March 6, 2000

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Jack E. Rosenthal, Chief **/RA/**
Regulatory Effectiveness and Human Factors Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS HUMAN FACTORS SUBCOMMITTEE, MARCH 15, 2000, ON SECY-00-0053, "NRC PROGRAM ON HUMAN PERFORMANCE IN NUCLEAR POWER PLANT SAFETY"

We have four reports that we would like to share with the Advisory Committee on Reactor Safeguards prior to our meeting with the Subcommittee on March 15, 2000. We are also providing copies of the four reports to the Public Document Room. The reports provide the technical basis for summary statements contained in SECY-00-0053, "NRC Program on Human Performance in Nuclear Power Plant Safety." The reports are:

(1) "Summary of INEEL Findings on Human Performance During Operating Events." Report No. CCN 00-005421, Transmitted by letter, February 29, 2000. (Attachment 1)

(2) "Accident Sequence Precursor (ASP) Qualitative Analyses." (Staff) (Attachment 2)

(3) O'Hara, John M., and Higgins, James C., "Risk Importance of Human Performance to Plant Safety," Brookhaven National Laboratory, Report W6546-T1-2-10/99, Transmitted by letter, February 28, 2000. (Attachment 3)

(4) "Human Performance Programs at Other Agencies." (Staff) (Attachment 4)

If you have any questions concerning this, please contact J. Persensky at 415-6759, or Joel Kramer at 415-5891.

Attachments: As stated

cc w/atts.:
N. Dudley
S. Duraiswamy

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ATTACHMENT I

Summary of INEEL Findings on Human Performance
During Operating Events

Summary of INEEL Findings on Human Performance
During Operating Events

Background

In response to a need to better understand how human performance influences the risk associated with nuclear power plant operations, the US NRC Office of Nuclear Regulatory Research (RES) initiated work at the Idaho National Engineering and Environmental Laboratory (INEEL) to identify and characterize the influences of human performance in significant operating events. The purpose of this work is to analyze operational sources of data, to determine areas of research on human performance, and to provide the technical basis to support the development of an NRC program on human performance at nuclear power plants (NPPs).

Method

A combination of quantitative and qualitative analyses of operating events was conducted. Quantitative analyses make use of existing probabilistic risk assessment (PRA) methods and models. In an earlier NRC program, the INEEL developed the standardized plant analysis risk (SPAR) models[1] used in this report. These models are condensed PRAs of US commercial NPPs. This study used these SPAR models that had been previously selected by the Accident Sequence Precursor (ASP) Program at Oak Ridge National Laboratory (ORNL) found to have a conditional core damage probability (CCDP) of 1.0E-5 or greater.[2] Such events are deemed risk significant. A subset of these events in which human performance was an important factor was analyzed.

An INEEL team comprised of a plant systems/SPAR analyst, human factors/HRA specialist, and plant operations expert conducted qualitative analyses of events. The selection process for analysis first emphasized those events having augmented inspection team (AIT) or incident investigation team (IIT) reports available. To date, the team has analyzed 35 operating events to identify how human performance contributed to each event. Eleven of these events had limited human performance contribution and were not analyzed further. The 35 events were selected from those analyzed by the ASP program and ranked on the basis of ASP-calculated CCDP. There was no discernable pattern in terms of CCDP for those events with human performance contributions versus those events having limited contribution. Of the twenty-four events with human performance contribution, twenty events were analyzed by quantitative and qualitative modeling methods, the remaining four events were only analyzed by qualitative means, because they occurred during shut down, a mode not covered by current SPAR models.

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[1] INEEL used the Accident Sequence Precursor (ASP) program most current methods to calculate measures of risk associated with human performance in the selected events. Using the ASP methodology, conditional cores damage probability (CCDP) was calculated for each event.

[2] Currently, ASP/SPAR models exist for all nuclear power generating stations, however only limited coverage is provided for operating modes other than full power. Some of the risk significant operating events analyzed occurred in a plant mode that the current version of SPAR models does not analyze. In those instances where models are not available, qualitative analyses were performed and important human performance influences noted.

Process for Event Selection. Selection of the risk-significant events for analysis using the Level 1 SPAR models was begun by reviewing the licensee event reports (LERs) and other reports for ASP events that had occurred between January 1, 1992 and December 31, 1997 that had an ASP-calculated CCDP greater than 1.0E-05. The CCDP of these events was determined as part of the Accident Sequence Precursor (ASP) Program at Oak Ridge National Laboratory (ORNL). The selected events were then analyzed using a Level 1 SPAR model if the review of the LER indicated that (1) human factors contributed significantly to the event and (2) if the event could be analyzed with the Level 1 SPAR model (i.e., the event occurred at full power and the impacted components are included in the model).

With one exception, these event analyses used Rev 2QA versions of the Level 1 SPAR models. The exception is that the Rev 3 SPAR Model was used for the Millstone Unit 2 event assessment. Rev 3 SPAR Models, currently under development at the INEEL, incorporate the large and medium break LOCA (loss-of-coolant accident) initiating events that are required for the analysis of the Millstone 2 event.

It was necessary to perform SPAR analyses of these events in order to estimate the risk-significance of human performance to the increased CCDP, since SPAR analyses allow for calculation of the human performance contribution to risk. It is not possible to extract this information from the ASP LER analyses reported in the “Precursors to Potential Severe Core Damage Accident”, NUREG/CR–4674 Volumes 17 through 25, because these reports are summaries of earlier analyses. Thus, they typically do not document the base core damage probability (CDP). Calculation of the risk factor increase (RFI) and event importance used in the present study requires the CDP as input. Additionally, significant improvements to methods and data have been accomplished in the ASP Program and it was decided to employ the latest generation of QA-checked models.

In this research, both a CDP and an updated CCDP are calculated for each event analyzed with the Level 1 Rev. 2QA SPAR models. (these same calculations were also performed for Millstone Unit 2 using the Rev 3 model). The Level 1 Rev 2QA SPAR model results do not necessarily match the results reported by the ORNL ASP Program nor should they be expected to do so. These differences are due to model version (enhanced detail of components and systems) and analysis methodology differences. For example, the models and software platform for ASP have evolved from split-fraction to linked fault tree analysis. Underlying basic event and initiating event probabilities have been refined as well.

Results

The analyses performed to date underscore the significant contributions that human performance has made to operating events. Specific failures in operating events were analyzed with SPAR models. This includes failed or erroneous human actions that caused event initiation, equipment unavailability, or demand failures. SPAR models thus analyze the sensitivity of plant risk to failed or erroneous human actions. In addition to human errors, random systems and equipment failure also occurred during several events. An event importance measure of greater than or equal to 1.0E-6 was used as screening criteria for inclusion of events in this study. This is in line with guidance suggested by REG GUIDE 1.174.

Additionally, a risk factor increase was developed to indicate the risk significance of an event.[3] This measure is calculated by dividing the event CCDP by the nominal CDP value.

Human performance produced significant increases in plant risk. Event importance for the twenty events ranged from 5.2E-3 to 1.0E-6. The human error percent contribution to event importance ranged from 10% (Comanche Peak 1 - LER 445/95-003) to 100% (for 16 out of the 20 remaining events). Three remaining events also demonstrated strong human error contribution to event importance (McGuire 2 with 82%; Haddem Neck at 48%, and DC Cook with 80%). The human error contribution to the event importance calculated in the present study represents a ratio of the portion of the event importance attributed to human error to the total event importance. This ratio is defined by the following equation:

$$\text{Human Error \% Contribution} = [(CCDP_{HE} - CDP)/ (CCDP_{event} - CDP)] \times 100\%$$[4]

Table 1 presents a summary of the operating events, ASP screening or reference selection value, corresponding date and LER number, conditional core damage probability, risk factor increase, event importance and the human performance contribution to the event importance. These risk increases were due in large part to errors and failures committed by personnel and organizations that operate and maintain these plants. For example, component failures due to human failure led to initiating events at Oconee Unit 2 (LER #27092004) and Dresden (LER # 2496004). The corresponding event importance for these events was 3.6E-03 (Oconee 2), and 2.6E-05 (Dresden). Other human failures resulted in initiating events without additional component failures at Sequoyah 1 and 2 (LER #327/92-027, with a CCDP of 1.1E-04) and Beaver Valley 1 (LER#334/93-013, with a CCDP of 6.2E-05). These events have CCDPs that represent a large departure from the nominal case.

The initiating event for Sequoyah Unit 1 operating event above in 1992 (LER #32792027) involved circuit breaker failure caused by failure of maintenance pre-installation testing and planning that, in turn, lead to loss-of-offsite power. The initiating event at Beaver Valley involved maintenance crew errors during an outage leading to inadvertent application of 125 Voltage DC in the switchyard. This resulted in the opening of 7 breakers in the 345KV system; 3 breakers in the 138KV breakers, and initiated the loss of electrical load at Unit 1. In 1996, the failure of a feedwater regulating valve (FRV) leading to subsequent reactor trip and ECCS actuation at Dresden could be traced to maintenance practices and running with only FRV operational. Switchyard faults resulting from failure to respond to industry notices and internal engineering notices lead to a loss of offsite power (LOOP) the recovery of which was complicated by inadequate procedures and poor work package preparation (Oconee 2 LER # 27092004). During other operating events analyzed, human errors and failures resulted in other equipment unavailability. As a result of this unavailability, plant systems did not perform their intended functions when demanded to do so by an automatic signal or manual command.

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[3] The factor increase compares the analyzed event CCDP to the baseline CDP (CCDP/CDP). For example, a factor increase of 2 represents a doubling of the core damage probability when given sets of components are guaranteed/postulated to be failed. For events with a CDP of 1.0E-05 or greater a factor increase of 1.1 would represent a risk change (delta) of at least 1.0E-06 meeting the guidance of Regulatory Guide 1.174 (1998)

[4] Terms are as follows: $CCDP_{HE}$ = conditional core damage probability for the event attributed to human error; CDP = nominal core damage probability as reflected in the PRA, and $CCDP_{event}$ = conditional core damage probability associated with the event.

Table 1. INEEL Results of SPAR Conditional Core Damage Probability Analyses Ranked by Event Importance

Analysis Number	ASP Reference and Screening Basis Value	Nuclear Power Generating Facility	Event Date	LER Number(s)	SPAR Analysis CCDP	Risk Factor Increase (CCDP/CDP)	Event Importance (CCDP-CDP)	Human Error Percent Contribution to Event Importance
1	2.1E-04	Wolf Creek 1	1/30/96	482/96-001	5.2E-03	24,857.0	5.2E-03	100
2	2.1E-04	Oconee 2	10/19/92	270/92-004	3.2E-03	86.5	3.2E-03	100
3	1.2E-04	Perry 1	4/19/93	440/93-011	2.1E-03	242.1	2.1E-03	100
4	2.2E-04	Oconee 2	4/21/97	270/97-001	7.1E-04	2.5	4.3E-04	100
5	1.3E-05	Limerick 1	9/11/95	352/95-008	4.8E-04	9.8	4.3E-04	100
6 [1]	2.0E-04 [2]	Indian Point 2	8/31/99	AIT 50-246/99-08	3.5E-04	25.0	3.4E-04	100
7	9.3E-05	McGuire 2	12/27/93	370/93-008	4.6E-03	2.4	2.7E-04	82
8	2.1E-04	Robinson 2	7/8/92	261/92-013, -017 & -018	2.3E-04	4.2	1.8E-04	100
9	6.5E-05	Haddam Neck	6/24/93	213/93-006 & -007 AIT 213/93-80	2.0E-04	4.3	1.5E-04	48
10	3.2E-05	Oconee 1 Oconee 2 Oconee 3	12/2/92	269/92-018	1.5E-04	125.0	1.5E-04	100
11	1.8E-04	Sequoyah 1 Sequoyah 2	12/31/92	327/92-027	1.1E-04	14,103.0	1.1E-04 [3]	100
12	5.5E-05	Beaver Valley 1	10/12/93	334/93-013	6.2E-05	10,690.0	6.2E-05 [3]	100

Analysis Number	ASP Reference and Screening Basis Value	Nuclear Power Generating Facility	Event Date	LER Number(s)	SPAR Analysis CCDP	Risk Factor Increase (CCDP/CDP)	Event Importance (CCDP-CDP)	Human Error Percent Contribution to Event Importance
13 [1]	NA [4]	Dresden 3	5/15/96	249/96-004	2.6E-05	15.3	2.4E-05	100
14	1.1E-04	St. Lucie 1	10/27/97	335/95-005	3.8E-05	2.9	2.5E-05	100
15	6.5E-05 [5]	Comanche Pk 1	6/11/95	445/95-003 & -004	1.9E-05	146.2	1.9E-05	10
16	6.0E-05	Arkansas Nuclear One 2	7/19/95	368/95-001	1.4E-05	73.7	1.4E-05	100
17	5.6E-04[6]	Arkansas Nuclear One 1	5/16/96	313/96-005	9.6E-06	50.5	9.4E-06	100
18	3.7E-05	D. C. Cook 1	9/12/95	315/95-011	3.3E-05	1.2	4.9E-06	80
19	1.3E-04	LaSalle 1	9/14/93	373/93-015	4.5E-05	1.07	3.0E-06	100
20	7.7E-05	Millstone 2	1/25/95	336/95-002	2.6E-05	1.04	1.0E-06	100

1. This event and corresponding LER and/or AIT were identified outside of the ASP screening approach and is one of many events where there was sufficient challenge to plant safety systems and human performance involvement had been noted
2. CCDP value obtained from AIT report.
3. The event importance value reported is actually the conditional core damage probability value as this analysis was strictly an initiating event assessment without additional component failures. The calculation of an event importance is not reasonable in these cases.
4. Not Applicable - this event was not analyzed by the ASP/LER program.
5. CCDP value reported in the 1995 ASP/LER NUREG/CR.
6. Two orders of magnitude difference between ANO-1 screening values and the SPAR analysis values represents technical refinements and enhanced ability to model a variety of transient types not available in earlier analyses.

Failures in design test adequacy resulted in main steam safety valve demand failure to close at Arkansas Nuclear One, Unit 1 (LER # 31396005) and main feed pump failure to run. The event importance for this event was 9.4E-06. Latent failures in the design review process for Arkansas Nuclear One, Unit 2 in 1995 (LER #36895001) contributed to auxiliary feedwater motor operated valve common cause failure. The event importance for this event was 1.4E-05. Design deficiencies failures combined with configuration management problems at Indian Point 2 resulted in loss of vital AC power and loss of DC power. (AIT#50-246/99). Key to this event was failure to control setpoints on safety-related equipment and failure to maintain the load tap changer in position required by licensing basis. The event importance for the Indian Point 2 event was 3.4E-04.

Latent failures at Wolf Creek (LER #48296001) in warming line design, lack of technical knowledge regarding conditions that cause frazil icing, failure to respond to industry notices, errors in technical specification interpretation and maintenance failures for packing of the turbine driven auxiliary feed pump coupled with active errors in terms of declaring equipment operable without engineering evaluation or root cause analysis, and failure to transfer information concerning the state of the ultimate heat sink, contributed to the event. The risk factor increase for this event at 24,578, was the largest observed in the sample of operating events analyzed to date. It is significant that almost all of this increase in risk was due to human performance issues. The event importance for this event was 5.2E-03. Human performance was a key factor in the initiation of these events and the risk increase that resulted.

The percent human error contribution to event importance presented in Table 1 was determined in the following manner. The team of analysts reviewed the components failed in the operational event and asked the following questions(s):

> Were component failures due to inadequate maintenance, surveillance, or testing?
> Did operators or maintenance personnel operate or maintain equipment improperly, leading to its failure or unavailability?
> Did work package design, procedure development or reviews contribute to the failure(s)?
> Did the technical knowledge of staff contribute to initiating events, failures or unavailability for components modeled in the PRA?
> Did the organization fail to respond to industry notices or delay corrections to known design deficiencies that would have prevented the event from occurring?

Affirmative answers to any of these questions for a component that was modeled as failed in the PRA resulted in a determination that the percent human error contribution to that component's failure was 100%. The actual contribution to the event varies, as a function of that component's contribution to the CCDP. For example, the value of 82% listed for McGuire 2 event on 12/27/93 represents an exact calculation of the contribution of human error to a subset of all failed components for that operational event.

During the course of the analysis, we conducted 15 initiating event (IE) assessments including LOOP (loss-of-offsite power), SGTR (steam generator tube rupture), SLOCA (small break LOCA) and TRANS (transient). Two of the events (McGuire 2 and ANO-1) were combinations of two initiating event assessments. The former contained SGTR and LOOP IEs and the latter contained SLOCA and TRANS IEs. Additionally there were seven condition assessments performed for instances where a component failure existed, but during the period the condition existed no initiating event occurred.

Qualitative analyses produced further insights regarding the role of human performance in operating events. Table 2 summarizes the human performance influences[5] and error types that were identified in the analysis of operating meeting the ASP screening value of 1.0E-05 or greater and includes qualitative and quantitative findings.

The most predominant errors that contributed to the increase in plant risk were latent errors. Reason (1990) characterized latent errors as those that occur previous in time to the event and influence the event in some manner. Active errors also occurred during events but to a lesser extent. Eighteen percent of errors were active versus 82% latent errors. Latent errors are more difficult to detect and mitigate because they occur with no immediate, observable impact.

Evidence from these analyses suggests that latent errors, including those associated with maintenance errors, are an important contributor to the significance of the highest conditional core damage probability events which have occurred over the last five years. Latent errors, as used here, are characterized by James Reason (1990) as those that occur previous to the event and influence the event in some manner. However, they are seldom explicitly modeled in PRAs because most analysts assume they are implicitly accounted for in equipment failure rates. Errors can be introduced by a variety of human and organizational sources, some of which influence the significance of operating events. In general, the work processes by which human errors were introduced include design, review, configuration management of drawings and procedures, maintenance, surveillance, and corrective actions. In a later work by Reason (1997), which is based on the review of numerous major accidents from around the world, the term latent conditions is used to characterize problems resulting from poor design, gaps in supervision, undetected manufacturing defects, maintenance failures, unworkable procedures, clumsy automation, shortfalls in training, or less than adequate tools and equipment. Such conditions may be present for many years before they combine with local circumstances and active failures to penetrate a system's defenses.

Multiple failures occurred in the events analyzed. Between six and 12 small individual failures were observed for most operating events. Most of these individual failures were minor, not sufficient by themselves to cause an accident. Their occurrence had a cumulative effect and, thereby, challenged plant systems and resources. For example, a single, inadequate design review may be insufficient to produce a major accident. Other opportunities exist for an organization to detect and correct deficiencies through maintenance, surveillance, and test. The most commonly-occurring failures observed in operating events stemmed from command and control deficiencies; inadequate maintenance practices, inadequate procedures, failure to correct known deficiencies, failure to respond to industry notices, failure to enforce standards, and inadequate testing after equipment service.

Design deficiencies, inadequate maintenance practices and deficiencies in procedures or the procedural review process had a major influence on operating events. Utility inattention to recurrent problems was also evident in a large number of events. This included inattention to

[5] Attempts were made to assign a single error to an individual performance category. In instances where an error crossed two categories, a 0.5 value was assigned to both appropriate categories. Thus, the double counting of a single error was precluded. In the present study, there are fourteen instances where double influence category representation for an error is appropriate.

NRC inspection findings, internal engineering department notices, industry notices, vendor notices, and previous LERs. In many cases, known problems were not called out, prioritized for work, or acted upon. This includes operating with known design deficiencies, permitting "work arounds" (i.e., requires alternate operator actions usually manual actions to operate the system), or documenting problems and solutions but failing to take action in time to prevent an equipment or system failure. In eight of twenty-four events, failure to follow plant or industry trends, respond to industry notices, owners groups reports, or pay attention to recurrent problems figured prominently.

Table 2. Performance Categories, Associated Human Performance Influence and Error Types for Significant (1.0E-5 or greater) events (N=24)

	Category (followed by human performance influence)	Latent Errors	Active
	Operations		
O1	Command and control issues including crew resource management,	4	14
O2	Failure to follow safe practices	1	
O4	Inadequate knowledge or training	12	2
O5	Incorrect operator actions	3	7
O6	Communications	3	2
	Design and Design Change Work Process		
D1	Design deficiencies	19	
D2	Design change testing	5	
D3	Inadequate engineering evaluation	8	
D4	Ineffective indications for abnormal condition	1	
D6	Configuration management	6	1
	Maintenance Work Process		
M1	Poor work package preparation, QA and use	7	
M2	Inadequate maintenance practices	17	
M3	Inadequate technical knowledge	4	
M4	Inadequate post-maintenance Testing	9	
	Procedural Design and Development Process		
P1	Inadequate procedures	18	1
	Organizational Learning and Corrective Action Program		
L1	Failure to respond to industry and internal notices	7	
L2	Failure to follow industry operating practices	2	
L3	Failure to identify by trending and problem reports	10	
L4	Failure to validate vendor reports		
	Work Prioritization		
W1	Failure to correct known deficiencies	15	
W2	Continue to operate during unstable conditions	1	2
	Management Oversight		
S1	Inadequate supervision	10	5
S2	Inadequate knowledge of plant systems and plant requirements	2	1

S3	Organizational structure	1	

Twenty-five percent of the total active, post-initiator failures involve command and control and resource management failures. For example, command and control between Oconee Unit 2 and Keowee hydro compromised plant response. Keowee staff was performing actions that affected emergency power at Oconee without notification or permission of Oconee control room management. The Beaver Valley LOOP event on 10/12/93 failed to include operations in maintenance planning and their were no clear cut protocols for the Unit 2 staff to direct operations at the switchyard. At McGuire 2, during the LOOP event on 12/27/93, the duties and responsibilities for the Senior Reactor Operator (SRO) during emergency conditions were not well defined. Command and control was an issue at other plants. Staffing problems and interference from the field also influenced crew response at Salem 1 when cooling water (CW) was lost during river grass intrusion.

Summary

Most of the significant contributing human performance factors found in this analysis of operating events are missing from the current generation of probabilistic risk assessments (PRAs) including the individual plant examinations (IPEs). The current generations of PRAs do not address well the kinds of latent errors, multiple failures, or the type of errors determined by analysis to be important in these operating events.

As previously stated, latent errors figured prominently in events. Their under-representation in PRAs is due, in part, to how most HRA methods model human error. Most HRAs in current generation PRA model active errors, i.e., those that occur following the initiation of an event. Most operator errors are modeled as failures associated with following procedures or time dependent factors in an event sequence (e.g., insufficient time to accomplish necessary actions). In the PRAs, human performance accounts for 5-8% of risk (i.e., contributes to less than 10% of core damage frequency estimates). Human performance is important in sequences that require operator actions to initiate or operate plant systems to mitigate the effects of an initiating event and subsequent equipment failures. Examples of such actions include switching to recirculation supply mode, “feed and bleed” or once-through core cooling, and depressurization and cooldown of the reactor.

In the 20 operating events analyzed to date using qualitative and quantitative SPAR methods, the average contribution of human performance to the event importance was over 90%. The events were selected because they were data rich, and human performance was well characterized. In contradiction to error modeling in PRAs, omissions and commissions in following procedures or taking actions within a given time were not found to be major determinants of risk increase. Furthermore, active errors, although important, were observed to represent the minority of human errors and failure events. The kinds of errors that dominated events were latent. They preceded the operating event and their effects were not directly observable. Most of them were insufficient in themselves to cause the event. Combined with other failures and errors, however, they produced challenges to plant systems and resources that increased risk. In most events, inadequate attention to industry and NRC Notices, as well as known deficiencies in the plant before the event contributed to their occurrence. In nearly all cases, plant risk more than doubled as a result of the operating event – and in some cases increased by several orders of magnitude over the baseline risk presented in the PRA. This increase was due, in large part, to human performance.

The results to date are preliminary. They represent analyses of approximately half of the risk-significant operating events identified for analysis in this program. If trends observed to date continue across the remainder of events, future research directions are suggested. For example, the preponderance of latent errors identified in operating events suggests a degree of importance not previously witnessed. This may support the need to determine whether the impact of latent errors is currently accounted for in models of plant risk, or to more thoroughly examine the mechanisms by which their prevention or detection could be improved. In addition, further analyses may support the need to better understand the impact of and mechanisms by which smaller, less-significant errors combine to produce larger, more significant effects. Other issues that may warrant additional study include the work processes and practices by which licensees control maintenance work, and mechanisms by which recurrent problems and notices are addressed.

In terms of modeling, the question of how to evaluate the potential impact of latent failures on events not covered by this review poses a challenge. For example, what is the expected impact of latent factors that have only been implicitly addressed by PRA in the past? Is there a way to determine generic impacts or should analysts consider these factors on a plant specific basis? Are the existing logic structures used in PRA the appropriate ones for incorporating this information? How does this information complement or support current efforts in the field of HRA to address the issues of errors of commission and context? What implications are suggested by this research for the way the NRC currently conducts its inspections?

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ATTACHMENT II

Accident Sequence Precursor (ASP) Qualitative Analyses

List of Tables

Accident Sequence Precursor (ASP) Qualitative Analyses

SUMMARY

This appendix presents the results of a qualitative analysis of 48 Accident Sequence Precursor (ASP) events with conditional core damage frequencies (CCDPs) of 1.0E-05 or higher that occurred between January 1, 1992 and December 31, 1997.

The objective of this analysis was to assess qualitatively the significance of human performance in these events.

The primary documents used in this analysis were: (1) the ASP reports and (2) Licensee Event Reports referred to in the ASP reports. Additional documents included Augmented Investigation Team (AIT) and NRC inspection reports, if available, and the Human Performance Event Database.

This analysis is subject to the following limitations: (1) The analysis was limited to the factual information contained in these documents, (2) Some of the results are based on a small number of cases, and (3) All of the data have not yet been given full independent review or quality assurance checks.

The general conclusion is that, despite these limitations, the analysis provides substantial, additional information supporting the conclusion that human performance in nuclear power plants is risk significant.

Specific conclusions are as follows:

- Human performance (including cause, positive recovery and negative recovery) was an important factor in 38 out of the 48 (79%) ASP events.

- Human performance was a cause of one or more equipment failures in 35 out of the 48 (73%) ASP events.

- In the 48 ASP events, there were 63 human action causes of equipment failure. Our criterion for calling a human action a “cause” of an equipment failure was: If human action had NOT occurred, the equipment failure would not have occurred. (Note: In many cases, a sequence of multiple human actions (i.e., causes) resulted in one equipment failure. Therefore, the number of human action causes can exceed the number of events.)

- The performance shaping factors (PSFs) associated with the highest number of causes were Procedure Quality, Communications, and Procedure Use.

- The job functions associated with the highest number of causes were Maintenance; Electrical or I&C; and Engineering or Design. Operations personnel performed only 6 actions that caused equipment failure, which is less than half the number performed by maintenance personnel.

EXPLANATION OF THE TABLES

Table 1 presents a list of the 48 ASP events analyzed in order of descending CCDP. It presents (1) the Licensee Event Report (LER) numbers associated with the event, (2) the CCDP, (3) the name of the event provided by the ASP report, (4) whether important human performance was involved in the event, (5) the percent power at the time of the event, and (6) whether the event was investigated by an NRC Augmented Investigation Team (AIT).

- Human performance (including cause, positive recovery and negative recovery) was an important factor in 38 out of the 48 (79%) ASP events.

Table 2 presents a list of the same 48 ASP events. However, Table 2 also lists specific human actions associated with each event. Human actions are classified as "Cause", "Positive Recovery", or "Negative Recovery.

- Human performance was a cause of one or more equipment failures in 35 out of the 48 (73%) ASP events.

- A "Cause" of an equipment failure was defined as: If that human action had NOT occurred, the equipment failure would not have occurred.

- Some ASP events involve multiple equipment failures. However, each "Cause" is a cause of one (and only one) of the equipment failures in an ASP event. (The major equipment failures are usually listed in the title of the ASP report.)

- Although the analysis of causes is comparatively rigorous, the analysis of "Recovery" is less so. It is less rigorous in part because recoveries can take place in multiple ways (e.g., manual action or automatic action), whereas causes of equipment failures can be specified.

- A "Positive Recovery" is a recovery, or a part of a recovery, in which the licensee staff generally performed well. A "Negative Recovery" is a recovery, or part of a recovery, in which the licensee staff had problems.

Table 3 presents a list of the same 48 ASP events, along with the human actions (cause, positive recovery, negative recovery) classified by: (1) job function and (2) performance shaping factors associated with the human actions.
The job functions (Column 1) are:

1. Operations.
2. Maintenance.
3. Electrical or Instrumentation and Control.
4. Engineering or Design.
5. Vendor or Contractor.
6. Training.
7. Management specified explicitly.

Blank Other; unspecified management; not known; not applicable.

The performance shaping factors (Columns 2 - 12) are:

2. Procedure Quality.
3. Procedure Use.
4. Training.
5. Knowledge.
6. Man/Machine Interface.
7. Staffing Level.
8. Workload.
9. Corrective Action (i.e., corrective action on prior similar events).
10. Communications.
11. Work Control.
12. Design Practices.

Each performance shaping factor is rated as either:

1. Positive influence (i.e., would tend to lower probability of error).
2. Negative influence.

Blank No effect; don't know; not mentioned.

The information relative to causes in Table 3 is further summarized in Tables 4 and 5.

Table 4 presents an analysis of the information in Table 3 on negative PSFs. (The few positive PSFs for causes are not analyzed.) Table 4 indicates that the PSF "Procedure Quality" had a negative influence on 12 causes of equipment failures. "Communication" had a negative influence on 7 causes of equipment failures, and so forth.

The total number of causes in Table 4 is less than the total number of causes in the database, because negative PSFs were not mentioned in the source documents for all of the causes.

Table 5 presents an analysis of the information in Table 3 on Job Functions associated with causes of equipment failures. Table 5 indicates that maintenance personnel performed 17 actions that caused equipment failure. Electrical and I&C personnel performed 14 actions that caused equipment failure. Operations personnel performed only 6 actions that caused equipment failure.

Table 1: Important Human Performance (HP) Items for 1992-97 ASP Events

Explanation: CCDP is the conditional core damage probability listed for the event. "HP: Yes, No" lists whether human performance (cause, positive recovery, or negative recovery) was identified for the event. "AIT: Yes, No" indicates whether or not an Augmented Inspection Team investigated the event. Note: A total of 48 events involving 56 units
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Event ID No. **Plant** **LER#**	**Date**	**CCDP**	**Event**	**HP** Yes No	**%Power**	**AIT**
1. Wolf Creek 48294013	09/17/94	3.0E-03	Inadvertent RCS Draindown	Y	0	N
2. Catawba 241496001	02/06/96	2.1E-03	Loss of Offsite Power with EDG B Unavailable	Y	100	N
3. Maine Yankee 30997004	01/22/97	8.2E-04	RCS Hot-Leg Recirc Valve Subject to Pressure Locking	N	0	N
4. Arkansas Nuclear 1 31396005	05/19/96	5.6E-04	Reactor Trip and Subsequent Steam Generator Dryout (SGTR)	Y	100	Y
5. Oconee 328797003	05/03/97	5.4E-04	Two HPSI Pumps Damaged; Low Water in LDST	Y	0	Y
6. St. Lucie 133597011	11/02/97	3.4E-04	Non-Conservative RAS Setpoint	Y	0	N
7. Fort Calhoun 28592023 28592028	07/03/92	2.5E-04	Rx High Pressure Trip and LOCA	Y	100	Y
8. Oconee 227097001	04/21/97	2.2E-04	Unisolable RCS Leak	N	100	N
9. Oconee 227092004	10/19/92	2.1E-04	LOOP with Failed Emergency Power Source	Y	100	Y
10. Robinson 226192017	08/22/92	2.1E-04	LOOP with SI Pump Recirc Line Obstructed	Y	100	N
11. Wolf Creek 48296001 48296002	01/30/96	2.1E-04	Reactor Trip with a Loss of Train A of Essential Service Water and TDAFW Pump	Y	98	Y

Event ID No. **Plant** **LER#**	**Date**	**CCDP**	**Event**	**HP** Yes No	**%Power**	**AIT**
12. Sequoyah 1 32792027 Sequoyah 2 32892027	12/31/92	1.8E-04 1.8E-04	LOOP and Dual Unit Trip	Y	100100	N
13. Turkey Point 3 Turkey Point 4 25192S01	08/24/92	1.6E-04 1.6E-04	LOOP Due to Hurricane Andrew	Y	0	N
14. Catawba 1 41393002 Catawba 2 41493002	02/25/93	1.5E-04	Essential Service Water Potentially Unavailable	Y	1000	N
15. Haddam Neck 21396016	08/01/96	1.5E-04	Potentially Inadequate RHR Pump NPSH Following a Large or Medium Break LOCA	Y	0	N
16. Haddam Neck 21394004	02/16/94	1.4E-04	PORVs and Vital 480 Volt ac Bus Degraded	Y	0	Y
17. LaSalle 137393015	09/14/93	1.3E-04	Scram and Loss-of-Offsite Power	Y	100	Y
18. Perry 44093011	03/26/93	1.2E-04	Service Water System and Problem Due to Clogged Suppression Pool Strainers	Y	100	Y
19. St. Lucie 133595004	01/09/95	1.1E-04	Failed Power-Operated Relief Valves (PORVs), Shutdown Cooling Availability and other problems.	Y	0	N
20. McGuire 237093008	12/27/93	9.3E-05	Rx Trip, then LOOP, Coincident with Failure of an MSIV to Close	Y	100	Y
21. Waterford 338295002	06/10/95	9.1E-05	Reactor Trip, Breaker Failure, and Fire	Y	100	Y
22. Millstone 233695002	01/25/95	7.7E-05	Containment Sump Isolation Valves Potentially Unavailable Due to Pressure Locking	Y	0	N

Event ID No. Plant LER#	**Date**	**CCDP**	**Event**	**HP** Yes No	**%Power**	**AIT**
23.Seabrook 144396003	05/21/96	7.6E-05	Turbine-Driven Emergency Feedwater Pump Unavailable in Response to a LOCA	Y	100	N
24. Oyster Creek 21992005	05/03/92	7.1E-05	LOOP Due to Forest Fire	N	100	N
25. Haddam Neck 21393006 21393007	06/27/93	6.5E-05	LOOP and Other Events While Conducting Tests	Y	100	Y
26. Arkansas Nuclear 2 36895001	01/19/95	6.0E-05	Loss of DC Bus Could Fail Both EFW Trains	Y	96	N
27. Quad Cities 226593010	04/22/93	6.0E-05	Degradation of Both Emergency Diesel Generators	Y	0	Y
28. Beaver Valley 133493013	10/12/93	5.5E-05	Dual-Unit Loss-of-Offsite-Power	Y	100	N
29. Prairie Island 1 28296012 Prairie Island 2 30696012	06/29/96	5.3E-05 5.3E-05	LOOP to Safeguards Buses on Both Units	Y	100100	N
30. Arkansas Nuclear 1 31393003	09/30/93	5.1E-05	Both Trains of Recirculation Unavailable for 14 Hours	Y	0	N
31. Palo Verde 252993001	03/14/93	4.7E-05	SG Tube Rupture	Y	99	Y
32.DC Cook 131595011	09/01/95	3.7E-05	One Safety Injection Pump Unavailable for Six Months	Y	0	N
33. Salem 1 27296002 Salem 2 31196002	01/10/96	3.6E-05 3.6E-05	Charging Pump Suction Valves from the RWST Potentially Unavailable	N	00	N

Event ID No. Plant LER#	Date	CCDP	Event	HP Yes No	%Power	AIT
34.Robinson 226192013	07/10/92	3.5E-05	Dam Introduced into the Suction Piping of Safety Injection, RHR, and Containment Spray Pumps and into RWST Could Have Affected Multiple Safety Components	Y	100	N
35. Oconee 1 26992018 Oconee 2 27092018 Oconee 3 28792018	12/02/92	3.2E-05	Both Keowee Units Potentially Unavailable	N	1001001 00	N
36. Comanche Pk Unit 1 44595003	06/11/95	2.9E-05	Reactor Trip and AFW Pump Trip with Second AFW Pump Unavailable	N	100	N
37. Zion 230494002	03/07/94	2.3E-05	Unavailability of Turbine-Driven Auxiliary Feedwater Pump	N	0	N
38. Arkansas Nuclear 1 31395005	04/20/95	2.0E-05	Trip with One Emergency Feedwater (EFW) Train Unavailable	N	100	N
39. Sequoya 1 Sequoya 2	12/31/92	1.8E-04	Loss of Offsite Power	Y	100100	N
40.River Bend 45894023	09/08/94	1.8E-05	Rx Trip but Turbine Generator Failed to Trip	Y	97	Y
41.Crystal River 330292001	03/27/92	1.7E-05	LOOP with Inoperable Vital Bus Inverter	Y	98	N
42. Byron 145496007	05/23/96	1.7E-05	Transformer Bus Fault Causes a LOOP	N	0	N
43. St. Lucie 238995005	11/20/95	1.4E-05	Failure of One Emergency Diesel Generator and the Potential for the Common Cause Failure of the Other Train EDG	N	0	N

Event ID No. Plant LER#	**Date**	**CCDP**	**Event**	**HP** Yes No	**%Power**	**AIT**
44. Callaway 48392011	10/17/92	1.3E-05	Loss of Main Control Room Annunciators Following Supply Power Loss	Y	100	Y
45. Calvert Cliffs 2 31894001	01/12/94	1.3E-05	Trip, Loss of 13.8-KV Bus, and Short-term Salt Water Cooling Unavailable	Y	100	N
46. Limerick 135295008	09/11/95	1.3E-05	Safety-Relief Valve Fails Open, Reactor Scrams, and Suppression Pool Strainer Fails	Y	100	N
47. South Texas 1 49893005 49893007	01/20/93	1.2E-05	Unavailability of One Emergency Diesel Generator and the Turbine-Driven Auxiliary Feedwater Pump	Y	95	N
48. Point Beach 1 26694002 Point Beach 2 30194002	02/08/94	1.2E-05	Both Diesel Generators Inoperable	Y	100 100	N

Table 2: Human Action (Cause, Positive Recovery and Negative Recovery) Descriptions

Explanation: “Event ID No.” is the identification number for the event, also used in Tables 1 & 3.“Action ID” is a sequential identification letter for each human action identified for each event, also used in Table 3.
“Human Action Description.” Human actions are categorized as (1) “Cause”, (2) +Recovery, which means a generally positive recovery, (3) “-Recovery”, which means a recovery with problems, and (4) Blank, which means no human action was identified.

Event ID No.	Plant Name	Action ID	Human Action (Cause and Recovery) Description
1	Wolf Creek 1	a	(Cause) Shift Supervisor granted permission, against procedures, to overhaul the valve. (-PSF: The outage was short.)
1	Wolf Creek 1	b	(Cause) Supervising Operator granted permission, against procedures, to the overhaul valve. (-PSF: The outage was short.)
1	Wolf Creek 1	c	(Cause) Electrician opened valve, as instructed, (it was not tagged out). (-PSF: The outage was short.)
1	Wolf Creek 1	d	+(Recovery) Relief crew Supervising Operator diagnosed the LOCA (ending the LOCA). (+PSF: He was one of the augmented crew.)
2	Catawba 2	a	(Cause) Preventive maintenance program failed to detect corrosion on the resistor.
2	Catawba 2	b	-(Recovery) (-PSF: Procedure inadequacy caused failure of the initial recovery attempts.)
3	Maine Yankee	a	No human action was identified.
4	Arkansas Nuclear 1	a	(Cause) Failure to follow good design practice in 1995 (no redundancy) allowed a single component (sensing probe) failure to cause a pump failure.
4	Arkansas Nuclear 1	b	(Cause) Improper equipment specifications allowed the use of an inappropriate signal limiter in the MFWP controls.
4	Arkansas Nuclear 1	c	(Cause) A cotter pin was installed inadequately. (PSFs: Design, personal work practices, procedures, and documents not followed correctly.)
4	Arkansas Nuclear 1	d	(Cause) Failure to act on industry information on the same failure mechanism.
4	Arkansas Nuclear 1	e	+(Recovery) Overall response to the event was excellent.
4	Arkansas Nuclear 1	f	-(Recovery) (-PSFs: Workarounds, Safety Parameter Display System)

Event ID No.	Plant Name	Action ID	Human Action (Cause and Recovery) Description
5	Oconee 3	a	(Cause) The licensee failed to implement proposed design modifications identified as early as 1980.
5	Oconee 3	b	(Cause) Maintainers improperly removed and reinstalled test caps. (-PSF: Procedures were not referenced in work package.)
5	Oconee 3	c	-(Recovery) Operator diagnosis was slow.
6	St. Lucie	a	(Cause) Engineering Dept. failed to communicate to I&C Dept. that I&C needed to change set point procedure.
7	Ft. Calhoun 1	a	(Cause 1 of 2 failures, Trip) (maintenance) Failure to use jumper caused instrumentation failure that caused turbine and reactor trips.
7	Ft. Calhoun 1	b	(Cause of 1 of 2 failures, Safety Valve) Lack of knowledge that temperature affects calibration caused miscalibration of Safety Valve.
8	Oconee 2	a	No human action was identified.
9	Oconee 2	a	(Cause) Use of an inadequate procedure for switchyard battery replacement resulted in a lockout of the switchyard and a LOOP.
10	Robinson 2	a	(Cause) A junction box with a drain hole had been inadvertently rotated so that water would not drain out.
11	Wolf Creek 1	a	(Cause) Operators were directed to align the ESW by memory; they did so incorrectly.
12	Sequoya 1&2	a	(Cause) A breaker test was performed incorrectly, which caused a flashover. (-PSF: procedures inadequate; they did not rule out pumping the breaker.)
12	Sequoya 1&2	b	-(Recovery) Recovery was delayed (-PSFs: low staffing and failure to follow procedures).
13	Turkey Point 3&4	a	(Cause of 1 of 2 failures) Use of a procedure during LOOP that was not designed for use during LOOP caused trip of EDG.
14	Catawba 1&2	a	(Cause) A technician failed to increase torque switch setting (-PSFs: They were not labeled on the equipment or in procedures).
15	Haddam Neck	a	(Cause) The process (communications) used to coordinate calculations led to a calculation error of HPSH, which led to RHR pump being inoperable.
16	Haddam Neck	a	(Cause of 1 of 4 test failures) During maintenance a part was not replaced correctly, causing failure of a circuit breaker test.
16	Haddam Neck	b	(Cause of 1 of 4 test failures) A set point was set too low, causing the failure of a test of a CVCS Relief Valve.

Event ID No.	Plant Name	Action ID	Human Action (Cause and Recovery) Description
16	Haddam Neck	c	(Cause) The manufacturer and licensee used a lubricant instead of a sealant. (-PSF: Manufacturer failed to tell licensee of a change.)
17	La Salle	a	(Cause of 1 of 3 failures) Due to inadequate preventive maintenance, water collected in a duct, causing a short, which tripped the transformer, causing a LOOP.
17	La Salle	b	(Cause of 1 of 3 failures) Due to lack of maintenance, a leak existed in the air supply of the actuator of a Safety Relief Valve, causing its failure to open.
17	La Salle	c	(Cause of 1 of 3 failures) During an inspection, the maintainer failed to inspect the windings of a motor. They were dirty, which caused degradation and a short.
17	La Salle	d	+(Recovery) The Operators generally performed well during recovery.
18	Perry 1	a	(Cause) Maintainers failed to clean up filter material; it entered the suppression pool, clogging strainers.
18	Perry 1	b	(Cause) The licensee cleaned *some* strainers, but failed to clean *all* of the strainers; so so some strainers remained clogged.
19	St. Lucie	a	(Cause) Improper installation of disc guides caused both PORVs to be unavailable.
19	St. Lucie	b	(Cause) Post maintenance testing was incomplete, and failed to detect the disc problem.
20	McGuire 2	a	(Cause) The jumpers had not been examined or changed since 1987. (Corrective action included tests during each outage.)
20	McGuire 2	b	(Cause) Vendor had not yet told licensee of a need to change installation instructions for guide rods on MSIV.
21	Waterford 3	a	+(Recovery) Operators followed procedures correctly.
21	Waterford 3	b	-(Recovery) The response of the fire brigade was slow. (-PSFs: training and procedures.)
22	Millstone Pt. 2	a	(Cause) Contractor evaluation failed to recognize the valves' susceptibility to pressure locking.
23	Seabrook 1	a	(Cause) Maintenance procedure not mention need for dial indicator. (-PSF: Skill of the craft, documentation, communication.)
24	Oyster Creek	a	No human action was identified.
25	Haddam Neck	a	(Cause of 1 of 2 failures, PORV) Operators failed to notice moisture indicator. (-PSFs: (Apparently) indication was hard to see. Procedures inadequate.)

Event ID No.	Plant Name	Action ID	Human Action (Cause and Recovery) Description
25	Haddam Neck	b	(Cause of 1 of 2 failures, DGs) Due to inadequate maintenance, a cooling fan had failed. Thus both DGs failed during a test under unprecedented conditions.
25	Haddam Neck	c	+(Recovery) Detection of Diesel vulnerability occurred during a test.
26	Arkansas Nuclear 2	a	(Cause) Human error during the design of a plant modification on a EFW Valve.
26	Arkansas Nuclear 2	b	(Cause) The design review did not catch the error, because the depth of the review was inadequate.
26	Arkansas Nuclear 2	c	+(Recovery) This design error was discovered while validating procedures on the training simulator.
27	Quad Cities	a	(Cause) Prior failure to detect original design deficiency. (PSF: Drawings do not show internal logic; this significantly hindered detection of the design deficiency.)
27	Quad Cities	b	+(Recovery) The design deficiency was discovered during a test.
28	Beaver Valley 1	a	(Cause) During a test, maintainers inadvertently made a connection via their multimeter that caused the LOOP.
29	Prairie Island 1&2	a	+(Recovery) +PSF: The shift manager augmented the plant staff.
30	Arkansas Nuclear 1	a	(Cause) Maintainer coupled motor to pump incorrectly. (-PSF: Maintenance procedures omitted material once considered "skill of the craft.")
31	Palo Verde 2	a	+(Recovery) A controlled cooldown was conducted in accordance with procedures.
31	Palo Verde 2	b	-(Recovery) Diagnosis of SGTR was delayed for one hour. -PSFs: Procedures (EOPs and FRP); Indicator inaccurate; training simulator lacked fidelity.
32	Cook 1	a	(Cause) Technicians method was incorrect, therefore they miscalibrated relays. (-PSF: Significant time had elapsed since training. There was no requalification program.)
33	Salem 1&2	a	No human action was identified.
34	Robinson 2	a	(Cause) Plastic material used during welding entered the Safety Injection piping.
35	Oconee 1,2&3	a	No human action was identified.
36	Com. Peak 1	a	No human action was identified.
37	Zion 2	a	No human action was identified.
38	Arkansas Nuclear 1	a	No human action was identified.

Event ID No.	Plant Name	Action ID	Human Action (Cause and Recovery) Description
39	Sequoya 1&2	a	(Cause) An inappropriate testing method caused an internal fault in a switchyard breaker.
39	Sequoya 1&2	b	(Cause) The Operations crew approved the inappropriate testing method. (-PSF: Operations crew staffing level was low.)
39	Sequoya 1&2	c	+(Recovery) Operators promptly diagnosed the plant conditions and took actions to stabilize the plant.
39	Sequoya 1&2	d	-(Recovery) Securing the secondary side was delayed. (-PSF: Low staffing level.)
39	Sequoya 1&2	e	-(Recovery) Unit 2 Operators failed to follow procedures, causing both CCPs to be removed from service.
40	River Bend 1	a	(Cause of 1 of 4 failures) Deficiencies in maintenance guidance caused two transmitters to be set at minimum damping.
40	River Bend 1	b	(Cause of 1 of 4 failures) Common mode calibration inaccuracies in calibrating reverse power relays.
40	River Bend 1	c	(Cause of 1 of 4 failures) Parts supplied by vendor were defective, causing corrosion.
40	River Bend 1	d	-(Recovery from failure of generator trip) Operator response was suboptimal. (-PSF: Procedures did not provide for positive verification of reverse power conditions.)
40	River Bend 1	f	+(Recovery) Operator actions were correctly prioritized throughout the event.
40	River Bend 1	e	-(Recovery from slow transfer of station load) Operator response was suboptimal. (-PSF: Training simulator and training were incorrect.)
41	Crystal River 3	a	(Cause of 1 of 2 failures) During post-maintenance troubleshooting, electricians made an incorrect connection.
41	Crystal River 3	b	(Cause of 1 of 2 failures) Failure of corrective action program: A similar failure occurred 5 months earlier.
41	Crystal River 3	c	(Cause of 1 of 2 failures) Licensee postponed repair of 1 gph leak on EDG (which became a 2 gpm leak when the EDG was started and loaded).
42	Byron 1	a	No human action was identified. (Caulking was thin and failed to prevent water leakage.)
43	St. Lucie 2	a	No human action was identified. ("The design ... contributed to the failure.")
44	Callaway	a	(Cause) Failure to take adequate corrective action from 12 previous similar annunciator failures in 9 years.

Event ID No.	Plant Name	Action ID	Human Action (Cause and Recovery) Description
44	Callaway	b	(Cause) I&C technicians caused a short, which caused the loss of control room annunciators. (-PSFs: Supervision and communications.)
44	Callaway	c	(Cause) Operators continued with plant activities, even though 164 annunciators were inoperable. (-PSFs: Knowledge, training, lack of use of procedures, staffing.)
44	Callaway	d	-(Recovery) Recovery was delayed. (-PSFs: Operator and management knowledge, staffing, communications, indications, design.)
45	Calvert Cliffs 2	a	(Cause) Contents in a safety evaluation were not incorporated into a modification procedure. (-PSF: design review, communications.)
45	Calvert Cliffs 2	b	(Cause) Testers assumed incorrectly, and did not verify, that contacts were functional. (-PSF: The modification process did not require verification.)
45	Calvert Cliffs 2	c	(Cause) Electrical leads were not taped, and minor movement of the door most likely resulted in their contact, causing the ground.
46	Limerick 1	a	(Cause) An incorrect engineering assessment 6 months earlier allowed the Safety Relief Valve to remain in service.
46	Limerick 1	b	(Cause) An incomplete cleaning program allowed foreign material to remain in the Suppression Pool.
47	South Texas 1	a	(Cause) A contractor clogged with paint a hole for a metering rod. (-PSF: procedures, testing, briefing, lessons learned, communications, defined responsibility.)
47	South Texas 1	b	(Cause) Pump was reassembled with defective parts and returned to service. (-PSF: Replacement parts were unavailable.)
47	South Texas 1	c	(Cause) Inappropriate closing of steam trap bypass valve caused water in steam supply line.
48	Pt. Beach	a	(Cause) Inappropriate closing of steam trap bypass valve caused water in steam supply line.
48	Pt. Beach	b	(Cause) Post-maintenance testing failed to include inspection for interference while rotating the generator

Table 3: Job Functions and Performance Shaping Factors for Human Actions.

Explanation: "Event ID No." is the identification number for the event as used in Tables 1 & 2."Action ID" is a sequential identification letter for each human action identified for each event as used in Table 2.1. "Job Function" classifies job functions: (1) Operations, (2) Maintenance, (3) Electrical, I&C, (4) Engineering or Design, (5) Vendor or Contractor, (6) Training, (7) Management Specified Explicitly, (Blank) Other; unspecified management; not known; not applicable. Legend for Performance Shaping Factors (PSF) 1 = positive influence. 2 = negative influence. Blank = no effect; not known.

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
1	Wolf Creek 1	a	1		2				1	2		2		
1	Wolf Creek 1	b	1		2				1	2				
1	Wolf Creek 1	c	3										2	
1	Wolf Creek 1	d	1				1		1					
2	Catawba 2	a	2											
2	Catawba 2	b	3	2										
3	Maine Yankee	a												
4	Arkansas Nuclear 1	a	4											2
4	Arkansas Nuclear 1	b	4											

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
4	Arkansas Nuclear 1	c	2	2	2									2
4	Arkansas Nuclear 1	d												
4	Arkansas Nuclear 1	e												
4	Arkansas Nuclear 1	f	1					2		2				
5	Oconee 3	a									2			
5	Oconee 3	b	2		2									
5	Oconee 3	c	1				1							
6	St. Lucie	a	4									2		
7	Ft. Calhoun 1	a	3											
7	Ft. Calhoun 1	b	4				2							
8	Oconee 2	a												
9	Oconee 2	a	3	2										
10	Robinson 2	a												
11	Wolf Creek 1	a	1		2									
12	Sequoya 1&2	a	3	2										
12	Sequoya 1&2	b	1		2				2					

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
13	Turkey Point 3&4	a		2										
14	Catawba 1&2	a	2	2				2						
15	Haddam Neck	a	4									2		
16	Haddam Neck	a	3											
16	Haddam Neck	b	2											
16	Haddam Neck	c	5									2		
17	La Salle	a	3											
17	La Salle	b	2											
17	La Salle	c	3											
17	La Salle	d	1	2		2	1		1			1		
18	Perry 1	a	2											
18	Perry 1	b	2											
19	St. Lucie	a	2											
19	St. Lucie	b	2											
20	McGuire 2	a												
20	McGuire 2	b	5										2	
21	Waterford 3	a	1											
21	Waterford 3	b	1	2		2		2						

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
22	Millstone Pt. 2	a	5											
23	Seabrook 1	a	2	2			2						2	
24	Oyster Creek	a												
25	Haddam Neck	a	1	2				2						
25	Haddam Neck	b	2											
25	Haddam Neck	c												
26	Arkansas Nuclear 2	a	4				2							
26	Arkansas Nuclear 2	b	4											
26	Arkansas Nuclear 2	c	6											
27	Quad Cities	a	4											
27	Quad Cities	b												
28	Beaver Valley 1	a	2											
29	Prairie Island 1&2	a	7						1					
30	Arkansas Nuclear 1	a	2	2										

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
31	Palo Verde 2	a	1		1									
31	Palo Verde 2	b	1	2		2		2		2				
32	Cook 1	a	3	2		2								
33	Salem 1&2	a												
34	Robinson 2	a	2											
35	Oconee 1,2&3	a												
36	Com. Peak 1	a												
37	Zion 2	a												
38	Arkansas Nuclear 1	a												
39	Sequoya 1&2	a							2					
39	Sequoya 1&2	b	1						2					
39	Sequoya 1&2	c	1		1		1							
39	Sequoya 1&2	d	1						2					
39	Sequoya 1&2	e	1		2				2					

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
40	River Bend 1	a		2										
40	River Bend 1	b	3											
40	River Bend 1	c	5											
40	River Bend 1	d	1	2										
40	River Bend 1	e	1			2								
40	River Bend 1	f	1				1							
41	Crystal River 3	a	3											
41	Crystal River 3	b									2			
41	Crystal River 3	c												
42	Byron 1	a												
43	St. Lucie 2	a												
44	Callaway	a									2			
44	Callaway	b	3									2		
44	Callaway	c	1		2	2	2		2					
44	Callaway	d	7				2	2	2			2		

				PSF										
Event No.	Plant Name	Act ID	JobFunct	Proc. Quality	Proc Use	Train	Knowledge	MMI	Staffing Level	Workload	Corrective Action	Commun	Work Controls	Design Process
45	Calvert Cliffs 2	a	4	2								2		2
45	Calvert Cliffs 2	b											2	
45	Calvert Cliffs 2	c	3											
46	Limerick 1	a	4											
46	Limerick 1	b	2											
47	South Texas 1	a	5	2							2	2		
47	South Texas 1	b												
47	South Texas 1	c	2											
48	Pt. Beach	a	3											
48	Pt. Beach	b	3											

<u>Table 4: Performance Shaping Factors Associated With Causes of Equipment Failures</u>.

Performance Shaping Factor	Number of Causes of Events
Procedure Quality	12
Communications	7
Procedure Use	6
Knowledge	4
Work Control	4
Corrective Action	4
Staffing Level	3
Design Practices	3
Training	2
Man/Machine Interface	2
Workload	2

<u>Table 5: Job Functions Associated With Causes of Equipment Failures</u>.

Job Function	Number of Causes of Events
Maintenance	17
Electrical and I&C	14
Engineering or Design	10
Operations	6
Vendor or Contractor	5
Training	0
Management specified explicitly	0
Other, Not Known	12

ATTACHMENT III

Risk Importance of Human Performance to Plant Safety

NUREG-0700, Rev. 2 Project
BNL Report W6546-T1-2-10/99

NRC JCN W-6546

Risk Importance of Human Performance to Plant Safety

Prepared for:

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

Prepared by:

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Department of Advanced Technology
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October 22, 1999

PREFACE

This draft report has been prepared by Brookhaven National Laboratory for the Division of Systems Technology of the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research. It is submitted as part of the requirements of the project *The Development oh Human-System Interface Design Review Guidance for NUREG-0700, Revision 2* (JCN W6546), specifically for Task 1,Human Performance and Risk. The NRC Project Manager is Jerry Wachtel (301 415-6498; jxw4@nrc.gov) and the BNL Principal Investigator is John O'Hara (516 344-3638; ohara@bnl.gov).

CONTENTS

List of Figures

List of Tables

1. INTRODUCTION

Nuclear power plant (NPP) safety, also called "safe operation of the plant," is a general term used to denote the technical safety objective as articulated by the International Atomic Energy Agency (IAEA, 1988):

> To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.

To ensure plant safety requires "defense in depth." Defense in depth includes the use of multiple barriers to prevent the release of radioactive materials and uses a variety of programs to ensure the integrity of barriers and related systems [a detailed discussion of this approach is provided in the IAEA basic safety principles (IAEA, 1988)]. These programs include, among others, conservative design, quality assurance, administrative controls, safety reviews, personnel qualification and training, test and maintenance, safety culture, and human factors.

Human factors plays a significant role in supporting plant safety and providing defense in depth. IAEA states:

> One of the most important lessons of abnormal events, ranging from minor incidents to serious accidents, is that they have so often been the result of incorrect human actions. Frequently such events have occurred when plant personnel did not recognize the safety significance of their actions, when they violated procedures, when they were unaware of conditions of the plant, were misled by incomplete data or incorrect mind set, or did not fully understand the plant in their charge. (p. 19)

Thus "human factors" was established as an underlying technical principle that is essential to the successful application of safety technology for NPPs. The principle states:

> Personnel engaged in activities bearing on nuclear power plant safety are trained and qualified to perform their duties. The possibility of human error in nuclear power plant operation is taken into account by facilitating correct decisions by operators and inhibiting wrong decisions, and by providing means for detecting and correcting or compensating for error. (p. 19)

Further, "...continued knowledge and understanding of the status of the plant on the part of operating staff is a vital component of defense in depth." This conclusion led to the following "safety principle":

> Parameters to be monitored in the CR (control room) are selected, and their displays are arranged to ensure that operators have clear and unambiguous indications of the status of plant conditions important to safety, especially for the purpose of identifying and diagnosing the automatic actuation and operation of a safety system or the degradation of defense in depth. (p. 43)

Similarly, the National Academy of Sciences reported that one of the first insights from studies of the Three Mile Island (TMI) accident was that errors resulting from human factors deficiencies in the control room (CR) were a significant contributing factor to NPP incidents and accidents (Moray and Huey, 1988). The errors at TMI were due to several factors, deficient training, inadequate procedures, and a poorly designed control room that provided inadequate resources for monitoring the basic safety parameters of plant functioning (Kemeny, 1979). Thus, the importance of human factors and human performance to NPP safety is widely acknowledged.

In response to the accident at TMI and subsequent investigations, the U.S. Nuclear Regulatory Commission (NRC) developed an action plan to address safety-significant deficiencies in commercial NPPs (NRC, 1980a and 1980b). From a human performance perspective, these included the conduct of control room design reviews, installation of safety parameter display systems, development of symptom-based emergency procedures, improved training, enhanced operator licensing, increased licensed operator staffing, and limitations on working hours.

In addition, the NRC initiated a formal human factors research program to focus on issues for which there were uncertainties in the scientific data needed to support regulation. An example of one such issue was the introduction into CRs and local control stations of advanced, computer-based HSI technology which was not used in TMI-era NPPs.

Research continues today to address human performance issues that are identified as important to the safe operation of nuclear power plants.

2. OBJECTIVES

The overall objective of this effort was to examine methodologies used to assess the risk importance of human performance issues.

3. METHOD

A brief review was performed of numerous studies that have examined the risk associated with human performance. The main characteristics of these studies is presented in Table 1. The studies are described with respect to the following attributes:

- quantitatively or qualitative study
- the analysis tool that was used
- the main objective of the study
- the aspect of human performance that was varied in the study
- the outcome measure
- whether the study was generic or plant specific
- whether human performance was the main focus of the study, and
- whether the study showed that human performance, in general, was a significant contributor to overall plant risk.

Appendix A contains a description of each study reviewed.

4. RESULTS

The importance of human performance, and the factors that influence it, on plant safety is internationally recognized, as noted in the introduction. This recognition stems from findings of investigations of significant events such as TMI and Chernobyl, analyses of less significant events such as those described in Licensee Event Reports (LERs), analyses of plant operating experience, numerous PRAs performed on NPPs, and the results of research investigations of how human performance variations impact plant performance.

Figure 1 illustrates the overall picture of how public health and safety can be affected by civilian nuclear power plant operation. This figure was developed as part of the NRC-Industry workshop on risk-informed regulation held in September, 1998. It illustrates three strategic performance areas: reactor safety, radiation safety, and safeguards. Under each strategic performance area are listed cornerstones that, when properly addressed, ensure the achievement of that strategic performance area. The first two cornerstones under reactor safety are initiating events and mitigation systems. These are two key items which determine core damage frequency (CDF) at a given plant. As illustrated, certain aspects of plant performance are cross-cutting issues that affect both CDF and all of the cornerstones, namely: human performance, safety conscious work environment, and problem identification & resolution.

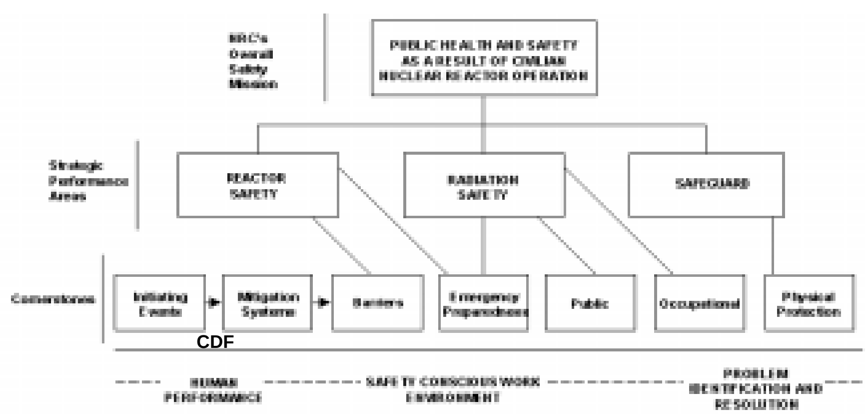

Figure 1. Public Health and Safety as a Result of Civilian Nuclear Reactor Operation

Figure 2 provides a more detailed breakdown of the many ways in which human performance impacts CDF. That is CDF for a given plant design is quantitatively determined by initiating event frequency, human errors, and the equipment reliability of the mitigating systems. Figure 2 then shows how each of these three factors is influenced by the various aspects of human performance, such as: training, procedures, human-systems interface (HSI), staffing and qualifications, and organizational factors. Besides affecting risk through core damage frequency, human performance also impacts the other two cornerstones of reactor safety, namely, barriers and emergency preparedness. The most important aspect related to barriers is the effect on primary containment integrity. The impacts of human performance on barriers and emergency preparedness is through mechanisms similar to those shown in Figure 2.

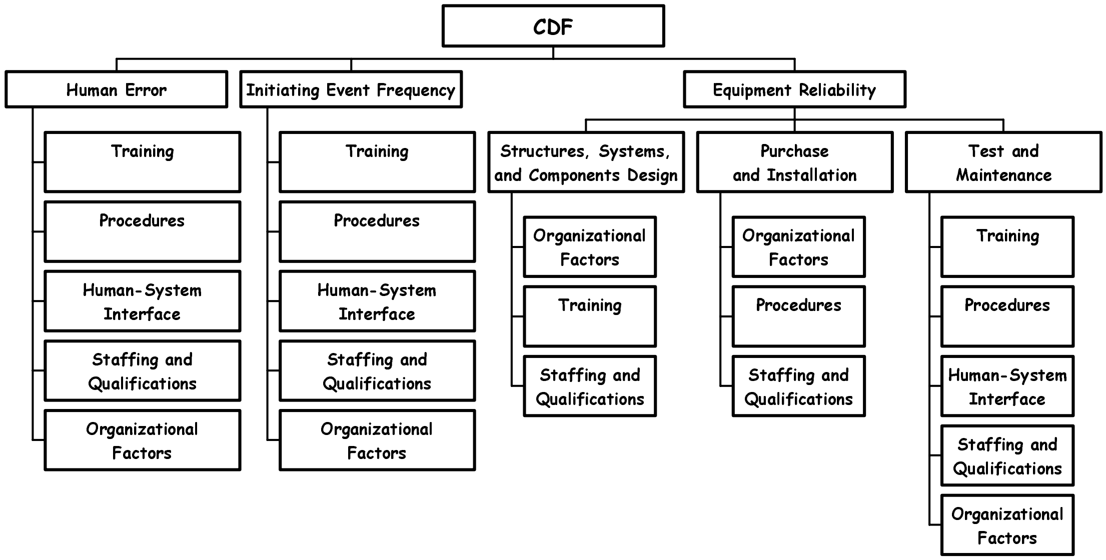

Figure 2. CDF Can Be Impacted in Multiple Ways by Human Performance

Whil e
many studies generally illustrate the effects of human performance on safety in a qualitative way, numerous investigations have also examined the effects of human performance on plant risk quantitatively. As noted in Section 3 above, Appendix A provides a summary of the studies reviewed. The last column of Table 1 indicates whether each study shows the overall risk importance of human performance in a quantitative manner. Eleven such studies have been reviewed and are included in the Table. Some of these are discussed briefly here.

The risk sensitivities studies (NUREG/CR-5319 and -5527) show the variation in the risk parameters (CDF and accident sequence frequency) due to hypothetical changes in human error probabilities (HEPs). For Oconee, the study shows significant sensitivity of the risk parameters to changes in HEPs, in both the increase and the decrease directions. Even relatively small changes in HEPs (a factor of 3 to 10 times) resulted in notable changes in core melt frequency (CMF). A decrease of HEPs by a factor of two reduced CMF by five times. Larger changes in HEPs (26 times up or down) resulted in CMF changing by four orders of magnitude (10 -6 to 10 -2 core melt events per year). For LaSalle, the study showed sensitivity of the risk parameters to changes in HEPs, in both the increase and the decrease directions. Changes in HEPs (29 times up or down) resulted in CMF changing by a factor of 3.5 in the decrease direction and a factor of 10 in the increase direction. Most of this change in CMF occurred with relatively small changes in HEPs (a factor 5 times).

An earlier study (NUREG/CR-3385) measured the risk importance (using a risk achievement worth-RAW measure) of human actions and plant systems at four NPPs. For each of the four plants, the RAW of the human actions is comparable to that of an important safety system (e.g., emergency core cooling injection or the emergency power system).

BNL Technical Report A-3702-10-89 examined the sensitivity of risk to MOVs (a significant component that appears in many safety systems) and human errors and found that the sensitivity to human errors was equal to or greater than that of the MOVs.

NUREG-1560 summarizes perspectives from the 75 IPEs submitted to the NRC. These IPEs cover 108 nuclear power plants units. The IPEs showed that human error can be a significant contributor to core damage frequency (CDF) and that correct human actions can substantially reduce the overall CDF. Tables are provided in Appendix A, page A2, that give the contribution to CDF of individual human errors.

In addition to these general studies, several studies have demonstrated the importance of human action to plant risk by way of varying some factor that effects human performance, such as the design of local control station design (NUREG/CR-5572), alarm design (BNL A-3967-89-2), and procedure design (NUREG/CR-5458). In each case, the effects of the design characteristic on risk was quantified in a multi-step process whereby the effects of the design variation on human error probabilities (HEPs) was determined, new HEP estimates were established, and the effect on plant risk was quantified by rerunning a PRA (see a description of these studies in Appendix A).

In conclusion, these studies demonstrate that human performance is a significant contributor to plant risk. Degraded human performance can increase risk and improved performance can lower risk. Further, human performance enters into risk in many areas of plant design, operation, testing, and maintenance.

Table 1 Summary of Studies Reviewed

Reports	Study Characteristics							
	Quant./ Qualit.	**Tool**	**Objective**	**Item Varied**	**Outcome Measure**	**General Specific**	**HP is Focus**	**Shows Quantitative Risk Import. of HP**
IPE Pgm Perspectives NUREG-1560	Quant	PRA	IPE Summ	None		G/S	N	Y
Precursor Study NUREG/CR-4674	Quant	PRA	Event analysis	None	Conditional CDP	S	N	N
ECCS switchover NUREG/CR-6432	Quant	PRA	GI - VI analysis	Design & CDF	Cost	G/S	N	N
Risk Sensitivity-Oconee NUREG/CR-5319	Quant	PRA	HE Sensitiv.	HEPs	CDF & ASF	S	Y	Y
Risk Sensitivity-LaSalle BNL	Quant	PRA	HE Sensitiv.	HEPs	CDF & ASF	S	Y	Y
Human Variability BNL L-1160 (1)	Quant	PRA	HE Var. & Sens.	HEPs	CDF	G/S	Y	Y
Comparison of Oc & LaSalle L-1160 (2)	Quant	PRA	HE Sensitiv	HEPs	CDF & ASF	G/S	Y	Y
Inadvert. Op. of MOVs E-2071 & E-3869	Quant	PRA - RIR	GI analysis	MOV f.r.- HEP/EOC	Risk Inc. Ratio	G	NA	Y
LCS Variation NUREG/CR-5572	Quant	PRA VI	HSI risk effect	HSI Design HEPs	CMF	G/S	Y	Y
Alarm Variation BNL A-3967-89-2	Quant	PRA VI	HSI risk effect	HSI Design HEPs	CMF	G/S	Y	Y
Operating procedure Prog. NUREG/CR-5458	Quant	PRA VI	Proced. Effect	Procedure Design	CMF	G/S	Y	Y
Shift Staffing Risk Ass'ment BNL W6152-T9-0	Quant	PRA Task Anal.	Evaluate Staffing	Base vs. Tech Spec	CDF	G/S	Y	Y

Table 1 Summary of Studies Reviewed

Reports	Study Characteristics							
	Quant./ Qualit.	**Tool**	**Objective**	**Item Varied**	**Outcome Measure**	**General Specific**	**HP is Focus**	**Shows Quantitative Risk Import. of HP**
Alarm Variation Lorenzo & Burne (1985)	Quant	HRA	HSI risk effect	HSI Design	HEPs	S	Y	N
MSF Model Assessment NUREG/CR-3837	Quant	HRA	Validate MSF	MSF vs Exp Data	HEP	S	Y	N
HRA comp NUREG/CR-4835	Quant	HRA Sim	Compare HRA	HRA Method	HEP	S	Y	N
ASEP Assessment NUREG/CR-6355	Quant	HRA Sim	Evaluate ASEP	ASEP vs Sim Data	HEP	S	Y	N
THERP Assessment NUREG/CR-3309	Quant	HRA Sim	Evaluate THERP	THERP vs Sim Data	HEP	S	Y	N
Risk Imp Measures NUREG/CR-3385	Quant	Imp. measure	Imp. measure	None	Imp measures	G/S	N	Y
RIGs BNL Tech. Reports	Quant	Imp measure	Risk ranking	None		S	N	N
HP trends in Precursors L-1170 1/91	Quant	Data analysis	Trning analysis	None		G	Y	N
HEPs from LERs NUREG/CR-3519	Quant	Data analysis	HEP Quant.	None		G	Y	N
Shift Staffing Modeling Plot & Laughery	Quant	MicroSaint	Model staffing	Base vs Tech Spec	Task time Task Delay	S	Y	N
Hybrid Topic Evaluation BNL J6012-T2-6/96	Qual.	Mod. 50.59	Risk Eval of Issues	Issue vs Base case	USQ Checklist Evaluation	G	Y	N
Interface Mgt. Evaluation BNL W6546-2-5-9/97	Qual.	Mod. 50.59	Risk Eval of Issues	Issue vs Base case	USQ Checklist Evaluation	G	Y	N

Table 1 Summary of Studies Reviewed

Reports	Study Characteristics							
	Quant./ Qualit.	**Tool**	**Objective**	**Item Varied**	**Outcome Measure**	**General Specific**	**HP is Focus**	**Shows Quantitative Risk Import. of HP**
AEOD annual reports NUREG-1272 series	Qual	Data analysis	Op exp Summ.	None		G	N	N
HP during Op events NUREG/CR-5953	Qual	Causal factor analysis	Eval. HF in events	None	Factors that affect HP	G/S	Y	Y
CBPs APG Rpt. 35	Qual	Human Rel Model	Eval. CBPs	CBP vs PBP	Prob. Human Error	G	Y	N

Appendix A

Documents Reviewed

NUREG-1272 Series
NRC Office of Analysis and Evaluation of Operational Data (AEOD)

This series of NUREGs (issued in volumes) has been published since 1984 and comprises the annual report of the activities of AEOD. The documents contain a wealth of operating data and experience for the year that is covered. Typical topics covered in the report are: performance indicator data (e.g., reactor scrams), abnormal occurrences, precursor events, special AEOD studies, LER data, incident response and investigations, and other plant data.

Some of the data is useful for making determinations about human performance in NPPs. Examples from Volume 9 are: a summary of AEOD/T94-02, Review of Mispositioned Equipment Events; a summary of AEOD/T94-03, Computer-based Digital System Failures; a summary of AEOD/E95-01, Operating Events with Inappropriate Bypass or Defeat of Engineered Safety Features; and tables with LER System personnel error cause code data.

NUREG-1560
Individual Plant Examination Program: Perspectives in Reactor Safety and Plant Performance

This NUREG summarizes perspectives from the 75 Individual Plant Examinations (IPEs) submitted to the NRC. These IPEs cover 108 nuclear power plant units. With regard to human actions, the document provides a chapter (Chapter 5) and tables that list the human actions that are important to severe accident prevention and mitigation. The IPEs showed that human error can be a significant contributor to core damage frequency (CDF) and that correct human actions can substantially reduce the overall CDF.

This document contains tables that list the important human actions for boiling water reactors (BWRs) and pressurized water reactors (PWRs). Importance was as reported by the licensees in their IPE submittals. The method of determination was therefore not the same across plants and was not always clearly documented in the submittals. The tables are rank-ordered by the percent of plants that found a particular action important. As an example, BWRs had four actions that were "important in at least 50%" of the plants. PWRs had three such actions. See the table below for a summary of these actions. Also included is the typical contribution to CDF of the important human actions. If a threshold of "important at 10%" of the plants is used, then the numbers of important human actions are 15 for BWRs and 13 for PWRs. Discussions of each of the important actions are also provided, which include the contribution to CDF of the human action.

This information can be useful because it covers all of the plants in the US. Review of the individual important human actions can identify the human performance aspects and performance shaping factors that could impact the correct execution of the important actions. Knowledge of the important human actions can be useful in the development of and the prioritization of regulatory activities.

Important BWR Human Actions	**% of BWR IPEs with HA as Important**	**Typical % of CDF contribution for HA**
Manual depressurization	80	1% to 45%
containment venting	55	contribution to containment failure and early release
align containment or SP cooling	55	1% to 5% of CDF plus contribution to containment failure and early release
Initiate SLC	70	1% to 3%
Important PWR Human Actions	**% of BWR IPEs with HA as Important**	**Typical % of CDF contribution for HA**
Switchover to Recirculation	70	<1% to 16%
Feed-and-bleed	60	<1% to 10%
Depressurization and cooldown	50	<1% to 7%

NUREG/CR-3309[******]
A Simulator-Based Study of Human Errors in NPP Control Room Tasks

The purpose of the study was to empirically establish error rates for control selection and operation during the performance of proceduralized tasks using simulated events. The results were compared with HEPs presented in THERP (NUREG/CR-1278).

Data on 128 operators using two simulators was collected during training exercises. The data were collected using the Performance Measurement System which records control manipulations and plant parameters. These records were compared with a list of procedural task elements. Actions not taken were considered errors of omission. For each type of error observed, the number of errors that occurred was divided by the number of opportunities for such an error. This provided the Observed Error Rate (OER). The OERs were in general agreement with the adjusted HEPs from THERP (adjusted for PSFs).

Some additional insights were:

- THERP may tend to overestimate dependency effects.
- No significant differences in performance were observed between novices (trainees) and experienced operators.
- Operators did better with symptom-based procedures when compared with event-based procedures.

The authors concluded that the use of data collected during simulator-based training was valuable. The automatically collected data was enhanced by operator interviews following training exercises.

[******]Beare, A. et al. (1984). A simulator-based study of human errors in NPP control room tasks (NUREG/CR-3309). Washington, D.C.: U.S. Nuclear Regulatory Commission.

NUREG/CR-3385
Measures of Risk Importance and Their Applications, July 1983

This document, while old, is still valuable, since it defined two of the key risk importance measures that are still in prominent use today: risk reduction worth (RRW) and risk achievement worth (RAW).

The RRW of an item measures the reduction of risk (e.g., core melt frequency) from the base case, when the item is assumed to have a zero failure rate. The RRW is useful for prioritizing plant features/ activities with regard to improvements to reduce risk. The RAW of an item measures the increase in risk from the base case when it is assumed to have a failure probability of 1.0 (i.e., it will fail). The RAW is useful for prioritizing plant features/activities with regard to reliability assurance and maintenance activities. Many types of features can be evaluated using these measures, for example: safety functions, systems, components, human actions, surveillance tests, and maintenance activities.

This document also calculated RRW and RAW for many features of the four PRAs that were included in the NRC RSSMAP program. Included in the calculations were Risk Worths (both RRW and RAW) for human actions. For each of the four plants, the RAW of human actions (shown in the figures of the Executive Summary) is comparable to that of an important safety system (e.g., emergency core cooling injection or the emergency power system). Since this program and the related four PRAs are quite old and newer PRAs have been performed for those four plants, the specific results are not particularly valuable at this time.

NUREG/CR-3519
Human Error Probability Estimation Using Licensee Event Reports, July 1984

This report examined the possibility of using LERs as a data source in estimating HEPs for use in a PRA/HRA. The report developed a methodology, ran a few trial cases and determined that it was a practical, acceptable, and useful method. However, over the 14 years since the report was issued LERs have not generally been used by the PRA/HRA community as a method for developing HEPs. Thus, this report does not appear to be useful for this current project.

NUREG/CR-3837*

Multi-Sequential Failure Model: Evaluation of and Procedures for Human Error Dependency

The multi-sequential failure (MSF) model was described in NUREG/CR-2211 (1981). In this report, the method is evaluated for it practicality, usefulness, and acceptability. The MSF model is used to estimate the level of dependence in the failure of redundant components in a system due to human interactions such as maintenance, testing, and calibrations to cause common mode failures. Thus, the model can be used to estimate the probability of multiple human failures in a system.

Predictions of the model were in general agreement with the results of a small-scale experiment involving five undergraduate students conducting PC presented test, calibration, and maintenance tasks.

A potential limitation of the model was that it required data (although only a limited amount) on dependency. However, such data may not always be available.

The report provides documentation on the model development and procedures for its use. The MSF model is to be used in conjunction with other more general methods, such as THERP, to account for dependency.

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*Samanta, P. et al. (1985). Multi-sequential failure model: Evaluation of and procedures for human error dependency (NUREG/CR-3837).Washington, D.C.: U.S. Nuclear Regulatory Commission.

NUREG/CR-4674, 24-volume series
Precursors to Potential Severe Core Damage Accidents

This 24-volume series, together with two earlier NUREG/CRs (-2497 and -3591), provides an overview of precursors in the U.S. nuclear power industry from 1969 through 1994. The study involves a review of licensee event reports (LERs) of operational occurrences to identify and categorize precursors to core damage accidents. Such precursors are typically infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition in which core cooling wold not be adequate. The analysis determines a conditional probability of core damage, given the events which actually occurred. These conditional probabilities are then used to rank and trend the precursor events.

In addition to the trending and overview information, the documents also provide a description and analysis of each important precursor event. Some of the precursor events (typically about 50%) contain human errors. There may be one or several human errors in a given event. These reports do not analyze, quantify, or trend the associated human errors.

The reports could be useful as a source of actual human errors that have occurred in operational events. However, the effort to extract and use this information may be significant.

NUREG/CR-4835*
Comparison and Application of Quantitative Human Reliability Methods for the Risk Methods Integration and Evaluation Program

HRA methods were compared by applying them to specific accident sequences from the LaSalle PRA and comparing their results. Guidelines for use of the selected method would be developed.

From a list of 20 candidate methods, 13 were identified as appropriate for use in NPP PRAs, and they were grouped according to their methods (with some falling in more than one group):

Confusion Matrix
- Confusion matrix
- Dougherty (published in Oconee document, 1981)

Time reliability correlations
- Dougherty
- Human Cognitive Reliability (HCR)
- Operator Action Tree/Time Reliability Correlation (OAT/TRC)
- Sandia recovery model
- Technique for Human Error Rate Predictions (THERP)
- Woods (1986)

Expert Estimation
- Expert estimation
- Success Likelihood Index Method/Multi-Attribute Utility Decomposition (SLIM/MAUD)
- Socio-technical approach to assessing human reliability (STAHR)
- HCR

Simulation
- HCR
- Simulator Data
- Sandia recovery model

Composite Techniques dealing with component-level errors
- THERP
- Fullwood (1976)
- Sandia recovery model

Data were collected by a team consisting of an HRA expert, systems engineer, and HFE specialist. The data were collected at the LaSalle plant and simulator. Walkthroughs of accident scenarios were conducted with operators and operations experts.

Differences between method estimates were found, in part, due to detail of modeling and analysis familiarity with methods. No quantitative comparisons were performed, but the report provides qualitative comparisons between techniques along with the strengths and weaknesses of each.

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*Haney, L. et al. (1989). Comparison and application of quantitative human reliability methods for the risk methods integration and evaluation program (RMIEP) (NUREG/CR-4835). Washington, D.C.: U.S. Nuclear Regulatory Commission.

NUREG/CR-5319
Risk Sensitivity to Human Error, January 1989

This document consists of sensitivity evaluations to assess the impact of human errors on the internal risk parameters of the Oconee plant. The results show the variation in the risk parameters (core melt frequency and accident sequence frequency) due to hypothetical changes in human error probabilities (HEPs). Also provided are insights derived from the results, which highlight important areas for concentration of risk limitation efforts.

Some of the reasons for the study include:

- estimation of HEPs is uncertain and errors in HEP estimation can result in over- or under-estimation of risk in PRAs.
- many HEPs are based on generic data and the performance at a particular may be significantly different from the generic.
- during the operating life of a plant, there may be times when the performance of a crew (or crews) is significantly worse (or better) than shown in the PRA.
- the performance of the humans in the plant can be significantly affected in the aggregate by plant management, resulting in systematic changes in HEPs.

The results of the study show significant sensitivity of the risk parameters to changes in HEPs, in both the increase and decrease directions. Even relatively small changes in HEPs (a factor of 3 to 10 times) resulted in notable changes in core melt frequency (CMF). A decrease of HEPs by a factor of two reduced CMF by five times. Larger changes in HEPs (26 times up or down) resulted in CMF changing by four orders of magnitude (10-6 to 10-2 core melt events per year). (Later studies showed that Oconee was one of the most sensitive plants to human error.)

The study also identified the types of human actions that tended to drive the risk. These important areas were: during-accident actions, recovery actions, operations activities, and actions by both licensed reactor operators and by non-licensed auxiliary operators.

Another interesting point was that, while risk could be reduced by lowering HEPs, there was a limit after which no further improvement was obtained. This limit was based on the risk due to equipment failure probabilities.

This study was important because it showed the overall importance of human actions and the particular types of human activities that affected risk. These methods can be used in a variety of prioritization activities. They are useful for evaluating planned actions when uncertainties are involved and when one wants to see the limitations of the benefits of changes due to asymptotic effects on risk.

NUREG/CR-5458*
Value-Impact Assessment for a Candidate Operating Procedure Program

Operating procedures were found to have major substantive deficiencies; the purpose of this study was to determine whether it was cost effective to upgrade these procedures. A value impact (V-I) analysis was performed to make this determination. The V-I methodology used was described in NUREG/CR-3568 (Heaberlin et al., 1983). The alternatives considered the status quo and upgraded normal and abnormal operating procedures (which involved improving the technical accuracy and presentation of the procedures).

The potential reduction to CMF was estimated by first estimating the current contribution to CMF of procedure-related operational errors during normal and abnormal operations. Then, the contribution of improved procedures was estimated, and the difference between the two was calculated. The methodology consisted of 11 steps divided into three elements.

Element 1 was to estimate the contribution of procedures:

1. Identify the potentially affected transient initiating events in the reference PRA by expert judgement.

2. Identify the potentially affected operator actions (recovery actions were not included since they are governed by emergency procedures - outside the study scope).

3. Identify the affected accident sequences and cut sets (those involving the initiating events from 1 or an affected operator action from 2).

4. Estimate the current contribution of procedure-related operator errors to the affected transient initiating events (based on LER analysis: the frequency that an "inadequate task description" cause was identified in those LERs involving one of the "affected transient initiating events").

5. Evaluate the role of procedures in the operator actions/errors by expert judgement.

6. Calculate the procedure-related portion of the base-case affected parameter frequencies based on the results of steps 4 and 5.

Element 2 was to determine the contribution of the procedure upgrade program:

7. Estimate the reduction to the procedure-related portion of the affected parameters using expert judgement (quantified in terms of percentage reduction to the initial affected probability).

8. Calculate the adjusted-case affected parameter values.

9. Calculate the adjusted-case affected core-melt frequency.

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*Grant, T. et al. (1989). Value-impact assessment for a candidate operating procedure program (NUREG/ CR-5458). Washington, D.C.: U.S. Nuclear Regulatory Commission.

Element 3 was to calculate the reduction in CMF due to procedure improvements:

10. Calculate the estimated change in CMF.

11. Correlate the generic estimates to actual plants by grouping plants into three categories of procedure quality and using expert judgement to adjust the CMF for each of the categories.

The results are provided below.

Estimated Reduction in Core-Melt Frequency
Plant Category Based on Procedure Category

	Poor	Intermediate	Good
Best Estimate	2.8E-05	8.5E-06	2.8E-06
Upper Bound	3.4E-05	1.5E-05	9.1E-06
Lower Bound	1.7E-05	0	0

NUREG/CR-5527
Risk Sensitivity to Human Error in the LaSalle PRA, February 1990

This document consists of sensitivity evaluations to assess the impact of human errors on the internal risk parameters for a BWR plant (LaSalle). The results show the variation in the risk parameters (core melt frequency and accident sequence frequency) due to hypothetical changes in human error probabilities (HEPs). Also provided are insights derived from the results, which highlight important areas for concentration of risk limitation efforts.

The reasons for the study are similar to those for the Oconee study in NUREG/CR-5319 and include:

- estimation of HEPs is uncertain and errors in HEP estimation can result in over- or under-estimation of risk in PRAs;

- many HEPs are based on generic data and the performance at a particular may be significantly different from the generic;

- during the operating life of a plant, there may be times when the performance of a crew (or crews) is significantly worse (or better) than shown in the PRA; and

- the performance of the humans in the plant can be significantly affected in the aggregate by plant management, resulting in systematic changes in HEPs.

The results of the study show sensitivity of the risk parameters to changes in HEPs, in both the increase and the decrease directions. Changes in HEPs (29 times up or down) resulted in CMF changing by a factor of 3.5 in the decrease direction and a factor of 10 in the increase direction. Most of this change in CMF occurred with relatively small changes in HEPs (a factor 5 times).

As for Oconee, this study also identified that the types of human actions that tended to drive the risk were: during-accident actions, recovery actions, operations activities, and actions by both licensed reactor operators and non-licensed auxiliary operators.

LaSalle used simulator modeling and testing to assist in the quantification of main control room human errors. Interestingly, the human errors that included outside of the control room actions and that could not effectively be modeled on the simulator (e.g., recovery of offsite power) tended to dominate risk in the LaSalle PRA. Another notable point was that the LaSalle PRA included very few pre-accident errors, such as calibration and failure to restore valves after test and maintenance.

This study was important because it also showed the overall importance of human actions and the particular types of human activities that affected risk. These methods can be used in a variety of prioritization activities. They are useful for evaluating planned actions when uncertainties are involved and when one wants to see the limitations of the benefits of changes due to asymptotic effects on risk.

NUREG/CR-5572[*]
The Effects of Local Control Station Design Variation on Human Performance and Plant Risk

A human factors analysis was performed to assess how selected design variations to local control stations (LCSs) in nuclear power plants (NPPs) affect both human performance and plant risk. Modifications in the design of individual control panels and changes in their functional centralization were considered. The analysis methodology was accomplished in three stages. First, a list of LCS human engineering design deficiencies was developed using data collected from a variety of sources including visits to NPPs. From these data, a set of potential upgrades were defined to correct the deficiencies. Second, the effects of the upgrades on human error probabilities (HEPs) were determined using a computer-based methodology for soliciting expert judgement. Third, the HEPs were propagated through a plant probabilistic risk assessment, and new core melt frequencies were established. The results indicated that implementation of either type of upgrades would improve human performance and lower risk, although the effect of functional centralization on performance and risk was found to be greater than panel design.

The latter two stages are described more completely.

Human performance effects analysis methodology

The methodology utilized to estimate LCS effects on performance was subjective evaluation by a group of expert judges using a consensus approach. To facilitate the consensus judgments, the Success Likelihood Index Method/Multi-Attribute Utility Decomposition (SLIM/MAUD) approach was used. The group included three individuals with collective expertise in NPP operations, human factors, human performance evaluation, NPP systems engineering, NPP inspections, and probabilistic risk assessment. One of the judges had extensive experience with LCSs in connection with safe shutdown inspections.

The judges were briefed on the Oconee PRA, the LCS-related human errors, the emergency operating procedures and LCS-related activities, and the design of the Oconee LCSs. Nine alternative LCS-panel configurations were defined without reference to their presumed effects on human performance. The judges were asked to evaluate the effect that each of the nine LCS configurations would have on human performance according to four "performance shaping factors" (PSFs) as per SLIM/MAUD methodology. They were communications requirements, control panel configuration, training effectiveness, and procedural detail requirements. Each was rated on a nine-point scale. It was judged by the project staff that these four factors reasonably captured the major influences of LCS design characteristics on human performance.

Each LCS configuration was rated separately on each of the four factors. The judges discussed their evaluations, and when consensus was achieved, the ratings were recorded by a session facilitator on an IBM PC-based version of SLIM/MAUD. After all design options were evaluated, the four factors were given weights by the judges to indicate their relative importance using the SLIM/MAUD weighting procedure. The resultant weightings were: Communications Requirements, .29; Procedural Detail Requirements, .29; Control Panel Configuration, .22; and Training Effectiveness, .21.

A Success Likelihood Index (SLI) was computed for each of the design options by a weighted linear combination of the PSFs. The SLI values were linearly transformed to a unit scale where a lower score

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[*]O'Hara, J.,Ruger, C., Higgins, J., Luckas, W., and Crouch, D. (1990). *An evaluation of the effect of local control station design configurations on human performance and plant risk* (NUREG/CR-5572). Washington, D.C.: U.S. Nuclear Regulatory Commission.

indicated poorer performance. The SLI ratings for LCS panel configurations 1 through 9 were: .94, .79, .59, .70, .56, .32, .34, .16, and .00, respectively.

The computed SLIs were used to adjust the HEPs in the PRA. This was accomplished by establishing two calibration points to link the SLI scale to HEP values. One calibration point was the SLI value for LCS design configuration 5. Since this was identified as the existing Oconee design, its SLI (which was .56) was linked to the base case HEP values of the Oconee PRA. To scale all the other HEPs, a second calibration point was required. Establishment of that point was accomplished by increasing the base-case HEP by one order of magnitude (factor of 10). Thus, the worst possible SLI rating of 0.0 was linked with an order of magnitude increase in HEPs (stopping at a value of 1.0 if such a change would raise the HEP above 1.0). The HEPs for all of the remaining LCS design options were then rescaled proportionately according to their respective SLI.

To determine the maximum extent to which LCS variation impacted human performance, the LCS design options with the lowest and highest SLIs were compared. Transitioning from the LCS design judged poorest to the LCS design judged best resulted in a decrease of 95% in mean HEP (a human error reduction factor of approximately 20).

Overall, the panel design dimension had a smaller effect on human performance than did the functional centralization dimension with a span of .29 mean HEP in comparison to .46 respectively for changes from LCS configuration 1 to configuration 9.

Plant risk effects analysis

To determine whether LCS design variation, through its effects on HEPs, was risk significant, the Oconee PRA results were recalculated once for each LCS design configuration with its unique set of HEPs. The measure of plant risk selected was core melt frequency (CMF). The transition from the poorest to highest LCS design configuration resulted in a decrease in CMF of 4.82E-4 events/reactor-year (RY) (a decrease of 77%). The panel design factor had a smaller effect on plant risk than did the functional centralization factor with ranges of 1.70E-4 events/RY and 2.65E-4 events/RY, respectively.

The data analysis for both HEPs and CMF exhibited an interaction effect that is not discussed in this summary.

Note that this methodology was submitted to peer review in the publication of a journal article.

NUREG/CR-5953

Studies of Human Performance During Operating Events, 1990-1992, November 1992

In order to better evaluate the human factors influencing operator performance during safety significant operating events, the AEOD performed onsite analyses of 16 selected events. The onsite analyses focused on those factors that helped or hindered operator performance. Causal factors for operator performance were identified and categorized. Individual reports were prepared for each event. This NUREG/CR describes each of the 16 events and generic observations and conclusions derived from all 16 studies.

The 16 events were estimated to comprise about 25 to 33% of those events that significantly challenged operating crews during the 2 ½ year time period, and represented a wide variety of scenarios.

The factors influencing operator performance that were found to be generic were presented in two groups:

- Plant conditions that affected performance included: configuration control, work control, quality of maintenance, incomplete work packages, human-machine interface (poor layout, alarm overload, incorrect labels, poor valve position indication), and poor allocation of function (between automation and manual operation).

- Operator resources that affected the events included: poor situation awareness, teamwork among reactor operators, communications between operations and maintenance, role of the STA, need for improvements in procedures and training, shift structure, and stress from various factors (e.g., long time on duty, tight time schedules, potential consequences of actions, and early morning hours).

This study (and other similar studies) could be useful for identification of activities that should be included in a human performance plan, since it presents human performance problems or issues that contributed to safety significant events across the NPP industry. It could also help in the prioritization from a qualitative (but not quantitative) standpoint.

NUREG/CR-6355[*]

A Limited Assessment of the ASEP Human Reliability Analysis Procedure Using Simulator Estimation Results

The report provides an assessment of the Accident Sequence Evaluation Program (ASEP) human reliability analysis (HRA) procedure (Swain, 1987). The ASEP post-accident, post-diagnosis, nominal HRA procedure is assessed within the context of an individual's performance of critical tasks on the simulator portion of requalification examinations administered to NPP operators. It is noted that a limitation of their evaluation is that the ASEP procedure reflects error probabilities of an entire crew, while the exams are individual operators (thus no recovery based on feedback from other operators was possible).

Pass/Fail information for groups of simulator critical tasks (ISCTs) were used to determine failure rates (rates for individual tasks were too small to provide statistically meaningful results). These rates were compared to estimates using ASEP.

The comparison indicated that for small HEPs there is little conservatism, but for larger HEPs there was considerable conservatism in the ASEP estimates.

[*]Gore, B. et al. (1995). A limited assessment of the ASEP Human Reliability Analysis Procedure using simulator examination results (NUREG/CR-6355). Washington, D.C.: U.S. Nuclear Regulatory Commission.

NUREG/CR-6432

Estimated Net Value and Uncertainty for Automating ECCS Switchover at PWRs, February 1996

This study is related to Generic Safety Issue (GSI) 24, "Automatic ECCS to Recirculation" for PWRs. A detailed Value-Impact (or Cost-Benefit) Analysis was performed to address the central question of the GSI. This question is whether or not PWRs that currently rely on a manual system for Emergency Core Cooling System (ECCS) switchover to recirculation should be required to install an automatic system. The study included consideration of modifications to make the manual systems either semi-automatic or fully automatic (as defined in the study itself).

The study used PRA analyses to calculate core damage frequencies (CDFs) for the various cases under consideration. This required a detailed HRA of the operator actions associated with the ECCS switchover process for the three cases of manual, semi-automatic, and fully automatic. Approximately two dozen human actions were modeled for the study. The report presents the results of a quite extensive and detailed study. It found that the changeover to a semi-automatic or fully automatic system would reduce CDF; however, requiring current plants to make such a modification would not be cost-beneficial. If one considers the case of license renewal, then it appears that such a change out would be cost-beneficial.

Part of the study included a review of pertinent HRA methods (as documented in Appendices D and E), and a selection of the combination of methods to use for the study at hand. Appendix D discusses two earlier HRA methods reviews (NUREG/CR-4835 for the RMIEP project by Haney et al. and "Comparative Evaluation of Methods of Human Reliability Analysis," GRS-71, by A. D. Swain, April 1989). NUREG/CR-4835 first screened 20 methods and dropped eight fully qualitative ones from further review. They then evaluated 12 methods, namely:

1. Confusion Matrix (Potash and Stewart)
2. Dougherty
3. Expert estimation (Comer and Seaver)
4. Fullwood and Gilbert
5. Human Cognitive Reliability (HCR) Model (Hannaman and Spurgin)
6. Operator Action Tree (OAT) (Hall and Fragola)
7. Sandia Recovery Model (SRM) (Weston and Whitehead)
8. Simulator Data (Beare and Dorris)
9. Success Likelihood Index Method (SLIM) (Embrey)
10. Socio-Technical Assessment of Human Reliability (STAHR) (Phillips and Humphreys)
11. Technique for Human Error Rate Prediction (THERP) (Swain and Guttmann)
12. Woods

This document (NUREG/CR-6432) screened the above 12 methods and eliminated four as requiring excessive time and number of subject-matter experts (numbers 1, 3, 9, and 10). An evaluation of the remaining eight methods was performed, using criteria specific to the study’s requirements. The authors selected THERP (11) for procedural activities and SRM (7) for knowledge-based behavior.

This report is useful as it provides a sample method and analysis for a specific type of quantitative study that involves detailed operator actions in a plant. It also provides a survey and discussion of HRA methods.

APG Report No. 35*
Research in Computerized Emergency Procedures Systems to Enhance Reliability of NPP Operating Crews

Orvis and Spurgin (1996) have estimated that approximately 89 percent of crew errors occur during detection, diagnosis, and decision making, and 11 percent occur during response implementation. Orvis and Spurgin further estimated that most of the 89 percent is emergency operating procedure (EOP)-related, largely due to an inability of senior reactor operators (SROs) to properly follow the EOPs. Properly following the procedures is sometimes complicated by the necessity to track several EOPs or EOP branches simultaneously. Current symptom-based procedures, therefore, appear to place significant workload on operators.

Orvis and Spurgin (1996) evaluated computer-based procedures (CBPs) from a perspective of a cognitive reliability model of crew reliability. It should be noted that the analysis assumed that CBPs had positive effects on performance and, therefore, the analysis was aimed at where improvements in crew reliability can be expected. For example, Moieni and Spurgin (1993) have noted that:

> ...computers can make up for some of the human limitations, such as short term memory and limited working memory capacity, and together with the human operator can be more effective and reliable than either acting separately. Thus, computers can help the user find his way through the procedures and help ensure that steps in the procedures are taken in the correct sequence. More importantly, they can support the crew in taking into account the correct set of symptoms, and help ensure that key elements are not ignored. In some systems, the computer can take control if the crew fails to follow the procedures as prescribed.

The cognitive model had two separate phases: the detection-diagnosis-decision making (DDD) phase and the implementation phase. There are three pathways to failure to provide the correct response within the required time. First, the crew may fail to detect the need to take action or the crew may make a misdiagnosis (P1). Second is the failure to complete the DDD phase within the required time window (P2). The third failure path is the crew's failure to complete all required actions (P3). The total probability of human error for a given human interaction is:

$$PHum,Tot = P1 + P2 + P3$$

Orvis and Spurgin felt the CBP should reduce the probability of all types of failure pathways. Since the CBP automatically detects parameters and matches them to EOP conditions, P1 could be essentially eliminated. When CBPs monitor whether a personnel action has occurred and notify the crew that it is still needed, P3 can be essentially eliminated as well. Thus,

$$PHum,Tot\ (CBP) < PHum,Tot\ (PBP)$$

*Orvis, D. and Spurgin, A. (1996). Research in computerized emergency procedures systems to enhance reliability of NPP operating crews (APG Report No. 35). San Diego, CA: Accident Prevention Group.

The features of CBPs that Orvis and Spurgin (1996) have determined will affect crew reliability are:

- display quality
- number of concurrent open windows
- coupling with plant parameters
- coupling with alarms
- display of control status
- display of plant mimics with component status
- automatic EOP selection
- easy navigation
- displays and operation similar to those for normal and abnormal operations
- automatic EOP place keeping
- limited amount of user configuration
- no lockup on erroneous use

The analysis by Orvis and Spurgin is meant to provide an assessment of the potential benefits of CBPs. Potentially negative factors were not examined, such as those examined by Stubler, Higgins, and O'Hara (1996). They are careful to point out that actual reliability improvements have to be made using tests performed at a simulator.

BNL Technical Report A-3702-10-89
Valve Sensitivity Study, October, 1989

This study analyzes the effect on plant risk, as measured by core melt frequency (CMF), of varying motor-operated valve (MOV) and check valve (CV) failure probabilities in selected PRAs. The study also compared the risk sensitivity of valves with that of human errors.

One table in the report shows the percent of base case risk (risk with all failure probabilities at their nominal values) affected by MOVs, CVs, and human errors (HEs):

Plant	MOVs	CVs	HEs
Oconee (PWR)	5.8%	<1%	97%
LaSalle (BWR)	1%	<1%	71%

As the failure probabilities are increased, the CMF will increase. This was plotted in the report to show the relative sensitivity of each item. The overall CMF was noted to be sensitive to increases in MOV and HE failure probability but not to CVs. For Oconee and LaSalle, the CMF was notably more sensitive to HEs than to MOVs for the same increase in failure probability. For Surry, the sensitivity of CMF to MOVs and HEPs was about the same. The table below shows the increase in CMF when either all MOV failure probabilities or all human error probabilities are increased by 23 times.

Plant	CMF Increase Factor for an increase in f.p. of 23x	
	MOVs	HEPs
Oconee	12	381
LaSalle	1.2	10
Surry	17	17

This report generally shows that the risk in selected PRAs is more dependent on human errors than valve failures. The methods could be used for determination of importance of human actions, either individually, in groups, or in total.

BNL Technical Report A-3967-89-2*
An Assessment of Potential Upgrades to Control Room Annunciators

Crouch et al., 1989, evaluated the importance of specific alarm system design characteristics based on expert judgement. The analysis of annunciator impacts on performance and risk was the primary objective. This study utilized a group of upgrades and evaluated their impact on plant safety through the application of probabilistic risk assessment (PRA) methodology. The Sequoyah PRA was utilized in the analysis. The human errors modeled in the PRA accident sequences were associated with annunciators available in the Sequoyah control room.

Four expert judges then evaluated the individual upgrades for their effects on the human errors in the PRA. The Success Likelihood Index Method/Multi-Attribute Utility Decomposition (SLIM/MAUD) (Embrey et al., 1984) approach was utilized to obtain a relative ranking of each upgrade. Four evaluation dimensions were utilized; each was rated on a nine-point scale with "1" representing the best case and "9" representing the worst.

- Does the upgrade minimize sensory stimuli and maximize information?
- Does the upgrade provide unique information to the operator?
- Is the upgrade useful in an accident scenario?
- What is the amount of training necessary for an operator to properly utilize an upgrade?

In addition to rating the upgrades, the judges weighted the relative importance of each evaluation dimension. A Success Likelihood Index (SLI) was then computed for each combination score across all rating dimensions and transformed to a unit scale (0-1). A higher SLI rating indicates greater importance. The results of analysis are presented in Table 1.

The SLI values were converted to human error probability (HEP) change factors, i.e., the degree to which HEPs modeled in the PRA could change when the upgrade was included. The change factors are also provided in Table 1. Alarm characteristics such as prioritization, inhibit first-out panel and reflash were all rated highly.

The combination of upgrades in a single alarm system was assumed not to be additive. Thus, it was assumed that no upgrade to the alarm system could influence HEPs by more than a factor of five. This upper limit may have been conservative in light of (1) the degree to which the alarm system can modify HEPs within the context of the annunciator response model in Technique for Human Error Rate Prediction (THERP) (Swain and Guttmann, 1983) which spans several orders of magnitude, and (2) the results of more recent human reliability analyses incorporating alarm analyses. Moore, Keil, and Fisher (1988), for example, increased HEPs by a factor of 10 when an event did not result in directly useful alarms.

Despite the possible conservatism of limiting the effects of upgrade packages to five, alarm system upgrades were found to have a significant effect on plant risk. The upgrades were grouped into three alternative packages and compared to a baseline condition of the alarm system without any upgrades. Following the PRA calculations, each package was found to reduce CMF. The reduction in CMF varied from a high of 1.33E-4 core melt events per reactor year to a low of 2.27E-5. The three alternative

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*Crouch, D., Higgins, J., Luckas, W., MacDougall, E., and Ruger, C., "An Assessment of Potential Upgrades to Control Room Annunciators," BNL A-3967-89-2, Brookhaven National Laboratory, Upton, New York, 1989.

upgrade packages were also evaluated in a value-impact (V-I) analysis and found to have favorable V-I ratios when compared to the standard NRC criteria of $1000/person-rem.

Table 1. Success Likelihood Index (SLI) Values and
HEP Change Factors for AWS Upgrades

POTENTIAL UPGRADES	SLI VALUE	HEP CHANGE FACTOR
Prioritization	0.84	1.8
Separate Silence and Acknowledge	0.84	1.8
Inhibit	0.72	1.7
First-Out Panel	0.66	1.7
Reflash	0.52	1.5
Elimination of Grouping	0.48	1.5
Tile Legibility/Intelligibility	0.48	1.5
Keying Procedures to Tiles	0.47	1.5
Blackboard for Normal Operations	0.45	1.5
Relocation of Tiles	0.45	1.5
Flashrate	0.28	1.3
Fail-Safe	0.25	1.3
Auditory Signal Intensity	0.20	1.2
Ringback	0.18	1.2

Note: Higher SLI values are associated with greater importance.

BNL Technical Report J6012-T2-6/96*
Evaluation of the Potential Safety Significance of hybrid Human-System Interface Topics

The purpose of this study was to determine whether human performance topics associated with hybrid HSIs that were identified in an earlier phase of the research raised potential regulatory concerns. The topics were evaluated using a safety significance analysis methodology to identify those topics that represent potentially safety significant issues; i.e., which have the potential to compromise plant safety.

The safety evaluation methodology consisted of the following steps: (1) Preliminary Screening, (2) Safety Significance Analysis, (3) Initial Prioritization, (4) Peer Review, and (5) Final Classification and Prioritization. In the first step, topics were screened out if they were already being addressed by the NRC in other projects. The second step was the safety significance analysis. The analysis was based on an adaptation of the approach developed by EPRI in Guideline on Licensing Digital Upgrades (EPRI, 1993), which was endorsed by the NRC in Generic Letter 95-02 (NRC, 1995). The general rationale and implementation of the method is discussed below.

Commercial nuclear power plant (NPP) licensees are permitted to make plant modifications without prior NRC review if the provisions of 10 CFR 50.59 (U.S. Code of Federal Regulations) for the determination of an unreviewed safety question (USQ) are satisfied. These provisions state that the licensee can (a) make changes in the facility as described in the Safety Analysis Report (SAR), (b) make changes in the procedures as described in the SAR, and (c) conduct tests or experiments not described in the SAR without NRC review and approval prior to implementation, provided that the proposed change, test, or experiment does not involve a change in the Technical Specifications or involve a USQ. A proposed modification is considered to involve a USQ under the following conditions [see 10 CFR 50.59(a)(2)]: (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; and (3) if the margin of safety as defined in the basis for any technical specification is reduced. The determination of whether or not a USQ may exist is made by the licensee based on a safety evaluation of the proposed change. The purpose of a 10 CFR 50.59 safety evaluation is not to determine whether or not a proposed change is safe. Further, a determination that a proposed change involves a USQ does not necessarily mean that the change is unsafe. It means that further NRC review is necessary prior to implementation of the change.

The EPRI guidance focuses on digital upgrade issues and was developed to assist licensees in implementing and licensing digital upgrades using the 10 CFR 50.59 evaluation criteria. The evaluation process may be performed qualitatively. The guidance begins with seven primary questions and a set of supplemental questions to help focus the analysis on important considerations. An answer of "yes" to any of the seven primary questions indicates that a USQ exists:

1. May the proposed activity increase the probability of occurrence of an accident evaluated previously in the Safety Analysis Report (SAR)?

2. May the proposed activity increase the consequences of an accident evaluated previously in the SAR?

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*Stubler, W., Higgins, J., and O'Hara, J. (1996). *Evaluation of the potential safety significance of hybrid human-system interface topics* (BNL Technical Report J6012-T2-6/96). Upton, New York: Brookhaven National Laboratory.

3. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR?

4. May the proposed activity increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR?

5. May the proposed activity create the possibility of an accident of a different type than any evaluated previously in the SAR?

6. May the proposed activity create the possibility of a malfunction of equipment important to safety when the malfunction is of a different type than any evaluated previously in the SAR?

7. Does the proposed activity reduce the margin of safety as defined in the basis for any technical specification?

The hybrid HSI human performance topics were considered to be within the context of potential plant modifications that need to be reviewed with respect to their potential as USQs. Thus, the EPRI guidance was used as a general model for the development of a safety significance analysis methodology for evaluating the hybrid HSI topics. However, it is important to consider two essential differences between the characterizations of the HSI topics and descriptions of actual plant modifications. First, the information describing the topics is less detailed than a description of an actual plant modification. Hybrid HSI topics are generic in the sense that they are relevant to broad classes of upgrades and the full range of operating NPPs. The description for an actual plant modification would contain detailed information regarding characteristics of the specific upgrade. Second, plant-specific information, such as SAR analyses, plant descriptions, and upgrade implementation plans would be available for an actual upgrade, but is not available for generic characterizations of hybrid HSIs.

With these differences in mind, the analysis process was modified somewhat to better reflect its use as a research tool. Each hybrid topic was described in terms related to a potential modification that could be made to existing NPPs. The example modifications were then evaluated using the EPRI guidance. The wording of the questions was modified slightly. The phrase "proposed activity" was replaced with the phrase "proposed modification." Because the evaluation was based on a characterization of an upgrade, rather than an actual upgrade, evaluations of "likely" or "not likely" were applied to the primary questions rather than the more definitive responses of "yes" or "no," which would be used in the evaluation of an actual plant modification. Associated with the seven primary questions were supplemental questions, which addressed specific characteristics of digital systems. A subset of the supplemental questions that pertained to personnel performance was considered in the evaluation. These supplemental considerations generally addressed (1) failure modes that are caused or aggravated by personnel actions, and (2) failure modes and equipment characteristics that have negative effects on personnel performance.

An additional modification related to the findings is in the evaluation of actual plant modifications, a "yes" response to any of the primary questions results in the identification of a USQ. In our analysis methodology, a "likely" response to any of the primary questions resulted in an identification of the topic as "potentially safety significant." Similar to a USQ, the identification of a topic as a potentially safety significant issue does not mean that the types of plant modifications represented by the topic are necessarily unsafe. It means that its human performance concerns have the potential to compromise plant safety. Therefore, should a review be necessary of plant modifications involving safety significant topics, guidance will be needed by NRC staff with the technical basis to help ensure that the modifications do not compromise safety.

A characterization of each topic was developed to serve as a basis for topic evaluations. Some topics were characterized as typical near-term modifications. These were modifications that could plausibly be performed today at an existing plant, such as the installation of a computer-based procedure system. Other topics were concerned with the process by which designs are developed and implemented. These were described in process-related terms. Each characterization also included a description of the human performance concerns that may be associated with the topic. For purposes of the evaluation, the topic of Upgrade Implementation and Transition and the topic of Acceptance and Training were combined and called Upgrade Implementation. This was done because of the overlap in considerations of these two topics.

Using these characterizations, each topic was evaluated using the seven primary questions and the subset of the supplemental considerations from the EPRI guidance. Then, an overall assessment of whether a topic was "potentially safety significant" was made based on the seven primary evaluation questions. The evaluations were performed by four BNL personnel with expertise in the areas of human factors, HSI design, NPP operations, probabilistic risk assessment, and SAR analysis.

In the next step, the topics that were identified as potentially safety significant issues were prioritized to support the effective use of research resources in the development of HFE guidance. The prioritization was based on a subjective analysis of (1) the degree to which the topic addressed the performance of personnel directly involved in the operation of the plant versus personnel involved in supporting roles, and (2) the degree to which the topic addressed HSI components that are primary sources of information and control capabilities for operators.

The fourth step was to obtain an independent review of the topic evaluation and prioritization.

BNL Technical Report L-1170 1/91
An Analysis of Human Performance Trends in the Precursor Database, January 1991

In 1990, the U.S. NRC Commissioners were briefed regarding NPP precursor data. The staff noted that the overall significance of precursor events decreased from 1985 to 1988. The Commissioners questioned whether this trend could be tied to improved training over the same time frame.

This study analyzed precursor, LER, human error and pertinent training information. During the time period, there were main training initiatives across the industry; namely, an increase in the number of plant-referenced (plant-specific) simulators, and the introduction of INPO training program accreditation. Events from the Accident Sequence Precursor (ASP) and related LERs were reviewed from 1984 through 1989. This consisted of 184 LERs, 93 of which contained a total 163 human errors. An expert panel determined that 89 of these errors could have been addressed by improved training.

The study used counts of human errors and related issues, as well as trend plots. It also used statistical correlation techniques to draw conclusions about the data.

The analyses were somewhat inconclusive in linking the training efforts to improved human performance. However, the study did note that overall risk as measured by the precursor program was improving. The study also noted that, despite the improving risk and the training initiatives, human errors were still occurring and that room for further improvement still existed.

This study found that the precursor program data was not amenable to drawing many robust conclusions about human performance, but that it did highlight human errors that had occurred in actual risk significant scenarios in the plants.

BNL Technical Report L-1170 (1) 10/90
The Quantification of Human Variability and Its Effect on Nuclear Power Plant Risk, October, 1990

This study examined the range of human performance (termed human variability) as it applied to commercial NPP risk. Human performance literature was reviewed but was found to be quite sparse. Data that was available was summarized and indicates that performance typically varies upward and downward from the mean by a factor of 2 or 3 for simple "action execution" tasks. A factor of 10 for more complex "action selection" tasks (e.g., diagnosis) was determined to be reasonable in this study. The effects of both human variability and the uncertainty in determining actual human error rates were addressed in this study.

The study also illustrated the effects of human variability and uncertainty on nuclear plant risk using sensitivity evaluations for two plant PRAs. These studies illustrated the risk benefits of limiting both the mean human error probabilities and human error variability. This presentation was an extension of that presented in NUREG/CR-5319 and -5527. Sensitivity curves showed that even if the quantitative effects of uncertainty and variability are separated and analyzed individually, their individual effects on risk are still notable. An interesting point is that the error factors (and resultant ranges of risk) are larger for uncertainty than for variability.

This study again showed the overall importance of human actions to NPP risk and added information on uncertainty and variability. It is also useful to be knowledgeable of ranges of human performance (variability) when making decisions about activities that address human performance or rely upon correct human performance. The methods are useful for prioritization activities.

BNL Technical Report L-1170 (2) 10/90
A Comparison of the Sensitivity of Risk to Human Errors in the Oconee and LaSalle PRAs, October, 1990

A comparative analysis was performed to assess the reasons for the large differences in the sensitivity of risk (core melt frequency) to human error variation that were observed in two earlier studies (NUREG/CRs -5319 and -5527) of the Oconee and the LaSalle NPP PRAs. This study provides both quantitative and qualitative analyses that illustrate and explain the various differences between the plant PRAs. The study provides both structural and underlying reasons for the large sensitivity differences.

A simple method of measuring the sensitivity of risk to human error probability (HEP) variation is to use the Risk Increase Ratio, defined as follows:

$$\text{Risk Increase Ratio (RIR)} = \frac{\text{Risk (HEP at Upper Bound Value)}}{\text{Risk (HEP at Lower Bound Value)}}$$

The RIR for Oconee is 394, while for LaSalle the RIR is 10.

The detailed comparative analysis determined that the reasons for the large difference in sensitivity could be categorized under two general headings: PRA structural reasons and underlying reasons. The most important structural reasons were:

1. the presence of multiple HEs (three or four) in cutsets of the dominant accident sequences of Oconee,

2. an overall larger number of HEs in the Oconee PRA database,

3. higher base case HEPs in the LaSalle PRA, and

4. notably different HEs, even for similar accident sequences.

The underlying reasons causing these structural differences in the two PRAs were explored. The two main underlying reasons were plant design differences and PRA/HRA modeling differences. In the plant design area, general PWR versus BWR plant differences were important as well as several Oconee-specific design features. The safety systems are significantly different between Oconee (a PWR) and LaSalle (a BWR) and this plays an important role in specifying the level of operator intervention and the consequences of human errors. Some of these plant differences include the diversity of and the automatic features of the LaSalle safety systems, which reduce the level of vulnerability to human errors during abnormal plant conditions. Some important Oconee-specific design features were the Standby Shutdown Facility which requires manual actions, and the Emergency Feedwater System which requires manual transfer during accident conditions. Examples of important PRA/HRA modeling differences are the decisions that resulted in very few pre-accident errors and no during-accident commission errors in LaSalle. While this may be considered a modeling issue, the PRA team noted that LaSalle had excellent procedures and used multiple independent checks by the plant staff to preclude pre-accident errors. Thus, there were some real plant reasons that drove these modeling decisions.

These results showed that plant-to-plant comparisons of baseline risk and risk sensitivity are complex and must be done with caution and with knowledge of the modeling decisions that were incorporated into the PRA studies. It also showed that one can examine in detail the PRA/HRA and obtain useful insights about the design of the plant and ways to limit risk.

BNL Technical Report W6152-T9-0*
NPP Shift Staffing Levels: A Risk-Based Assessment of Selected Accident Scenarios

The objective of the study was to assess the adequacy of minimum shift staffing levels in NPPs. Earlier research concluded that the necessary staffing for emergencies may not always be considered. This effort used a risk perspective to reexamine those results and to assess the potential implications of postponement or non-performance of tasks due to heave workload and/or reduced staffing.

Scenarios that were significant contributors to NPP core damage frequency (CDF) were examined in detail. A base case of normal backshift levels (defined in the earlier research) was assessed and compared with the more restrictive (less staff) technical specification requirements. Delayed tasks were identified and a quantitative assessment of the change in CDF was made. The types of scenarios examined included for example non-isolable LOCA:

- BWR fire with reactor scram followed by initiation of RCIC on RPV low-level, with a RCIC supply steamline rupture and a failure to isolate the line.

- PWR fire with reactor trip followed by SGTR with a stuck open steam generator atmospheric dump value.

These represent scenarios that are based on design basis accidents but include additional and reasonable beyond-design-basis burdens of a fire and a radioactive release. Also examined were BWR station blackout (SBO), PWR SBO, and PWR control room fire and evacuation. Several PRAs were used.

The methodology was to adjust timelines (developed from detailed task analyses for each scenario in earlier research, the "base case" assuming normal staffing levels) for the alternative staffing configurations (referred to as a "sensitivity case" that was represented by the staffing levels in the technical specifications, which was typically lower than the normal shift). Expert opinion was used to adjust the timelines. In such an analysis, the increased time required can affect the CDF.

The results indicated that shift staffing levels can have a significant safety impact (primarily due to overload resulting from fewer available personnel). Additional insights were:

- The base case sequence contribution to total CDF is less important than the reasons for that contribution.

- Scenarios with early core damage are significantly affected by staffing assumptions.

- Staffing can be influenced by plant design.

- Dual-unit plants have added concerns.

*Travis, R. and Shurberg, D. (1995). NPP Shift Staffing Levels: A Risk-Based Assessment of Selected Accident Scenarios BNL Technical Report (W6152-T9-0). Upton, New York: Brookhaven National Laboratory.

The analysis described in this study used basically the same safety significance methodology as described in Stubler, Higgins, and O'Hara (1996), but the overall approach included some additional considerations discussed below.

In a previous phase of this research, human performance issues associated with interface management were identified. It was suggested that the interface management demands may be excessive under some circumstances and that the additional workload may interfere with the operator's primary tasks and detract from their ability to handle process disturbances. Thus, interface management tasks have the potential to affect plant safety. The objective of this study was to determine whether these issues are relevant to complex process control domains and to assess them with respect to their:

- near-term relevance,
- safety significance, and
- sufficiency of existing review guidance for addressing these issues.

Each is discussed briefly below.

Near-Term Relevance to NPP HSI Design

Near-term relevance means that the issue is one that could conceivably be encountered in an NRC design review today. Near-term relevance is established for an issue to the extent that any of the following conditions are met. The HSI design characteristics and functions associated with the human performance issue:

- Exist in current nuclear plants.

- Exist in HSI upgrades available to the nuclear industry or exist in closely allied industries, such as process control facilities.

- Are incorporated into the design of the next generation of plants (and thus the NRC staff may have to consider them in a design certification application).

If at least one of these conditions is not met, then the HSI design is not considered near-term with respect to NRC regulatory activities.

Potential Safety Significance

Again, basically the same methodology as Stubler, Higgins, and O'Hara (1996) was used. One modification was that the descriptions of the proposed plant modifications were developed based on the HSI characterizations for the plant sites visited (interviews and walk-through exercises were conducted at several facilities to examine interface management technologies, tasks, and effects on primary task performance). The modification was based on features extracted from all three sites visited; thus, the analysis was not based on a characterization for one specific plant. The baseline configuration was a

[*]Stubler, W. and O'Hara, J. (1997). Evaluation of Human-System Interface Management Issues (BNL Technical Report W6546-2-5-9/97). Upton, New York: Brookhaven National Laboratory.

typical, conventional control room design and the plant HSI modification was a computerized display and procedure system with soft control capability.

Sufficiency of Current NRC Guidance

To determine whether current NRC guidance is available to address an issue, a comparison was made of the HSI design features associated with each issue and the guidance available in NUREG-0700, Revision 1. The NUREG-0700 guidance is insufficient to the extent that the HSI features and functions associated with the issue are not addressed.

Based on these analyses, the interface management issues were found relevant to complex process control facilities. In addition, the criteria of near-term relevance, safety significance, and insufficiency of existing review guidance were satisfied.

Risk-Based Inspection Guides (RIGs), various documents, e.g., BNL Technical Reports A-3875-T2B, Rev. 1, December 1990, and A-3453-87-5, September 1987

A series of risk-based inspection guidance documents were developed by the national labs for the NRC in the late 1980s and early 1990s. These are two example documents. The format and contents of these documents evolved over the time period in which they were developed. The intent of the program was to extract the risk important items from the plant PRAs and present it in a format that would be useful to NRC inspectors in prioritizing their inspection activities. Included among the items presented in these reports were the important or dominant human errors. These were sometimes listed in a separate table and sometimes included in the tables for each plant system. The human errors were included in the RIG tables based on determinations made in the licensees' PRAs that those errors were risk significant (typically based on importance calculations).

As an example, report A-3875-T2B, Rev. 1, was for the Trojan Nuclear Plant and contained five important human errors: failure to recover offsite power, failure to switch from the RWST to the containment sump, failure to manually start the locked out AFW pump, given an ATWS - failure to manually scram or failure to initiate and successfully perform emergency boration, and failure to successfully isolate an interfacing LOCA condition.

In the report A-3453-87-5 for the Grand Gulf Nuclear Station, there were nine important human errors listed. These included: failure to manually initiate the diesel generators, failure to manually initiate HPCS, failure to manually initiate RCIC, failure to open the suppression pool suction valve for RCIC switchover, and failure to restore valves in the SLC test line thus diverting SLC flow from the reactor.

These documents and the lists of important human errors are plant specific and provide insight on the types of human actions that are risk significant. This information is useful for each plant but also generically in determining NRC activities.

Lorenzo and Burne, 1985*

Lorenzo and Burne (1985) evaluated alarm system design within the framework of expert judgement and PRA. The study compared digital control rooms to conventional control rooms with respect to human reliability analysis (HRA) to determine the impact of computer-based interfaces on human error probabilities (HEPs). A conventional control room was compared to an advanced computer-based control room using the HRA methodology developed by Swain and Guttman (1983). It was concluded that the computer-based interfaces reduced HEPs by factors ranging from 6 to 50, depending on the type of interface being considered. For example, the presentation of a large number of parameters in a compact form was associated with HEP reductions of as much as a factor of 50. However, based upon the analysis, the authors recommended that the planned use of computer-based alarm message presentation be changed to a conventional tile approach. VDU displays of alarm information would then be used only as a backup to provide alarm sequence records. However, no rationale was provided to support the recommendation. (It should be noted that while this analysis suggests that computer-based interfaces may significantly improve human reliability, the study must be considered preliminary since it was limited by the fact that little, if any, data exists on HEPs associated with advanced interface technology. Thus, the uncertainties associated with HRA methodologies are made greater by the lack of data and industry experience with advanced interfaces.)

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*Lorenzo, D., and Burne, R., "Computer-Based Controls Can Reduce Operator Errors," paper presented at the 7th International System Safety Conference, San Jose, California, 1985.

Plot, Laughery, and Haagensen (1996)*
NPP Shift Staffing Levels: Development of Task Network Models to Study Staffing Levels

This effort was performed by MicroAnalysis and Design as part of the BNL Shift Staffing Study. The purpose was to determine if shift staffing effects could be assessed using task network modeling (MicroSAINT).

A task network model was developed for several of the scenarios investigated in other phases of the project (see BNL Technical Report W6152-T9-0). These models were then used to estimate reduced staffing effects on task time and the amount of delay that resulted from increased workload. The data from one model were compared with simulator data. The authors determined that the model estimates "accurately reflected the data."

*Plot, B., Laughery, R., and Haagensen, B. (1996). NPP Shift Staffing Levels: Development of Task Network Models to Study Staffing Levels. Upton, New York: Brookhaven National Laboratory.

ATTACHMENT IV

Human Performance Programs at Other Agencies

Human Performance Programs at Other Agencies

The purpose of this report is to demonstrate the breadth of the application of human factors technology throughout the government and the domains that government agencies impact. It also demonstrates the commonality of the kinds of work being done in these different applications. NRC Human Performance staff are familiar with many of these programs and interact with staff from these other agencies on specific issues, so that the NRC can benefit from, as well as share with others, the work of common interest.
Several Federal agencies have in place active human factors programs in research, development, regulation or oversight. At least two such programs are Department-wide, covering activities across all organizations within a Cabinet-level Department. These two are the U.S. Department of Transportation (DOT) and the U.S. Department of Defense (DoD). A brief overview of their human factors programs is provided below. In addition, an overview is provided of the importance of human factors in the establishment of the U.S. Chemical Safety and Hazard Investigation Board. Although space does not permit a discussion here, other Federal agencies with active human factors programs include: The National Aeronautics and Space Administration (NASA), the Occupational Safety and Health Administration (OSHA), and the U.S. Department of Energy's (DOE) Chemical Safety Program. Table 1 provides an overview of a sample of these agencies and their human factors activities. More detailed information for these other agency programs can be found at their WEB sites.

U.S. Department of Transportation

In June 1999, the Secretary of Transportation forwarded to Congress the Department's Program for *Human-Centered Systems – The Next Challenge in Transportation,* with the following words:

> This document … calls attention to the most critical component of our transportation systems, the human operator. We recognize that, as we increasingly incorporate new technologies into our transportation systems, the human contribution to transportation operations becomes ever more crucial. Today's U.S. transportation system in structurally, mechanically, and technologically sound. The greatest challenge is to ensure it is fully designed for human operators, maintainers, and users. We need to employ human factors expertise in the design, development, evaluation and use of transportation technologies and systems to ensure that we do not exceed the limits of human performance, and that we use the full capabilities of the human. We compromise the capabilities of technologies when we fail to consider human performance issues associated with their use. We need transportation systems that adapt to humans instead of humans adapting to them.

For research and development (R&D) alone, DOT's human factors budget for FY '99 was $41.3 million. The Human-Centered Transportation initiative was developed under the Secretary's Human Factors Coordinating Committee (HFCC), established in 1991. The HFCC's responsibilities include the development and implementation of a national strategic agenda for human factors research in transportation, and facilitation of the implementation of the human factors elements of the *DOT Strategic Plan.* Of the nine modal agencies within DOT, four have active human factors programs: the Federal aviation Administration, the U.S. Coast Guard, the Maritime Administration, and the Federal Railroad Administration

Federal Aviation Administration (FAA):

The FAA has established a “National Plan for Civil Aviation Human Factors,” which includes a comprehensive Human Factors Program Management system. This program, under the auspices of the agency’s Chief Scientific and Technical Advisor for Human Factors, is carried out by the agency’s HFCC. Via Administrative Order the agency has described in detail its Human Factors Policy, the purpose of which is, in part:

> (To establish) policy and responsibility for incorporating and coordinating human factors considerations in Federal Aviation Administration (FAA) programs and activities to enhance aviation safety, efficiency, and productivity.

Key excerpts from the Order are contained in Enclosure A.

U.S. Coast Guard (USCG) and Maritime Administration (MARAD):

The USCG and MARAD have jointly developed a program titled *Prevention Through People (PTP).* The objective is to provide information to the maritime community about the role of the human element in all maritime operations, to promote the need for increased preventive measures, and to promote a human-centered approach to the design and operation of marine systems. Program elements include: improving crew alertness, human factors in casualty investigations, and performance-based testing. In addition MARAD and USCG have signed a Memorandum of Agreement to develop and implement a non-attribution national maritime safety incident reporting system to serve the interests of the U.S. public and maritime stakeholders by identifying safety problems and facilitating appropriate preventive actions.

MARAD’s 1997 Annual Report to Congress summarized the agency’s human factors activities this way:

> MARAD continues to emphasize safety and human performance in the maritime industry, focusing on the combined effects of human factors, training, management, organization, operating procedures, design, construction, and ship and shore relationships upon the safe and efficient operation of vessels. Human factors contribute to about 80 percent of all accidents. Improvements are key to achieving reliable, efficient, and competitive marine transportation that is safe for crew, passengers, and cargo while reducing the potential for pollution from accidents. This area is of equal concern in the shipbuilding, ship repair, and longshore industries.

Federal Railroad Administration (FRA):

The FRA’s program for “Enhancing Rail Safety” recently issued a report which describes the agency’s human factors concerns as follows:

> The number and proportion of train accidents attributable to track and equipment-related causes dropped dramatically between 1978 and 1986. However, train accident rate trends subsequent to 1986 imply that continued emphasis on (the traditional) approach is not reducing the number of accidents fast enough to meet agency performance goals. The proportional rise in human factor caused accidents, since 1986, now comprise the largest single causal factor for railroad accidents. They also represent a

disproportionate number of the most serious accidents. There is no doubt that increasing safety through infrastructure investment is a more clear-cut and quantifiable safety challenge than is the challenge of effectively dealing with human factor issues. In response to the proportional rise in human factor caused accidents, railroad industry restructuring, the need to address systemic safety problems, and new legislative and regulatory requirements, FRA examined ways to improve the railroad industry's safety record. Ten Administrator's Roundtable Discussions with rail labor, rail management, industry research experts, suppliers, contractors and other stakeholders, along with internal audits, and scores of external meetings with individuals and groups in every element of the railroad industry, coupled with FRA's databases and experience, produced a compelling mandate for change.

In addition to the agency-specific programs described above and others not mentioned here, DOT has underway a number of multi-modal and inter-departmental human factors activities, primarily with DoD and NASA. Significant elements of these programs include: fatigue management, operator education, qualifications and training, human/system interfaces, cognitive workload, situational awareness, system-induced errors, diversity and aging issues, fitness for duty (including the effect of prescription drugs on performance), and advanced instructional technology (AIT). The Staff is available to provide additional information on these programs should it be requested by the Commission.

The National Highway Traffic Safety Administration (NHTSA)

NHTSA is the USDOT agency responsible for driver and vehicular safety on our nation's highways. NHTSA is an active participant in USDOT's Human Centered Transportation Program. In addition NHTSA has several major human factors initiatives underway. These include:

The National Advanced Driving Simulator (NADS), now under construction, will provide a research tool to conduct fundamental research into the operation of the complex driver-vehicle-environment system. Of the three elements of this system, the driver is unique as the single element that is non-deterministic, i.e., driver behavior defies description or prediction by means of common physical laws. The NADS will offer the capability to study driver crash avoidance behavior and carry out related accident reconstruction. The complete control of highway environment and traffic scenarios provided by the NADS will allow researchers to present hazardous situations and measure driver response. This same experimental control capability will allow conditions that replicate real accident cases, and study the response possibilities and limitations of various participants and the response limitations of involved vehicles. The NADS will provide the capability for safely evaluating advanced vehicle communication, navigation, and control technologies which are now being developed as part of the Intelligent Transportation System (ITS) program. Important questions regarding the effect of these systems on driver workload, attention, behavior, and overall safety need to be addressed during the development phase. The NADS will provide the means by which experimental data can be generated to determine beforehand whether any of these advanced systems will have an unintended or detrimental impact on driver performance or highway safety. A clear understanding of driver behavior under these circumstances will lead the way to developing effective strategies and countermeasures for improved crash avoidance and reduced injuries and fatalities.

Other current NHTSA human factors programs include a congressionally mandated effort to develop educational countermeasures to the effects of fatigue, sleep disorders, and inattention on highway safety. Drowsy driving is a serious problem that leads to thousands of automobile crashes each year. A report, sponsored by the National Center on Sleep Disorders Research (NCSDR) of the National Heart, Lung, and Blood Institute of the National Institutes of Health, and NHTSA was prepared to provide direction to an NCSDR/NHTSA educational campaign to combat drowsy driving. The report presents the results of a literature review and opinions of the Expert Panel on Driver Fatigue and Sleepiness regarding key issues involved in the problem.

NHTSA also has a large program, jointly sponsored with Federal Highway Administration (FHWA), titled: "The ITS (Intelligent Transportation Systesm) Crash Avoidance Research Program." This activity seeks to develop a broad base of understanding that can lead to introduction of advanced crash avoidance systems within the context of the advanced electronic, computer, and communication technologies being rapidly developed in vehicles and on our nations highways. The program provides an opportunity for seeking new remedies that can help drivers avoid crashes.

Federal Highway Administration (FHWA):

FHWA is the USDOT agency responsible for safe and efficient traffic movement on our nations roads. The agency's current Strategic Plan states, in part:

> FHWA will identify and promote deployment of safety technology with particular emphasis on technologies that address high priority areas, including run-off-road and pedestrian and bicycle incidents. Advancement of ITS technologies will also be a key part of the safety initiatives. FHWA's longer term safety strategy is a technology-based systematic approach to enhance the safety of the roadway, vehicles, and users. They will facilitate the research, development, and application of technology to dramatically change operations and integrate these new technologies to create a fail-safe highway environment. The initial steps toward this include the testing of the automated highway system, automated traffic management systems, pedestrian signal communications systems, intelligent vehicles and vehicle systems, and commercial vehicle and passenger car technologies that monitor the fatigue and safety performance of individual drivers. Focusing on Human Behavior - in the Headquarters and field, the FHWA will add its resources to work on educational and enforcement activities designed to change human behavior while using the roadway environment. The FHWA will join all the modal administrations in such activities that increase the use of seat belts, reduce the number of red light running crashes, and reduce the number of alcohol related crashes. It is estimated that increasing the use of seat belts and decreasing drunk driving have the highest potential for reducing fatalities on the Nation's highways.

OFFICE OF MOTOR CARRIERS AND HIGHWAY SAFETY (FHWA-OMC)

In the past, the motor carrier program was primarily built on regulations and enforcement. Today, it is a safety program which identifies and implements innovative and performance-based programs. FHWA will: promote safe driving practices in the vicinity of large trucks; build partnerships to improve motor carrier safety and performance of commercial motor vehicles and drivers; target enforcement on the highest-risk motor carriers, and identify and deploy new technologies to enhance the safety performance and productivity of the motor carrier industry.

Specific safety programs include:

Driver Training and Performance
Driver Alertness and Fatigue, Hours of Service
Driver Qualification
Accidents
Safety records
Equipment inspection and maintenance
Frequency and severity of Rule and regulatory violations
Adequacy of safety management controls
Accident countermeasures
Guidelines for determining preventability of accidents
"Commercial Vehicle Preventable Accident Manual"
Federal Motor Carrier Safety Regulations Part 385 - Safety Fitness Procedure

U.S. Department of Defense (DoD)

The Department of Defense Human Factors Engineering Technical Group (DoD HFE TAG) was implemented by a Memorandum of Understanding signed by the Assistant Secretaries of the Services in November 1976 for the purpose of coordinating and communicating research and development at the working level among the services and other Government agencies involved in Human Factors Engineering.

The major goal of the HFE TAG is to provide a mechanism for the timely exchange of technical information in the development and application of human factors engineering by enhancing the coordination among Government agencies involved in HFE technology research, development, and application. The HFE TAG also assists, as required, in the preparation and coordination of tri-service documents, and sponsors in-depth technical interaction, which aids in identifying HFE technical issues and technology gaps.

Because of the diversity of the subject matter covered by the HFE discipline, the scope of the technical areas addressed by the HFE TAG is broad. For the purposes of the HFE TAG, HFE is defined as dealing with the concepts, data, methodologies and procedures which are relevant to the development, operation and maintenance of hardware and software systems. The subject matter subsumes all technologies aimed at understanding and defining the capabilities of human operators and maintainers.

In order to accomplish the work of the TAG, several active SubTAGs are established as needed. Currently these include: Controls and Displays/Voice-Interactive Systems, HFE/Human Systems Integration, Personnel Screening and Performance Prediction, Telemedicine and Biomedical Technologies, Standardization, Test and Evaluation, Human Modeling and Simulation, Sustained/Continuous Operations, System Safety/Health Hazards/Survivability, Workload Coordinating, and User-Computer Interaction.

Periodically, the TAG reports on “Human Factors Hot Issues.” In its April 1998 report the TAG identified 45 such issues, many of which suggest direct parallels with current human factors issues faced in the commercial nuclear industry.

National Institute for Occupational Safety and Health (NIOSH)

Major human factors activities of NIOSH relate to on-the-job safety in industries as diverse as mining, farming and fire prevention. Major NIOSH human factors initiatives include studies of fatigue, shiftwork and shift rotation, interactive simulation for training and performance assessment, equipment design, environmental factors and their effects on human performance, hazard recognition, safe work practices, workplace ergonomics, and accident investigations. Specific NIOSH programs with a human factors orientation have included: MERITS - Mine Emergency Response Interactive Training Simulator; design of Life Support Equipment Safety Systems; Human Factors Design Recommendations for Underground Mobile Mining Equipment; a study of Environmental and Human Factors Issues Associated with Incidence of Farm Equipment Accidents; and a Fire Fighter Fatality Investigation and Prevention

Occupational Safety and Health Administration (OSHA)

On November 23, 1999, the Occupational Safety and Health Administration published Ergonomics Standard for public comment as a Department of Labor rule. Under the OSHA proposal, about 1.6 million employers would need to implement a basic ergonomics program -- assigning someone to be responsible for ergonomics; providing information to employees on the risk of injuries, signs and symptoms to watch for and the importance of reporting problems early; and setting up a system for employees to report signs and symptoms. OSHA estimates that an average of 300,000 workers can be spared from painful, potentially disabling, injuries, and $9 billion can be saved each year under a proposed ergonomics program standard.

The OSHA proposal identifies six elements for a full ergonomics program: management leadership and employee participation, hazard information and reporting, job hazard analysis and control, training, musculoskeletal disorders (MSD) management and program evaluation. OSHA intends that ergonomics programs be job-based, i.e., cover just the specific job where the risk of developing an MSD exists and jobs like it that expose other workers to the same hazard. Ergonomics programs need not cover all the jobs at the workplace. The proposal would require that workers who experience covered musculoskeletal disorders receive a prompt response, evaluation of their injury and follow-up by a health care professional, if necessary. Workers who need time off the job to recover from the injury could get 90 percent of pay and 100 percent of benefits. Workers on light duty would receive full pay and benefits.

U.S. Chemical Safety and Hazard Investigation Board:

The U.S. Chemical Safety and Hazard Investigation Board (CSB) was established by law in 1990, though did not begin work until 1998, as an independent federal agency to serve as a resource in the effort to enhance industrial safety that acknowledges the growing risk that chemicals present and the need to work in partnership with industry to reduce the likelihood and effects of chemical-related accidents. The CSB is modeled after the National Transportation Safety Board (NTSB).

The CSB was established to (1) conduct investigations and report on findings regarding the causes of chemical accidents both at fixed facilities and "on the road," (2) evaluate and advise Congress on the effectiveness of any duplication of effort among 14 other federal agencies (including the "enforcement" bodies, i.e., the U.S. Environmental Protection Agency [EPA] and the U.S. Department of Labor's Occupational Safety and Health Administration [OSHA]) in preventing industrial chemical accidents, (3) conduct special studies, and (4) develop and communicate recommended actions (based on research and investigative findings) to improve the safety of operations involved in the production, transportation, and industrial handling, use and disposal of chemicals.

In the language discussing the establishment of the Board, the U.S. Senate wrote (Senate Report No. 101-228, 1989):

> ...special emphasis should be put on expertise in 'human factors' and the role that operator failures play in causing accidents. In other fields, the United States has fallen behind the international community in the use of operator training and the development of operating and emergency procedures to prevent accidents and minimize their consequences (p. 229).

The CSB is not listed in Table 1 because they do not yet have a specific human factors program, though they do have human factors people among their investigation staff.

Table 1 Overview of Federal Agency Programs in Human Factors

Human Performance Issues Addressed	FAA	USCG, MARAD	FRA	NHTSA, FHWA, OMC	DoD	NIOSH, OSHA
Operator selection and training	X	X	X	X	X	X
Safety of work places	X	X	X	X		X
Fatigue, staffing, alertness, shiftwork, hours of service, workload	X	X	X	X	X	X
Equipment Design	X		X	X	X	X
Impacts of advanced technology, human-centered automation	X	X	X	X	X	X
Lighting	X		X	X		X
Hazard recognition	X	X	X	X		X
Markings	X			X	X	X
Rule and regulatory violations – frequency and severity	X	X	X	X		X
Management -safety controls	X		X	X	X	X
Safety records	X	X	X	X	X	X

Human Performance Issues Addressed	FAA	USCG, MARAD	FRA	NHTSA, FHWA, OMC	DoD	NIOSH, OSHA
Root cause accident investigations	X	X	X	X		
Team work	X	X	X		X	
Communications	X	X	X	X	X	
Accident countermeasures	X		X	X		X
Operator performance assessment	X	X	X	X	X	
Information management; information display overload	X	X	X	X	X	
Fitness-for-duty	X	X	X	X	X	X
Anonymous reporting system for incidents and “near misses”	X	X				
Industry restructuring and deregulation	X		X	X		
Environmental factors	X	X	X	X	X	X
Human performance modeling, simulation	X	X	X	X	X	X
Safe work practices	X	X	X	X	X	X

Excerpts from Federal Aviation Administration
Administrative Order 9550.8, “Human Factors Policy”

> This order (stresses) that human factors should be emphasized early in planning and execution for maximum benefit. The order reaffirms the establishment of the Human Factors Coordinating Committee (HFCC) with its prescribed membership and purpose of providing a cross-organizational forum for the exchange of human factors information.

1. PURPOSE. This order establishes policy and responsibility for incorporating and coordinating human factors considerations in Federal Aviation Administration (FAA) programs and activities to enhance aviation safety, efficiency, and productivity. This order also prescribes the composition and function of the Human Factors Coordinating Committee (HFCC).

4. BACKGROUND.

a. The human factor has been widely recognized as critical to aviation safety and effectiveness. In the report entitled "Safe Skies for Tomorrow," the Office of Technology Assessment (OTA) concluded that long-term improvements in aviation safety will come primarily from human factors solutions. Further, such solutions will be established through consistent, long-term support for human factors research and development, analysis, and the application of human factors information. Subsequent to the OTA report, Congress enacted the Aviation Safety Research Act of 1988 (PL 100-591) calling for the FAA to augment its research effort in human factors and ensure coordination with other agencies performing such research. These assessments and directions resulted in the FAA's emphasizing new and coordinated efforts in the area of human factors with National Aeronautics and Space Administration, Department of Defense, and a multitude of professional groups such as the Human Factors and Ergonomic Society and the Air Transport Association Human Factors Task Force, whose members include pilot and contractor unions, airframe and parts manufacturers, as well as major airlines. One product of such efforts includes the National Plan for Aviation Human Factors.

b. This plan, together with the agency's recognition of the importance of human factors, calls for a systems approach to human factors and the need to institutionalize the application of human factors principles within FAA. This systems approach will:

(1) Conduct human factors research on existing systems and operations to define problems and identify cost-sensitive solutions to achieve performance enhancements;

(2) Acquire the necessary human performance information in the research, development, and engineering process; and

(3) Apply that information to FAA Capital Investment Program system acquisitions and to FAA regulatory activities in the promotion of civil aviation.

c. The benefits of this approach are increased personnel efficiency and effectiveness, improved system performance, reduced operations and maintenance costs, increased availability of objective data for use in FAA regulatory activities, and enhanced aviation safety.

7. HUMAN FACTORS COORDINATING COMMITTEE. This order prescribes the HFCC as stated in Appendix 1, Human Factors Coordinating Committee. The committee will facilitate the generation and application of human factors information throughout FAA.

8. POLICY. Human factors shall be systematically integrated into the planning and execution of the functions of all FAA elements and activities associated with system acquisitions and system operations. FAA endeavors shall emphasize human factors considerations to enhance system performance and capitalize upon the relative strengths of people and machines. These considerations shall be integrated at the earliest phases of FAA projects.

9. OBJECTIVES. The human factors-oriented approach is to:

a. Conduct the planning, reviewing, prioritization, coordination, generation, and updating of valid and timely human factors information to support agency needs;

b. Develop and institutionalize formal procedures that systematically incorporate human factors considerations into agency activities; and,

c. Establish and maintain the organizational infrastructure that provides the necessary human factors expertise to agency programs.

10. RESPONSIBILITIES. To ensure that human factors are systematically included in FAA endeavors, executive directors and assistant and associate administrators shall, as appropriate within their organizational purview, establish and assign responsibilities to accomplish the policy and objectives cited in paragraphs 8 and 9.

APPENDIX 1. HUMAN FACTORS COORDINATING COMMITTEE

The Human Factors Coordinating Committee (HFCC) shall continue in force to facilitate FAA human factors endeavors constructively and enhance the use of human factors information. Composition and functions of the HFCC are as follows:

1. Composition. The committee shall be sponsored and chaired by the FAA Chief Scientific and Technical Advisor for Human Factors (Office of the Executive Director for System Development). Executive directors, associate administrators, assistant administrators, and center directors shall each designate one member who shall have the authority to represent that organization in human factors matters. The HFCC shall encourage participation in its activities by others with significant human factors responsibilities.

2. Functions. The HFCC shall function as an intra-FAA committee to coordinate human factors information, matters, and activities among the executive directors, associate administrators, and assistant administrators. Specific functions of the HFCC are:

a. Enhance the identification of human factors research requirements and the coordination of research results;

b. Foster the dissemination and exchange of human factors information among agency organizations;

c. Facilitate the integration of human factors into rulemaking, systems acquisitions, and other activities within the agency;

d. Identify the need for changes to existing policies, processes, research programs, regulations, or other human factors-related activities and programs; and

e. Monitor the efficacy of human factors efforts and programs within FAA.