeguards, letter to N.J. Palladino, Chairman, U.S. Nuclear Regulatory Commission, "ACRS Report on Draft NRC Staff Evaluation of Pressurized Thermal Shock Dated September 13, 1982," October 14, 1982.

56. J. C. Byers, et al, *Revision of the 1994 ASP HRA Methodology* , Idaho National Engineering and Environmental Laboratory, INEEL/EXT-99-00041, 1999.

57. D. Bley, et al, *RSO Evaluation Table* , letter report prepared for the U.S. Nuclear Regulatory Commission under JCN W6994, June 2000.

58. N. Siu, H. Woods, M. Dey, *NRC Fire Risk Research Plan: Fiscal Years 2001-2002* , U.S. Nuclear Regulatory Commission, Draft Report for Information, ADAMS ML003773018, November 5, 2000.

59. S.J. Collins, Director, Office of Nuclear Reactor Regulation, Memorandum to A.C. Thadani, Director, Office of Nuclear Regulatory Research, "User Need Request Related to Steam Generator Severe Accident Response and Testing of Steam Generator Tubes During Severe Accident Conditions," February 8, 2000.

60. L.J. Callan, Executive Director for Operations, memorandum to the Commission, "Status of the Environmental Qualification Task Action Plan (WITS Item 9300107)," February 5, 1998.

61. A.C. Thadani, Director, Office of Nuclear Regulatory Research, memorandum to W.F. Kane, Director, Office of Nuclear Material Safety and Safeguards, "Dry Cask Storage Probabilistic Risk Assessment; Draft Program Plan and Schedule," June 26, 2000.

62. U.S. Nuclear Regulatory Commission, *NRC Research Plan for Digital Instrumentation and Control* , Draft SECY letter, November 20, 2000.

63. N. Siu, "Risk assessment for dynamic systems: an overview," *Reliability Engineering and System Safety, 43*, No. 1, 43-73, 1994.

64. Y. Chang and A. Mosleh, "ADS-IDACrew: dynamic probabilistic simulation of operating crew response to complex system accidents," *Proceedings of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM5)*, Osaka, Japan, November 27-December 1, 2000.

65. W.T. Russell, Director, Office of Nuclear Reactor Regulation, memorandum to D.L. Morrison, Director, Office of Nuclear Regulatory Research, "Request for Research on Reactor Pressure Vessel Integrity," August 11, 1995.

66. C.J. Paperiello, Director, Office of Nuclear Material Safety and Safeguards, memorandum to D. Morrison, Director, Office of Nuclear Regulatory Research, "User Need Memorandum for a Dry Cask Storage System PRA," April 2, 1997.

67. A.C. Thadani, Director, Office of Nuclear Regulatory Research, memorandum to W.F. Kane, Director, Office of Nuclear Material Safety and Safeguards, "Response to Request for Probabilistic Risk Assessment of Dry Storage of Spent Nuclear Fuel," April 7, 2000.

68. R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts*, NUREG/CR-6372, April 1997.

69. N. Siu, "Uncertainty analysis and pressurized thermal shock: an opinion," white paper, U.S. Nuclear Regulatory Commission, September 3, 1999.

70. G. Apostolakis, "The concept of probability in safety assessments of technological systems," *Science*, *250*, 1359-1364, 1990.

71. G. Apostolakis, "A commentary on model uncertainty," in *Model Uncertainty: Its Characterization and Quantification*, A. Mosleh, N. Siu, C. Smidts, and C. Lui, eds., Center for Reliability Engineering, University of Maryland, College Park, MD, 1995, pp. 13-22.

72. U.S. Nuclear Regulatory Commission, *Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program*, draft NUREG-1742, April 2001.

73. U.S. Nuclear Regulatory Commission, *Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture*, NUREG-1570, 1998.

74. R. Lofaro, E. Grove, M. Villaran, P. Soo, and F. Hsu, *Assessment of Environmental Qualification Practices and Condition Monitoring Techniques for Low-Voltage Electric Cables*, NUREG/CR-6704, Vol. 1, February 2001.

75. U.S. Nuclear Regulatory Commission, *Office of Nuclear Regulatory Research Operating Plan*, October 20, 2000.

76. C.L. Smith, V.N. Shah, T. Kao, G. Apostolakis, *Incorporating Aging Effects into Probabilistic Risk Assessment — A Feasibility Study Utilizing Reliability Physics Models*, NUREG/CR-5632, November 2000.

77. Nuclear Energy Research Advisory Committee, *Long-Term Nuclear Technology Research and Development Plan*, report prepared for the U.S. Department of Energy, June 2000.

78. J. O'Hara, W. Stubler, and J. Higgins, *Hybrid Human-System Interfaces: Human Factors Considerations*, BNL report J6012-T1-4/96, Brookhaven National Laboratory, 1996.

79. B. Hallbert, A. Sebok, and D. Morrisseau, *A Study of Control Room Staffing Levels for Advanced Reactors*, NUREG/IA-0137, US Nuclear Regulatory Commission, November 2000, Washington DC.

80. "Memorandum of Understanding Between U.S. Nuclear Regulatory Commission and Electric Power Research Institute on Cooperative Nuclear Safety Research," signed by M.R. Knapp and R.L. Jones, 1997.

81. W. D. Travers, Executive Director for Operations, letter to D. A. Powers, Chairman, Advisory Committee on Reactor Safeguards, "NUREG-1624, Rev. 1, 'Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA)'," January 20, 2000.

Appendix A - ACRS Recommendations for ATHEANA and Resolution

In a December, 1999 review of a draft version of NUREG-1624, Rev. 1, the Advisory Committee on Reactor Safeguards (ACRS) provided a number of recommendations to the NRC staff [53]. In March 2001, the ACRS provided additional recommendations as part of a review of the NRC reactor safety research program (see p. 25 of Ref. 54).

Regarding the December 1999 recommendations, these are summarized below, along with notes indicating how these recommendations will be resolved in the HRA Research Program. Note that these resolutions are generally consistent with an earlier staff response to the ACRS review [81].

1. The quantitative portion of ATHEANA needs significant development.

   This issue will be dealt with by Task 4 of the HRA Research Program, "HRA Quantification and Uncertainty" (see Section 3.5 of this report).

2. The scope of ATHEANA should be extended to include normal activities that may cause a plant event.

   This issue will be dealt with by Task 11, "Latent Errors in HRA" and Task 12, "HRA Extended Applications."

3. The term "error forcing context" should be replaced by an alternative, more descriptive term.

   This term has become associated with ATHEANA, and has been used in a number of non-NRC publications and presentations. Nevertheless, its use will be reconsidered as that part of Support Task D, "Industry and International HRA Activities," aimed at developing a common language among cooperating HRA research programs.

4. A screening process should be developed.

   This issue will be dealt with by Task 13, "Formalized Methods: Screening, Individual and Crew Modeling."

5. The ATHEANA search process for deviations should take advantage of industry experience from developing symptom-based procedures.

   This issue will be covered under Task 1, "HRA Data Collection and Analysis."

6. Plant safety culture should be explicitly considered when evaluating the error forcing contexts.

   Task 11, "Latent Errors in HRA," will investigate sources of dependencies between latent errors. Through Support Task D, "Industry and International HRA Activities," the HRA Research Program will also keep track of international efforts looking at management and organizational factors (e.g., through the COOPRA working group on organizational influences on risk).

7. More example applications of ATHEANA need to be developed.

Task 4 (Pressurized Thermal Shock HRA), Task 5 (Fire HRA), and Task 6 (Steam Generator Tube Rupture HRA), will provide additional applications for ATHEANA. Depending upon the detailed analysis needs of the dry cask HRA effort (see Task 8), an ATHEANA analysis may be performed for this problem as well. Note that the earlier fire HRA application reviewed by the ACRS was in draft form, and will be substantially revised before being finalized.

The March 2001 ACRS recommendations are that "... the ATHEANA effort be terminated and a new plan for the quantification of the probability of human unsafe acts be developed." The second recommendation (regarding quantification) is being addressed by Tasks 1 ("HRA Data Collection and Analysis") and 3 ("HRA Quantification and Uncertainty") of the HRA Research Program. Regarding the first recommendation, as stated at the beginning of this report, the staff believes that ATHEANA has reached a state of development that allows its use in realistic applications. Therefore, the development of ATHEANA will be brought to an orderly close. ATHEANA will be used (as appropriate) in the staff's ongoing and upcoming challenging HRA applications (e.g., PTS, fire risk, SGTR). Methodology upgrades in areas of recognized weakness (e.g., see Task 12, "HRA Extended Applications") and problem areas arising during applications work are currently expected to use ATHEANA as a starting point, but may involve alternate HRA methods as well.