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USES OF HUMAN RELIABILITY ANALYSIS PROBABILISTIC RISK ASSESSMENT RESULTS TO RESOLVE PERSONNEL PERFORMANCE ISSUES THAT COULD AFFECT SAFETY

J.N. O'Brien and C.M. Spettell

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DEPARTMENT OF NUCLEAR ENERGY, BROOKHAVEN NATIONAL LABORATORY
UPTON, LONG ISLAND, NEW YORK 11973

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Prepared by
ENGINEERING ANALYSIS AND HUMAN FACTORS GROUP
DEPARTMENT OF NUCLEAR ENERGY
BROOKHAVEN NATIONAL LABORATORY
UPTON, LONG ISLAND, NEW YORK 11973

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ABSTRACT

This report is the first in a series which documents research aimed at improving the usefulness of Probabilistic Risk Assessment (PRA) results in addressing human risk issues. This first report describes the results of an assessment of how well currently available PRA data addresses human risk issues of current concern to NRC.

Findings indicate that PRA data could be far more useful in addressing human risk issues with modification of the development process and documentation structure of PRAs. In addition, information from non-PRA sources could be integrated with PRA data to address many other issues.

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EXECUTIVE SUMMARY

This report is the first in a series which documents research aimed at improving the usefulness of Probabilistic Risk Assessment (PRA) results in addressing human risk issues. This first report describes the results of an assessment of how well currently available PRA data address human risk issues of current concern to NRC. A three-step process was used to make that assessment.

In the first step, all Human Reliability Analysis (HRA) data included in 19 PRAs of commercial nuclear power plants were identified, collected, and stored on a computer. For each HRA/PRA datum identified, all descriptive information presented in the PRA concerning that datum was entered into the data record which includes information on the (1) personnel, (2) actions, (3) performance-shaping factors (PSFs), (4) situations, and (5) systems involved.

It was found that of the 1976 HRA/PRA data records collected and stored in this effort, 78% address operator actions, roughly 10% address maintenance personnel actions, 10% I&C personnel actions, and the remainder were shift supervisor and STA-related. Specific actions were identified for all of the 1976 HRA/PRA data records. It was found that 48% address operating, 14% address testing, 12% address maintaining, 9% address calibrating and the remainder address other actions. Only 59% of the 1976 HRA/PRA data records (1162 of 1976) explicitly considered PSFs. Of these 1162 HRA/PRA data records, approximately 32%, were related to use of procedures, 26% related to stress, 14% to time available, and the remainder to other PSFs. Of the 1976 HRA/PRA data records collected, 45% could be classified as errors of omission while 23% could be classified as errors of commission. Very little evidence of consistent consideration of specific systems or accident sequences was observed. Instead, different PRAs modeled very different systems in detail. Similarly, aside from LOCA-type accidents, no consistent set of sequences appeared to be analyzed.

In the second step, a list of human risk "working level issues" of concern to NRC was compiled. This effort was done by reviewing Generic Safety Issues and NRC planning documents and by conducting structured interviews with cognizant NRC staff members. A list of 175 working level issues was produced which represented the data needed by NRC to address Generic Safety Issues. Working level issues were grouped into nine categories: (1) staffing and qualifications, (2) training, (3) licensing examinations, (4) procedures, (5) man-machine interface, (6) human reliability, (7) management and organization, (8) trustworthiness and (9) maintenance. In each category, each working level issue was posed as a single-sentence question concerning a particular, specific need for data or information. Each issue was then analyzed using a method which identified data needed to address it in terms of relevant (1) personnel, (2) actions, (3) PSFs, (4) situations, and (5) systems. Using all reasonable combinations of these five information categories for all the issues, a total of 30,292 individual sets of unique data needs were identified.

Finally, the HRA/PRA data which were collected from 19 PRAs in the first step were compared with the data needs identified in the second step to assess the extent to which currently available PRA data are useful in addressing human risk issues of concern to NRC. It was found in this step that of the 30,292 unique data needs identified that 3.5% (1071 of 30,292) were operator-related and that of those, 17% could be addressed by current HRA/PRA data. Further, 3.1% were related to auxiliary operators and of those, 4% were addressed by current HRA/PRA data. A paucity of HRA/PRA data related to other personnel position data needs was observed. Overall, the currently available HRA/PRA data address less than 1% of all issue data needs arising from the list of working level issues.

Overall findings indicate that PRA documents and results could be substantially more useful in addressing human risk issues if modification of the developmental process and documentation structure of PRAs was undertaken. In addition, it was found that information from non-PRA sources could be integrated with PRA data to address a broader range of issues than is currently possible.

Recommendations made in the report include:

- The HRA segment of the PRA process should be improved and expanded so that it considers all quantitative and qualitative data and information related to risk quantification and risk reduction at the plant level.

- The HRA segment of a PRA should be documented in such a way that it can be used as a technical basis for addressing a broader range of human risk issues of immediate and long-term concern to NRC.

- HRA information and data from PRAs should be systematically used, along with information and data from non-PRA sources, to address human risk issues of immediate and long-term concern to NRC.

1.0 INTRODUCTION AND SUMMAR

1.1 Purpose of this Report

The purpose of the research described in this report is to make an initial assessment of the degree to which reliability data from risk assessments of nuclear power plants are useful in addressing human risk reduction issues of concern to the Nuclear Regulatory Commission (NRC). It presents a tabulation of human reliability analysis (HRA) data currently available from published Probabilistic Risk Assessments (PRAs) of nuclear power plants and an initial comparison of these data with a representative list of NRC data needs in the area of human risk quantification and reduction. This is the first in a series of reports on enhanced methods and procedures for more adequately assessing the impact of human performance on overall risk from plant operation and for systematically using HRA/PRA results to resolve human risk issues of regulatory significance to the NRC.

The NRC has determined the PRA methodology to be a primary tool for use in analyzing the safety of plant systems; the results of these analyses can be used with the same methodology to make licensing and enforcement decisions.[1] The purpose of the research described in this report is to support NRC's PRA-related efforts by undertaking a systematic examination focusing on how risk assessments can best be conducted and documented to address human risk issues. To date, the human risk component in safety system reliability has been analyzed in only a peripheral manner in PRAs, even though 40 to 50% of all system failures are reported to involve human error (NUREG/CR-2497, 1982). This historic lack of attention to the human risk component may be attributed to an absence of qualified HRA specialists as full participants in the overall PRA process. In general, the HRA specialist has been brought into the PRA process only after critical accident sequences are identified, thereby,

[1]NRC currently has several PRA-related efforts underway aimed at improving the usefulness of PRAs. These include the Probabilistic Safety Assessment (PSA) Program, the Interim Reliability Evaluation Program (IREP), and the Risk Methodology Integration Evaluation Program (RMIEP). In addition, similar efforts are underway in the industry, including studies by EPRI and the IEEE.

significantly limiting their input. As a result, significant opportunities for fully understanding human risk factors that could enhance or degrade plant safety systems are not pursued to their fullest potential during this otherwise data-rich PRA process. As part of its overall efforts to improve the usefulness of the PRA process in this regard, NRC has established a multifaceted human reliability research program to develop and test improved methods, models, and procedures for (1) acquiring both quantitative and qualitative human performance data for risk quantification and reduction and (2) for using those data to address human risk issues of concern to the NRC.

1.2 Overview of the Research Method

The research reported here entailed (1) extraction, collection, storage on a computer, and analysis of human performance data from 19 PRAs, (2) preparation of a representative list of human risk issues currently of concern to NRC and data needed to address them, and (3) a comparative analysis of how well current HRA/PRAs accommodate these data needs. The results of the comparative analysis will be used to develop procedures for assessing the impact of human performance on overall plant risk and, thereby, addressing human risk issues that could affect safety.

1.3 Summary of Findings

Human performance data (i.e., HRA/PRA data) from 19 PRAs were identified, collected, and stored. This process yielded a total of 1976 HRA/PRA data records. An HRA/PRA data record is defined as a human performance datum used during a PRA as part of a larger system failure/accident sequence analysis. Details contained in these records included the following information on each datum when available: point estimate, uncertainty bounds, personnel involved, actions undertaken, type of error, performance-shaping factors (PSFs), system involved, and situation at the time of the error. All these data were stored in a computer data base and were subjected to analysis.

Results indicate that most human reliability data documented in PRAs are not accompanied by information on how the numbers used (i.e., point estimates of human error probabilities) were determined. For example, of the 1976 HRA/PRA data records collected, only 193 (9%) were complete (i.e., included information on the point estimate, personnel involved, actions involved, PSFs considered, the situation at the time of the error, and the plant system involved). Conversely, 91% of these data records are incomplete in this regard. Another finding was that 78% of the HRA/PRA data records collected involve reactor operator actions, 11% involved maintenance personnel error, and 10% involve instrumentation and control (I&C) personnel. Other personnel of concern in NRC human risk issues (e.g., Shift Technical Advisor, Plant Manager, Shift Supervisor) are currently not subjected to close scrutiny in the PRA process.

A total of 175 human risk issues were identified from Generic Safety Issues (NUREG-0371, -0471, -0660, -0606; NUREG/CR-0933) and further refined and clarified by a review of NRC action plans (NUREG-0985; NUREG/CR-2833, -3520) and interviews with 28 members of the NRC staff. All of the refined issues were tabulated and categorized into nine classes as follows: (1) Staffing and Qualifications (30 issues), (2) Training (21 issues), (3) Licensing Examinations (16 issues), (4) Procedures (21 issues), (5) Man-machine Interface (23 issues), (6) Human Reliability (22 issues), (7) Organization and Management (13 issues), (8) Trustworthiness (13 issues), and (9) Maintenance (16 issues). Subsequently, these 175 human risk issues were analyzed to determine the data needed to address them in terms of personnel, actions, PSFs, situations, and systems. In order to make these data needs comparable to the HRA/PRA data records collected in the initial phase of this work, combinations of personnel, action, PSF, situation, and system pertinent to each issue were generated. Each of these combinations represents a unique data need and is defined as an issue data record. A total of 30,292 issue data records (i.e., unique data needs) were systematically generated from the 175 human risk issues identified from the Generic Safety Issues.

It was recognized that some human risk issues would require data and information not contained in current PRAs or readily expressed in the form of an HRA/PRA data record. Therefore, for this analysis, each was classified a type A, B, or C issue. Type A issues were defined as those for which a complete set of HRA/PRA data records could be identified that, if filled, would comprise a complete technical basis for their resolution. Of the 175 human risk issues, 26 (15%) were classified as Type A. Type B issues were defined as those for which a partial set of HRA/PRA data records could be identified, but for which supplementary human performance information from non-PRA sources would be required to provide a complete technical basis for their resolution. Of the 175 human risk issues, 101 (58%) were classified as Type B. Finally, Type C issues were defined as those requiring data expressed in forms other than data records retrievable from current PRAs for their resolution (e.g., plant operational histories, management policies and practices, physical security). Of the 175 human risk issues, 48 (27%) were classified as Type C.

Finally, available HRA/PRA data were compared with Types A and B issue requirements which could be expressed in the form of issue data records. A total of 30,292 unique issue data records were identified as being required to provide complete or partial technical bases for resolving the 127 Types A and B issues alluded to above. Comparisons between complete data records available from the 19 PRAs analyzed in this study and issue data requirements indicate that less than 1% of the data requirements were met (193 out of 30,292). When both complete and incomplete data records were included, approximately 6.5% of the data requirements were met (1976 divided by 30,292). Approximately 98% of the data records retrieved from the PRAs used in the study involved either the reactor operator, maintenance mechanic, or instrumentation and control technician. Finally, it must be cautioned that the percentages are overestimates since not all human performance data records retrieved from current PRAs were unique.

1.4 Organization of This Report

Section 2 of this report discusses, in detail, human performance data extracted from published PRAs. Section 3 categorizes and describes human risk issues identified from Generic Safety Issues, NRC action plans, and interviews with NRC staff of the Offices of Research (RES), Reactor Regulation (NRR), Inspection and Enforcement (IE), and Materials Safety and Safeguards (NMSS). Section 4 compares Section 2 HRA/PRA data with Section 3 human risk issues to assess the degree to which existing HRA/PRA data address them. Section 5 represents a summary of conclusions drawn from the analysis of the PRA data and issues, and recommends future research on HRA/PRA tools to better fulfill NRC human risk data needs.

2.0 COLLECTION AND ANALYSIS OF HUMAN RELIABILITY DATA FROM PRAs

This section discusses the method used for the collection of human reliability data from existing PRAs and presents them in tabular form.

2.1 Data Identification and Collection

Published reports (65 volumes) of PRAs from 19 nuclear power plants were obtained through the NRC, utilities, and contractors. This set of PRAs represents 63% of PRAs underway or completed and is, thus, considered representative of utility organizations, PRA sponsors, and risk analysis contractors. Table 2.1 lists the name and type of plant, the sponsor of the PRA, the level of the PRA, and the risk analysis contractor for the PRAs included in this study.

Each page of the 65 volumes obtained was screened by three technical readers for specific keywords which referred to humans, human actions, operators, and test and maintenance activities. The keywords used, which were developed during pilot screening of PRAs, were: action(s), error(s), human, human factors, human reliability, maintenance, manual, operator(s), performance-shaping factors (PSFs), personnel, procedures, recovery actions, shift, shift supervisor, and test. Each time a keyword or phrase was identified, it was examined by experienced scientists to determine if it referred to a quantitative estimate of human performance at a given nuclear power plant task. If so, the human error probability (HEP) or human error rate (HER) was included as a "record" in the HRA/PRA data base. The minimum criteria necessary for a HEP or HER to be included in the HRA/PRA data base were (1) that the error probability be stated in quantitative terms as a point estimate or in terms of both upper and lower uncertainty bounds, (2) that the individual or personnel group to which the point estimate referred was explicitly stated, and (3) that the action in which the individual or personnel group was engaged was identified. Many PRAs discussed the quality of human performance during nuclear power plant tasks. However, no systematic way to interpret such dis-

Table 2.1 PRAs Included in HRA Analysis for This Project

Name and Type of Plant	PRA Sponsor	PRA Level*	Risk Analysis Contractor
Arkansas Nuclear One Unit 1, Pressurized Water Reactor (PWR) 2-Loop	NRC (Interim-Reliability Evaluation Program II ([IREP-II])	2	Sandia National Labs (SNL), Battelle Columbus Labs (BCL), Science Applications Inc. (SAI)
Calvert Cliffs Unit 2, PWR/2-Loop	NRC (Reactor Safety Study Methodology Application Program ([RSSMAP])	2	BCL, SNL, Evaluation Associates
Calvert Cliffs Unit 2, PWR/2-Loop	NRC (IREP-II)	1	SAI
Crystal River Unit 3, PWR/2-Loop	NRC (IREP-1)	2	SAI
Indian Point Units 2 and 3, PWR	Power Authority of NY	3	Consolidated Edison of NY Power Authority of NY
Midland Power Plant Units 1 and 2, PWR	Consumers Power Company	3	Pickard, Lowe, and Garrick Inc. (PL&G)
Oconee Unit 3, PWR/2-Loop	NRC (RSSMAP)	2	SNL
Seabrook Station Unit 1, PWR	Public Service Company, Yankee Atomic Elec. Co.	3	PL&G
Sequoyah Unit 1, PWR/4-Loop	NRC (RSSMAP)	2	SNL, BCL
Surry Unit 1, PWR/3-Loop	Atomic Energy Commission (AEC)/NRC (WASH-1400)	3	N. Rasmussen, Massachusetts Institute of Technology (MIT)
Yankee Rowe, PWR	Yankee Atomic Co.	-	Energy Inc., Yankee Atomic Electric Co.
Zion Units 1 and 2, PWR/4-Loop	Commonwealth Edison	3	
Big Rock Point, Boiling Water Reactor (BWR)	Consumers Power Co.	3	Consumers Power Co.
Browns Ferry Unit 1, BWR/4	NRC (IREP-II)	2	EG&G Idaho, Energy Inc.
Grand Gulf Unit 1, BWR/6	NRC (RSSMAP)	2	SNL, BCL
Limerick Generating Station Units 1 and 2, BWR/4	Philadelphia Electric Co.	3	Philadelphia Electric Co., General Electric Co., SAI
Millstone Unit 1, BWR/3	NRC (IREP-II)	2	SAI
Peachbottom Unit 2, BWR/4	AEC/NRC	3	N. Rasmussen, MIT
Shoreham Nuclear Power Station, Unit 1, BWR	Long Island Lighting Co.	3	SAI

*PRA level refers to the extensiveness of the PRA methodology. Level 1 PRAs include an analysis of events and systems in relation to core-melt processes. Level 2 PRAs include an analysis of radionuclide release and transport as well as an analysis of core-melt processes. Level 3 PRAsinclude an analysis of core-melt processes, an analysis of radionuclide release transportation, and an analysis of environmental transport and conseqences.

cussions was found. Thus, this analysis was limited to quantitative data on human performance in PRA. This is reasonable because it is the point estimates that actually drive the risk calculation in a PRA.

Each HEP or HER was entered as a "record" into a computerized data base. Each record contains up to 17 entries. A sample record for an HEP from a PRA is presented in Figure 2.1 to illustrate the information stored in the data base. This data base was developed using the commercial software package dBASE-III (manufactured by Ashton-Tate) and an IBM/PC personal computer. A separate file for each PRA was established; thus, the data base contains a total of 19 individual files.

2.2 General Classification Scheme

Each HEP or HER was entered into a PRA file in the data base as a record. Each record contains up to 17 entries providing information about the HEP or HER, which are illustrated in Figure 2.1.

Record No.	1
PTESTIMATE	1 x 10-2
UPPERBOUND	0.0
LOWERBOUND	0.0
ESTIMATYPE	HUMAN ERROR
PERSONNEL	OPERATOR
ACTIONS	OPERATE TO SWITCHOVER AND INITIATE AFWS
SYSTEM	AFWS
ACCSEQUENC	LOCA LARGE
ERROROMCOM	OMISSION
ERRORTYPE	PROCEDURAL
ERRORTYPE2	NONE
PSF1	TIME AVAILABLE = 6-15 MINUTES
PSF2	TRAINING AND EXPERIENCE HIGH
PSF3	PROCEDURES AVAILABLE
PLANTNAME	CALVERT CLIFFS - PWR
COMMENTS	FROM NUREG/CR-1278

Figure 2.1 A sample HEP data record.

Entry 1: Record No.

Each record was automatically numbered in sequential fashion as it was entered into the data base file. Figure 2.1 is the first record entered into the PRA file.

Entry 2: PTESTIMATE

PTESTIMATE is an abbreviation for Point Estimate. The actual HEP or HER was entered at this point in the record.

Entry 3: UPPERBOUND

UPPERBOUND refers to the upper uncertainty bound of the point estimate. If this was reported in the PRA for the particular HEP or HER, it was entered in the record here. If it was not reported, 0.0 was entered into this category.

Entry 4: LOWERBOUND

LOWERBOUND refers to the lower uncertainty bound of the point estimate. If this was reported in the PRA for the particular HEP or HER, it was entered in the record here. If it was not reported, 0.0 was entered into this category.

Entry 5: ESTIMATYPE

ESTIMATYPE is an abbreviation for Estimate Type. It was used to classify the HEP or HER as either a "HUMAN ERROR" (as illustrated in Figure 2.1) or a "SYSTEM UNAVAILABILITY." An HEP or HER was classified as a human error if it referred to a probability of failure to perform a particular nuclear power plant task which contributed to a system failure and thus to risk. An HEP or HER was classified as a system unavailability if it referred to the unavailability of a necessary safety system as a result of human actions such as test and maintenance activities.

Entry 6: PERSONNEL

PERSONNEL refers to the personnel category of the individual to which the HEP or HER refers. The staff positions and organizational groups which could be entered in this category are listed in Table 3.1 of this report.

Entry 7: ACTIONS

The particular action or actions to which the HEP or HER referred was included under the entry ACTIONS in the record. Actions were classified using the verbs listed under Actions in Table 3.1. This classification was based on NUREG/CR-2744 (1983) which presented a detailed taxonomy of nuclear power plant tasks and actions for different personnel classifications. As can be seen in Figure 2.1, additional descriptions of the action were included in the ACTIONS category following the main action verb (from Table 3.1) if that information was available in the PRA.

Entry 8: SYSTEM

SYSTEM refers to the nuclear power plant system for which the HEP or HER was included in the analysis. The acronyms used in the PRA were used in the record. When no information was provided in the PRA to classify the system, NONE was entered under SYSTEM. In Figure 2.1, AFWS under SYSTEM refers to Auxiliary Feedwater System.

Entry 9: ACCSEQUENC

ACCSEQUENC refers to the accident situation for which the HEP or HER was included in the analysis. Accident situations were classified according to the information provided in the PRA for the particular HEP or HER. Accident situations were classified as loss-of-coolant accidents (LOCAs), or as another type of transient. When no information was provided in the PRA to classify the Accident Situation, NONE was entered under ACCSEQUENC.

Entry 10: ERROROMCOM

ERROROMCOM is used to classify the HEP or HER as an error of either omission or commission. When no information was provided in the PRA to make such a classification, NONE was entered under ERROROMCOM. In Figure 2.1, the operator's failure to switch over and initiate the AFWS was classified as an error of omission.

Entries 11 and 12: ERRORTYPE AND ERRORTYPE2

ERRORTYPE and ERRORTYPE2 are used to include any additional information about the HEP or HER which was available from the PRA. For example, if the HEP was described as a "common-mode error," common-mode was entered under ERRORTYPE. When additional information which could be used to classify the HEP was included in the PRA, this information was entered under ERRORTYPE2. When no information was available to further classify the HEP or HER, NONE was entered under ERRORTYPE and ERRORTYPE2. ERRORTYPE in Figure 2.1 illustrates that the HEP was considered to be a "procedural" error. That is, the operator failed to follow the correct procedures in order to switch over during a large LOCA and initiate the Auxiliary Feedwater System.

Entries 13, 14, 15: PSF1, PSF2, PSF3

PSF1, PSF2, and PSF3 refer to PSFs which were discussed in the PRAs with respect to the particular HEP or HER entered in the record. PSFs which could be entered in these categories include those from Table 3.1 of this report. In Figure 2.1, the HEP for operator failure to switch over and initiate the AFWS during a large LOCA was reportedly modified by a consideration of the time available to perform the actions (6 to 15 minutes), the training and experience level of the operator, and the fact that procedures were available to help the operator perform the action. When no PSFs were mentioned in the PRA with respect to the particular HEP or HER, NONE was entered under PSF1, PSF2, and PSF3.

Entry 16: PLANTNAME

The name of the plant is entered at this point in the data record.

Entry 17: COMMENTS

Additional information about the HEP or HER is included in the category labeled COMMENTS. In most data records, NONE was entered in this category. In Figure 2.1, the source of the HEP was included in the COMMENTS category.

Thus, the sample data record illustrated in Figure 2.1 shows that a probability of 1×10^{-2} was assigned to the likelihood that an operator would fail to switch over and initiate the auxiliary feedwater system during a large LOCA. This human error was classified as an error of omission, and as a procedural error (i.e., the operator failed to follow the procedures properly to carry out the switchover). As a result, the AFWS failed. The source of this HEP was NUREG/CR-1278 (1983) and was apparently modified by a consideration of the time available to perform the action, the training and experience level of the operator, and the availability of procedures. No upper and lower uncertainty bounds were reported for this HEP.

2.3 Results and Interpretation

A total of 1976 HRA/PRA data records involving human errors were obtained as a result of the analysis of 19 PRAs. (Data records containing information on system unavailabilities were not subjected to analysis.) Tables 2.2 through 2.7 present the results of this analysis. These tables provide quantitative information of breadth of HRA data contained in current PRAs. As indicated, the vast majority of HRA/PRA data collected were not accompanied by information that fully describes the errors under consideration.

Table 2.2 presents the number of records for each nuclear power plant personnel category analyzed in the HRA segment of the 19 PRAs. This table provides subtotals by both PRA and personnel category across all PRAs. It shows that the data records of 9 personnel categories have been considered in assessing risks of nuclear power plant operation. Of these, the vast majority (78%) refer to the actions of operators.

Table 2.3 presents the PSFs considered in the HRA segments of PRAs analyzed. This table provides subtotals by PRA and for each PSF across all PRAs. Of the 1162 PSFs considered in the quantification of HEPs across all PRAs, the most frequently considered were procedures, stress, and time available (representing 32%, 26%, and 14% of these 1162 data records, respectively). The quality of procedures most often considered to modify HEPs were whether written procedures were available to assist the person acting and whether check lists were used properly. Stress is a nonspecific PSF referring to the subjective state of the person acting. Thus, severity of accident conditions, number of annunciators or alarms sounding, perceived risk, time available, and amount of distracting stimuli were all at times labeled stress in various PRAs. Time available refers to the amount of time the individual had to perform an action before nonrecoverable system failure occurred.

Table 2.4 presents the number of records for each human action in each PRA for PWRs and BWRs. This table provides subtotals by PRA and for each action across PRAs. The most frequently considered actions for which HEPs were reported were OPERATING, TESTING, MAINTAINING, and CALIBRATING. OPERATING encompasses actions covering a range of complexity--from manipulating individual switches, valves, and pumps to performing a sequence of actions to achieve a certain system state (e.g., switchover from injection to recirculation). Data records involving the action TESTING usually refer to failure by the operator to restore a valve or switch to operational status after it was tested. Data

Table 2.2 Number of Records for Each Personnel Category by PRA

Pressurized Water Reactors

Personnel	ANO-1	Calvert Cliffs RSSMAP	Calvert Cliffs IREP	Crystal River	Indian Point	Midland	Oconee	Sea-brook	Sequoyah	WASH-1400	Yankee	Zion
Individuals												
Shift Supervisor	6	0	0	0	0	0	0	1	0	0	0	0
Shift Tech. Advisor	0	0	0	0	1	0	0	0	0	0	[illegible]	0
Rx. Operator	186	39	19	126	85	97	60	44	15	57	[illegible]	24
Auxiliary Operator	0	0	0	0	1	3	0	0	0	[illegible]	[illegible]	0
Maintenance Mech.	14	0	1	54	1	8	2	6	0	0	4	3
I&C Tech.	0	6	0	3	1	35	14	15	1	10	3	3
Engineers	0	0	0	0	0	0	0	0	0	0	0	0
Contractor Personnel	0	0	0	0	0	0	0	0	0	0	0	0
Plant Management	0	0	0	0	0	0	0	0	0	0	0	0
SUBTOTALS BY PRA	207	45	20	183	89	143	76	66	16	67	204	30

Boiling Water Reactors

Personnel	Big Rock Point	Browns Ferry	Grand Gulf	Limerick	Millstone	Shoreham	WASH-1400	Subtotals by Personnel (BWR) and PWR PRAs Combined)
Individuals								
Shift Supervisor	0	3	0	1	0	2	0	13
Shift Tech. Advisor	0	0	0	0	0	0	0	1
Rx. Operator	100	86	17	92	172	103	20	1539
Auxiliary Operator	1	1	0	0	0	1	0	8
Maintenance Mech.	0	13	8	3	90	3	1	211
I&C Tech.	0	12	5	11	56	5	11	191
Engineers	0	2	0	0	0	1	0	3
Contractor Personnel	0	0	0	3	0	3	0	6
Plant Management	0	0	0	1	0	3	0	4
SUBTOTALS BY PRA	101	117	30	111	318	121	32	1976

Table 2.3 Number of PSFs by PRA

Pressurized Water Reactors

PSFs	ANO-1	Calvert Cliffs RSSMAP	Calvert Cliffs IREP	Crystal River	Indian Point	Midland	Oconee	Sea-brook	Sequoyah	WASH-1400	Yankee	Zion
Equipment Design	6	0	0	0	2	1	0	0	0	0	20	3
Workplace Layout	0	0	0	0	2	0	0	0	0	0	0	0
Information Feedback	8	0	0	8	2	5	2	1	2	0	0	1
Man-machine Interface	0	0	0	0	0	0	0	0	0	0	0	0
Procedures	121	0	0	3	10	29	3	15	5	9	26	12
Time Available	1	6	1	6	7	18	1	33	0	5	2	13
Staffing	2	0	0	0	6	11	0	1	0	0	0	0
Job-related Training	3	1	0	0	1	4	0	0	0	0	0	0
Task Complexity	6	1	0	2	0	0	0	0	0	0	1	8
Regulations	0	0	0	0	0	0	0	0	0	0	0	0
Stress	16	6	1	7	4	19	2	24	0	0	165	0
Perceived Risk	45	0	0	0	0	0	0	0	0	0	0	0
Job-related Experience	0	0	0	0	0	4	0	0	0	0	0	0
SUBTOTALS	208	14	2	26	34	91	8	74	7	14	214	37

Boiling Water Reactors

PSFs	Big Rock Point	Browns Ferry	Grand Gulf	Limerick	Millstone	Shoreham	WASH-1400	Subtotals by PSFs (PWR and BWR PRAs Combined)
Equipment Design	14	12	0	15	0	10	0	83
Workplace Layout	6	0	0	0	12	0	0	20
Information Feedback	4	3	0	3	7	2	0	48
Man-machine Interface	4	0	0	3	0	0	0	7
Procedures	27	32	0	33	21	21	6	373
Time Available	7	14	9	19	13	0	12	167
Staffing	8	7	0	2	0	14	0	51
Job-related Training	2	0	0	1	0	5	0	17
Task Complexity	2	9	0	2	0	5	0	36
Regulations	0	0	1	0	7	0	0	8
Stress	15	11	0	20	0	5	3	298
Perceived Risk	0	1	0	0	1	0	0	47
Job-related Experience	2	1	0	0	0	0	0	7
SUBTOTALS BY PRA	91	90	10	98	61	62	21	1162

Table 2.4 Number of Records for Each Action by PRA

Pressurized Water Reactors

Actions	ANO-1	Calvert Cliffs RSSMAP	Calvert Cliffs IREP	Crystal River	Indian Point	Midland	Oconee	Sea-brook	Sequoyah	WASH-1400	Yankee	Zion
Testing	93	0	4	0	19	10	10	8	4	4	6	4
Operating	46	39	15	97	57	48	48	25	11	52	118	18
Monitoring	8	0	0	3	0	2	0	0	0	1	32	0
Inspecting	9	0	0	0	3	4	0	5	0	0	21	2
Checking	20	0	0	2	5	11	0	4	0	0	12	0
Deciding	4	0	0	6	1	7	0	3	0	0	4	0
Managing	0	0	0	0	0	0	0	0	0	0	0	0
Communicating	4	0	0	0	0	0	2	0	0	0	4	0
Calibrating	0	6	0	3	1	35	14	15	1	10	3	3
Responding	9	0	0	0	2	3	0	0	0	0	0	0
Maintaining	14	0	1	72	1	8	2	6	0	0	4	3
SUBTOTALS BY PRA	207	45	20	183	89	143	76	66	16	67	204	30

Boiling Water Reactors

Actions	Big Rock Point	Browns Ferry	Grand Gulf	Limerick	Millstone	Shoreham	WASH-1400	Subtotal by Action (PWR and BWR PRAs Combined)
Testing	10	0	0	7	87	3	0	269
Operating	61	69	15	56	80	60	20	950
Monitoring	11	3	0	7	0	7	0	74
Inspecting	8	4	2	11	0	14	0	83
Checking	8	4	0	10	5	12	0	93
Deciding	2	5	0	7	0	14	0	53
Managing	0	0	0	1	0	2	0	3
Communicating	0	4	0	2	0	3	0	19
Calibrating	0	14	5	3	56	2	11	182
Responding	1	1	0	4	0	1	0	21
Maintaining	0	13	8	3	90	3	1	229
SUBTOTALS BY PRA	101	117	30	111	318	121	32	1976

records involving the action MAINTAINING apply to Maintenance personnel only, and MAINTAINING encompasses a range of activities that maintenance personnel might perform (e.g., troubleshooting, restoration.) For the action CALIBRATING, data records apply to Instrumentation and Control personnel only. Data records which include the action word OPERATING account for 48% of the total HEPs in the HRA/PRA data base. Data records involving other actions (e.g., monitoring, inspecting, checking, deciding, and communicating) account for 16% of the total number of data records in the HRA/PRA data base.

Table 2.5 presents the frequency with which individual systems are included in HRA segments of PRAs. This table indicates no common emphasis among systems considered in HRA/PRAs. Different PRAs have concentrated on different systems so that aggregration of these data would be misleading. For example, in the HRA segments of the Midland PRA, 23% of all data records collected involve instrumentation and control systems while in the HRA segment of the Oconee PRA, only 1% involve these systems.

Table 2.6 presents the number of HRA/PRA data records for each accident situation and subtotals by accident situation for all PRAs. While it is clear that LOCAs are frequently analyzed, no consistent pattern of analyses is shown for PRAs generally.

Table 2.7 presents HEP values from the 19 PRAs that were analyzed. It contains the range of probabilities of human errors of omission and commission. A total of 883 data records were classified as involving errors ofomission. A total of 446 data records were classified as involving errors of commission. Thus, 67% of the total number of HEPs in the HRA/PRA data base could be classified as an error of omission or commission. For the remainder (33%), insufficient information was presented to classify the error as omission or commission. Of those that could be classified, errors of omission were twice as numerous as those of commission.

Table 2.5 Number of Systems by PRA

Pressurized Water Reactors

Systems	ANO-1	Calvert Cliffs (RSSMAP)	Calvert Cliffs (IREP)	Crystal River	Indian Point	Midland	Oconee	Seabrook	Sequoyah	WASH-1400	Yankee	Zion
Air	0	0	0	0	0	0	0	0	0	0	0	0
Condensate	0	0	0	36	0	0	0	0	0	0	0	0
Containment	40	0	2	0	15	6	16	1	7	5	0	2
Electrical Distribution	0	0	1	1	0	0	1	0	0	0	2	0
Emergency Core Cooling (ECCS)	124	0	2	82	20	52	41	3	8	9	16	3
Emergency Power (EPS)	0	0	1	0	20	12	2	0	0	0	0	1
Engineered Safety Features (ESFS)	0	0	2	0	0	0	0	0	0	0	0	0
Feedwater (FWS)	0	0	7	25	17	9	10	5	2	0	6	0
Fire Protection (FPS)	0	0	0	0	0	0	0	0	0	0	0	0
Instrumentation and Control	20	0	0	2	3	36	1	2	0	8	14	0
Generator	0	0	1	0	0	17	0	0	0	0	4	0
Reactor Coolant	0	0	0	33	1	14	0	0	0	0	0	0
Turbine	0	0	0	0	0	0	0	0	0	0	7	0
Water	3	0	0	0	5	6	3	0	0	4	0	0

Boiling Water Reactors

Systems	Big Rock Cliffs	Browns Ferry	Grand Gulf	Limerick	Millstone	Shoreham	WASH-1400
Air	2	0	0	0	0	0	0
Condensate	6	3	0	0	8	0	0
Containment (CS)	10	32	0	0	37	1	5
Electrical Distribution	0	1	13	0	3	0	0
Emergency Core Cooling (ECCS)	8	63	21	2	35	1	33
Emergency Power (EPS)	7	21	1	0	0	0	1
Engineered Safety Features (ESFS)	0	0	0	0	0	0	0
Feedwater	0	0	0	0	121	10	1
Fire Protection (FP)	1	0	0	0	1	0	0
Instrumentation and Control	0	1	6	1	29	3	23
Generator	0	0	0	0	0	0	0
Reactor Coolant (RCS)	3	13	2	0	0	1	2
Turbine	3	3	0	0	0	0	0
Water	8	20	3	0	56	1	4

Table 2.6 Number of Records for Each Accident Situation by PRA

Pressurized Water Reactors

Situation	ANO-1	Calvert Cliffs RSSMAP	Calvert Cliffs IREP	Crystal River	Indian Point	Midland	Oconee	Seabrook	Sequoyah	WASH-1400	Yankee	Zion
Loss of Coolant Accident (LOCA)	25	21	6	141	33	21	31	9	0	51	9	6
LOCA with Other Transient	0	0	0	0	3	0	0	0	0	0	0	0
Station Blackout	0	2	1	0	5	0	2	0	0	0	0	0
Loss of Off-site Power* (LOSP)	0	13	4	2	7	5	5	1	0	1	2	12
Degraded Power Conditions	0	0	0	0	6	0	3	0	0	0	31	0
Anticipated Transient w/o Scram	0	1	2	0	0	1	2	0	0	0	0	0
Rx. Trip	0	0	0	0	1	0	0	0	0	0	0	0
Turbine Trip	0	0	0	2	2	0	4	4	0	0	2	0
Steam Generator Tube Rupture	10	0	0	0	0	43	0	3	0	0	2	0
Loss of Feedwater	0	4	0	0	0	1	0	0	2	0	2	0
Main Steam Isol. Valve Closure	0	0	0	0	0	0	0	0	0	0	0	0
Unclassified*	1	4	0	6	17	19	0	5	0	2	2	0
SUBTOTAL BY PRA	36	45	13	151	74	90	47	22	2	54	50	18

Boiling Water Reactors

Situation	Big Rock Point	Browns Ferry	Grand Gulf	Limerick	Millstone	Shoreham	WASH-1400	Subtotals by Accident Situation (PWR & BWR PRAs Combined)
Loss of Coolant Accident (LOCA)	8	11	10	15	54	14	27	492
LOCA with Other Transient	0	0	3	0	0	0	0	6
Station Blackout	3	1	0	0	0	0	0	14
Loss of Off-site Power (LOSP)	9	0	10	0	0	0	0	71
Degraded Power Conditions	0	0	0	0	0	0	0	40
Anticipated Transient w/o Scram	2	2	0	2	0	0	0	12
Rx. Trip	0	2	0	10	5	4	0	22
Turbine Trip	2	3	0	0	0	0	0	19
Loss of Feedwater	[illegible]	0	0	0	19	0	0	28
Steam Gen. Tube Rupture	0	0	0	0	0	0	0	58
Main Steam Isol. Valve Closure	2	2	0	0	0	0	0	4
Unclassified*	18	16	7	0	6	23	0	126
SUBTOTALS BY PRA	44	37	30	27	84	41	27	892

*Based on information in PRA, these accident situations could not be classified under other categories.

Table 2.7 Range of Probabilities of Human Errors of Omission and Commission by Personnel and Action Across PRA

Omission

Personnel	Action	Range		n
Shift Supervisor	Deciding	1×10^{-4}	1×10^{-2}	3
	Communicating	1×10^{-2}		1
	Inspecting	1×10^{-2}	5.1×10^{-2}	3
	Checking	1.4×10^{-2}		1
	Operating	1.5×10^{-1}	5×10^{-1}	2
Operator	Deciding	1.09×10^{-5}	1×10^{-1}	17
	Communicating	1.6×10^{-3}	3×10^{-3}	2
	Responding	7×10^{-5}	5×10^{-3}	6
	Checking	5.08×10^{-8}	5×10^{-1}	38
	Monitoring	2×10^{-4}	6.1×10^{-3}	15
	Operating	2.2×10^{-9}	1.0	473
	Inspecting	1.37×10^{-5}	1.0	30
	Testing	4.7×10^{-3}	9.5×10^{-1}	138
Maintenance	Maintaining	1.0×10^{-7}	5×10^{-1}	144
	Communicating	3×10^{-3}	1×10^{-2}	3
I&C	Calibrating	1×10^{-3}	3×10^{-3}	2
	Inspecting	1.0	--	1
Auxiliary Operator	Operating	2.18×10^{-3}	4.4×10^{-2}	3
Shift Tech. Adv.	Checking	1.99×10^{-2}	--	1
TOTAL				883

Commission

Personnel	Action	Range		n
Shift Supervisor	Communicating	5×10^{-1}	--	1
Operator	Deciding	1.28×10^{-6}	1.26×10^{-2}	3
	Checking	3×10^{-8}	1.0×10^{-2}	18
	Monitoring	1.2×10^{-3}	2.5×10^{-1}	6
	Operating	1×10^{-5}	2.18×10^{-1}	219
	Inspecting	1×10^{-3}	2×10^{-1}	5
	Testing	2.8×10^{-6}	3×10^{-3}	11
Maintenance	Maintaining	3×10^{-6}	2.5×10^{-2}	9
I&C	Calibrating	7×10^{-12}	1.0	174
TOTAL				446

3.0 IDENTIFICATION OF HUMAN RISK ISSUES OF CONCERN TO THE NRC

The primary objective of the research presented in this report was to assess the degree to which currently available HRA/PRA data are related to human risk issues of concern to the NRC. To make such an assessment, a benchmark list of human risk issues had to be identified. It was recognized that a complete set of issues could not be developed, since NRC concerns vary over time. It was possible, however, to prepare a representative list of issues of present concern to NRC that includes a human risk component. The method used to prepare that list is presented in this section of the report. The complete list of human risk issues and related data needs developed with this method are presented in Appendices A and B.

3.1 Identification of Formal Issues With Human Performance Components

Contemporary concerns of the NRC are termed Generic Safety Issues. Four sources were reviewed in order to compile an initial list of formal Generic Safety Issues (i.e., TMI Action Plan [NUREG-0660], Task Action Plan [NUREGs-0371 and -0471], Unreasonable Safety Issues Summary [NUREG/CR-0606], New Generic Safety Issues [NUREG/CR-0933]). Items contained in each of these sources were examined to determine whether or not they contained human performance components. Generic Safety Issues from each of these sources identified as containing human performance components are listed in Appendix A (column 1).

3.2 Identification of Working Level Issues

In order to articulate the Generic Safety Issues contained in Appendix A and make them compatible with available HRA/PRA data, it was necessary to clarify, expand and refine each into the form of a question or questions asking for specific data on human performance. The compilation of questions related to the Generic Safety Issues identified above resulted in an initial list of "working level issues." The final list of working level issues derived from the Generic Safety Issues is contained in Appendix B, and was developed using a two-step process: (1) review of NRC planning documents, and (2) interviews with NRC staff familiar with those planning documents and parent Generic Safety Issues. Column 2 of Appendix A contains the working level issue designations for the issues related to each Generic Safety Issue.

Four contemporary NRC planning documents were used to refine, expand, and clarify working level issues. These documents included the Human Factors Program Plan, Revision 1 (NUREG-0985), Human Factors Society Report (NUREG/CR-2833), Safeguards Human Factors Research Plan (NUREG/CR-3520) and Maintenance and Surveillance Program Plan (Approved by the EDO January 11, 1985). This review yielded over 150 working level issues involving human performance which clarify and refine the data needed to address the Generic Safety Issues described in Section 3.1. These 150 working level issues were then arranged under nine categories: (1) Staffing and Qualifications, (2) Training, (3) Licensing Examinations, (4) Procedures, (5) Man-machine Interface, (6) Human Reliability, (7) Management and Organization, (8) Trustworthiness, and (9) Maintenance.

To further clarify, refine, and verify the working level issues organized under the nine categories above, individual interviews were conducted with 28 cognizant members of the NRC staff of the Offices of Research (RES), Reactor Regulation (NRR), Inspection and Enforcement (IE), and Materials Safety and Safeguards (NMSS). These interviews included detailed reviews and discussions of the working level issues list, or segments thereof, and yielded a final list of 175 working level issues related to contemporary Generic Safety Issues of concern to the NRC. Appendix A (Column 2) references the final list of working level issues and shows their relationship to Generic Safety Issues listed in Column 1. Appendix B lists all the final working level issues.

3.3 Identification of Working Level Issue Data Elements

Finally, in order to make direct comparisons between available HRA/PRA data and working level issues, and through those issues, the Generic Safety

Issues, it was necessary to break each working level issue down into constituent data elements. The method used to accomplish this task is referred to as the "data element and record method" because it focuses on particular aspects (i.e., elements) of each working level issue and yields a set of issue data records needed to address the issue. It reduces each working level issue to a set of personnel, action, PSF, accident situation, and plant system combinations (i.e., issue data records) capable of direct comparison to the existing HRA/PRA data records described in Section 2 of this report.

3.3.1 The Data Element and Record Method

By identifying relevant components of the questions stated in the working level issues, a complete set of data elements pertinent to all the issues was developed. These data elements fall into five categories: (1) the nuclear power plant personnel involved, (2) the actions involved, (3) the presence of factors affecting performance (e.g., stress, procedures), (4) the normal situation or transient involved, and (5) the nuclear power plant systems, structures, or components involved. The complete list of data elements derived from the working level issues for each of these five categories is presented in Table 3.1. The data elements derived from each working level issue separately are presented in Appendix C. An "issue data record" is a combination of one element from each category reflecting an individual need for specific, unique data on human performance. The greater the breadth of a particular working level issue, the larger the number of data elements and, therefore, the number of issue data records it will generate.

The form of an issue data record can be stated as a question which reflects a specific data need:

How does <u>a PSF</u> affect <u>an action</u> by <u>personnel</u> on a <u>system</u> during a <u>situation</u>?

By identifying the individual data elements in each category relevant to an issue and using all reasonable combinations of those elements in the form

Table 3.1 Categories and Elements for Developing Data Elements From Working Level Issues.

Individuals and Groups	Performance Shaping Factors	Actions	Situations	Systems
Plant Manager	Equipment Design	Testing	Loss of Coolant Accident	Safety-related Systems
Shift Supervisor	Workplace Layout	Operating	Loss of Off-site Power	Structures
Shift Tech. Advisor	Habitability	Monitoring	Other Transients	Non-safety Systems
Senior Reactor Op.	Time Available	Inspecting	System Isolation	
Reactor Operator	Staffing	Checking	Normal Operation	
Auxiliary Operator	Organizational Climate	Deciding	External Event	
Maintenance Mech.	Job-related Training	Managing	Outage	
I&C Tech.	Information Feedback	Communicating		
Chemistry Tech.	Task Complexity	Calibrating		
Health Physics Tech.	Regulations	Responding		
Engineers	Stress	Maintaining		
Security Guard	Fatigue			
QA/QC Tech.	Attitude			
Contractor Personnel	Job-related Experience			
Operations Org.	Fitness for Duty			
Maintenance Org.	Perceived Risks			
I&C Org.	Procedures			
Chemistry Org.				
Health Physics Org.				
Engineering Org.				
Plant Management				
QA/QC Org.				
Security Org.				
Off-site Response Personnel				

of issue data records, a complete set of issue data records can be generated. Appendix D contains an illustration of this method.

This method has some limitations in addressing issues which are not directly related to the type human risk of data currently developed in PRAs. To accommodate this, three classes of working level issues were identified and designated as Types A, B, and C. These are defined as:

Type A. Working level issues for which a set of quantitative issue data records can be generated that, if addressed by competent data, provide a complete technical basis for addressing the issue in question.

Type B. Working level issues for which a partial set of quantitative issue data records can be generated, but for which additional data not currently provided in PRAs may be needed to establish a complete technical basis for addressing the issue in question. These issues may require additional information such as operational history data or information on the availability of a sufficient work force.

Type C. Working level issues which require data or information not compatible to the form of data records to provide a technical basis. Instead, these issues typically require information in forms other than the data record format such as data on operational history or information on the availability of a sufficient work force.

In order to identify a complete set of issue data records that reflect all the data needed to address Types A and B working level issues, all reasonable combinations of the elements in Table 3.1 were generated. This was done by eliminating combinations of elements that were not realistic such as operators performing maintenance or I&C technicians working on plant structures. This resulted in a total of 30,292 issue data records needed to completely address all Type A working level issues and to partially address Type B.

4.0 COMPARATIVE ANALYSIS OF AVAILABLE HRA/PRA DATA AND WORKING LEVEL ISSUES

In this section available human performance data described in Section 2 are compared with working level issue data requirements described in Section 3.

4.1 Analysis of Total Data Set

Table 4.1 presents a summary analysis of human error data extracted from 19 PRAs and the degree to which they respond to the data record requirements of the 127 Types A and B working level issues described in Section 3.2. In the far left column are listed separate personnel categories for which data are needed to address Types A and B issues. [Of the 175 working level issues, 26 (15%) are Type A, 101 (58%) are Type B, and 48 (27%) are Type C.] The adjacent Column labeled 1 contains the total number of complete and incomplete data records for that personnel category retrieved from the 19 PRAs. Column 2 contains the number of Column 1 incomplete records involving action, PSF, and situation data elements only (i.e., no information on systems). Column 3 contains the number of incomplete Column 1 records involving action, PSF, and system data elements only (i.e., no information on situation). Column 4 contains the number of incomplete Column 1 records involving action, system and situation data elements only (i.e., no information on PSFs). Column 5 contains the number of incomplete Column 1 records involving action and PSF data elements only, and Column 6 records contain action data elements only. Column 7 contains records involving action and system data elements only. Column 8 contains records involving action and situation data elements only. Column 9 contains complete, <u>but not necessarily unique</u>, human performance data records extracted from all 19 PRAs for each personnel category. Column 10 contains the total number of complete and <u>unique data records</u> required as a complete technical basis for addressing Type A and a partial technical basis for addressing Type B issues for each personnel category. Finally, Column 11 contains the percentage of complete, but not necessarily unique, data records from available PRAs which address issues associated with each personnel category.

Table 4.1 Overall Data Record Comparison

	Column 1	2	3	4	5	6	7	8	9	10	11
		Number of Partial HRA/PRA Data Records									
Personnel	Total Records for Personnel Category	Action/PSF/ Situation	Action/PSF/ System	Action/System/ Situation	Action/ PSF	Action	Action/ System	Action/ Situation	No. of Compl. Records	Issue Records Required	Percent Addressed
Plant Manager	0	0	0	0	0	0	0	0	0	1274	0%
Shift Supervisor	13	0	0	0	5	2	0	0	6	2142	3%
STA	1	0	0	1	0	0	0	0	0	1428	0%
Senior Rx. Op.*	1539	118	96	372	358	82	294	39	180	1071	17%
Rx. Op.	1539	118	96	372	358	82	294	39	180	1071	17%
Auxiliary Op.	8	1	1	1	1	0	0	0	4	952	4%
Maint. Mech.	211	0	11	83	23	9	84	0	1	1666	.06%
I&C Tech.	191	0	3	76	9	20	72	10	2	952	.2%
Chemistry Tech.	0	0	0	0	0	0	0	0	0	135	0%
HP Tech.	0	0	0	0	0	0	0	0	0	1428	0%
Engineers	3	0	2	0	0	1	0	0	0	1666	0%
Security Guard	0	0	0	0	0	0	0	0	0	952	0%
QA/QC Tech.	0	0	0	0	0	0	0	0	0	714	0%
Contractor Pers.	6	0	0	0	0	6	0	0	0	2142	0%
Operations Org.	0	0	0	0	0	0	0	0	0	595	0%
Maintenance Org.	0	0	0	0	0	0	0	0	0	1428	0%
I&C Org.	0	0	0	0	0	0	0	0	0	714	0%
Chemistry Org.	0	0	0	0	0	0	0	0	0	714	0%
HP Org.	0	0	0	0	0	0	0	0	0	1666	0%
Engineering Org.	0	0	0	0	0	0	0	0	0	1666	0%
Plant Management	4	0	0	0	0	4	0	0	0	1666	0%
QA/QC Org.	0	0	0	0	0	0	0	0	0	1904	0%
Security Org.	0	0	0	0	0	0	0	0	0	1666	0%
Off-site Pers.	0	0	0	0	0	0	0	0	0	680	0%
TOTALS	1976	119	112	533	396	124	450	49	193	30292	.64%

*PRAs failed to distinguish between Senior Reactor Operators and Reactor Operators. Therefore, in the analysis, the same data were applied to each category.

Inspection of the table indicates that of 24 personnel categories for which data are needed, 10 are addressed at least once across the 19 PRAs (Column 1). Of these 10, six are addressed through at least one complete data record (Column 9). Inspection of Column 9 indicates that 180 of the total 193 (93%) of the complete data records involve the Senior Operator/Reactor Operator combination. Finally, of the 30,292 complete and unique data records required for resolution, or partial resolution of the 127 Type A and B issues identified in Section 3, 193 or 0.64% of those records are provided from currently available PRAs.

4.2 Analysis of Data on Selected Categories

Tables 4.2 through 4.4 display HRA/PRA data record summaries for the most frequently analyzed personnel categories (i.e., Senior Operator/Reactor Operator Combination, Maintenance Mechanic, Instrumentation, and Control Technician).

Table 4.2 uses the same format at Table 4.1, and displays HRA/PRA data records for the Senior Operator/Reactor Operator category by action across all 19 PRAs. Table 4.2 indicates that the vast majority of records involve the operation of one system or another (Columns 4 and 6). Table 4.2 also indicates that approximately 18% of the data needs of issues involving the Senior Operator/Reactor Operator category are satisfied (Column 8 total divided by Column 9 total).

Tables 4.3 and 4.4 display the Maintenance Mechanic and Instrumentation and Control Technician data, respectively. As might be expected, data records in the HRA/PRA data base focus on maintaining systems (Table 4.3) and calibrating systems (Table 4.4). As indicated in Table 4.3, 0.08% of issue data requirements involving the Maintenance Mechanic are satisfied; whereas in Table 4.4, approximately 0.27% of the Instrumentation and Control Technician requirements are satisfied.

Table 4.2 Analysis of Operator HRA/PRA Data

	Column 1	2	3	4	5	6	7	8	9	10
	Number of Partial HRA/PRA Data Records									
Action	Action	PSF/ Situation	PSF/ System	Situation/ System	PSF	System	Situation	No. of Complete Records	No. of Issue Data Records Req'd.	Percent Addressed
Testing	9	2	49	56	26	140	1	2	51	4%
Operating	41	86	38	296	173	138	37	138	119	116%
Monitoring	9	8	2	4	44	0	0	6	119	5%
Inspecting	1	3	4	1	59	4	1	1	119	.8%
Checking	7	0	2	10	42	12	0	8	119	7%
Deciding	10	11	0	2	6	0	0	18	119	15%
Managing	0	0	0	0	0	0	0	0	119	0%
Communicating	3	0	0	1	5	0	0	2	119	2%
Calibrating	0	0	0	0	0	0	0	0	119	0%
Responding	2	8	1	2	3	0	0	5	119	4%
Maintaining	0	0	0	0	0	0	0	0	0	0%
TOTALS	82	118	96	372	358	294	39	180	1003	18%

Table 4.3 Analysis of Maintenance Technician HRA/PRA Data

	Column 1	2	3	4	5	6	7	8	9	10
	Number of Partial HRA/PRA Data Records									
Action	Action	PSF/ Situation	PSF/ System	Situation/ System	PSF	System	Situation	No. of Complete Records	No. of Issue Data Records Req'd.	Percent Addressed
Testing	0	0	0	0	0	0	0	0	0	0%
Operating	0	0	0	0	0	0	0	0	0	0%
Monitoring	0	0	0	0	0	0	0	0	119	0%
Inspecting	0	0	0	0	0	0	0	0	238	0%
Checking	0	0	0	0	0	0	0	0	238	0%
Deciding	0	0	0	0	0	0	0	0	119	0%
Managing	0	0	0	0	0	0	0	0	0	0%
Communicating	0	0	0	0	0	0	0	0	238	0%
Calibrating	0	0	0	0	0	0	0	0	0	0%
Responding	0	0	0	0	0	0	0	0	238	0%
Maintaining	9	0	11	83	23	84	0	1	102	.9%
TOTALS	9	0	11	83	23	84	0	1	1292	.08%

Table 4.4 Analysis of I&C Technician HRA/PRA Data

	Column 1	2	3	4	5	6	7	8	9	10
	Number of Partial HRA/PRA Data Records									
Action	Action	PSF/ Situation	PSF/ System	Situation/ System	PSF	System	Situation	No. of Complete Records	No. of Issue Data Records Req'd.	Percent Addressed
Testing	0	0	0	0	0	0	0	0	51	0%
Operating	0	0	0	0	0	0	0	0	0	0%
Monitoring	0	0	0	0	0	0	0	0	51	0%
Inspecting	0	0	0	0	0	0	0	0	119	0%
Checking	8	0	0	0	0	0	0	0	119	0%
Deciding	0	0	0	0	0	0	0	0	119	0%
Managing	0	0	0	0	0	0	0	0	0	0%
Communicating	0	0	0	0	0	0	0	0	119	0%
Calibrating	20	0	3	76	9	64	9	2	51	4%
Responding	0	0	0	0	0	0	0	0	119	0%
Maintaining	0	0	0	0	0	0	0	0	0	0%
TOTALS	28	0	3	76	9	64	9	2	748	.27%

5.0 SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

5.1 Summary

This report describes research aimed at assessing the extent to which available HRA/PRA data address a representative set of human risk issues of immediate concern to NRC.

All of the human risk data and associated information presented in 19 PRAs were identified, collected, and stored on a computer. This produced a collection of 1976 data records containing the point estimate for each human error considered along with information on the personnel, actions, PSFs, situations, and systems involved, if available.

In order to assess the extent to which the HRA/PRA data collected address the human risk issues currently facing NRC, a list of working level issues was developed. First, human risk questions relevant to Generic Safety Issues (NUREGs-0371, -0471, -0660; NUREG/CR-0933) were compiled into an initial list of working level issues (i.e., questions needing to be addressed to resolve the Generic Safety Issues). This list was refined, expanded, and clarified using NRC planning documents (NUREG-0985; NUREG/CRs-2833, -3250), and interviews with 28 cognizant NRC staff members. The final working level issues list is presented in Appendix B.

The data needed to address all of these issues were systematically identified. This was done by breaking the issue into its elements in the categories of personnel, actions, PSFs, situation, and systems for which data are needed. The issue data records generated in this manner were then compared with the HRA/PRA data records collected from 19 PRAs. The extent to which the HRA/PRA data meet the requirements of issue data records needed to address the issues was assessed.

5.2 Conclusions

It can be seen from the tables presented in Section 4 that the vast majority of HRA/PRA data in existing PRAs are not accompanied by sufficient information to clarify the conditions surrounding the human errors considered or why they were considered important. In addition, the method by which these data were generated is virtually never identified. As a result, most HRA/PRA data extracted from published PRAs cannot be used to address most human risk issues associated with efforts to reduce risk without additional analysis being performed. Further, as documented, an existing PRA cannot be used to make an evaluation of the effects of changes, such as new retrofit requirements or new information, on risk. If information on each category of data elements accompanied each human error considered in the PRA document, the methods used to generate these data were made explicit and the process of identifying and quantifying critical human errors documented, it would be possible to address significantly more human risk issues than can now be addressed using existing PRA results.

In addition, it can be observed that a great deal of the information required to address human risk issues cannot be stated in terms of probabilities of single human errors. These are the Types B and C issues (accounting for 85% of all working level issues). Further work is being done in this research program to develop means of acquiring, manipulating, and considering information from non-PRA sources to address additional human risk issues.

From the comparison of currently available HRA/PRA data and human risk issues of immediate concern to NRC, the following conclusions are drawn:

- Only 15% (26 of 175) of the working level issues identified in this study as Type A issues could be directly compared to available HRA/PRA data. Only an additional 58% (101 of 175) could be compared to these data. This appears to be a result of the tremendous emphasis on the quantitative aspects of individual human performance in PRAs. Many PRAs do discuss qualitative aspects of human performance, but no systematic way of interpreting this information could be found.

- Less than 1% of the data needed to address the Types A and B issues were found in the 19 PRAs analyzed in this report. This may be attributable to the relative lack of input to the PRA process from qualified HRA specialists. Their full participation in the PRA process (i.e., from start to finish) would ameliorate this problem to some extent. In addition, more systematic documentation of the HRA segments of the PRA may make PRA data more directly applicable to a broader range of human risk issues. Documentation should include a complete explanation of HRA/PRA methods, data sources, sensitivity analyses, and results.

- Among the 19 PRAs analyzed in this study, 93% (180 of 193) of the complete data records identified had to do with the actions of Operators. On the other hand, human risk issues were found to be associated with 24 personnel categories of which only 8% are operators.

This suggests that modifications in the otherwise data-rich process and documentation involved in a PRA would yield substantially more information of use in the regulatory area. This is especially true in terms of addressing human risk issues. Several efforts are underway which offer a vehicle for initiating such modifications. For example, the SHARP process (EPRI-XXXX,) for better integration of HRA into PRA provides a framework for improving the consideration of human errors in PRAs. As new methods for generating HRA/PRA data and better ways of using these data are developed in this and other research programs SHARP may offer a framework for integration of these methods into the PRA process. Another effort which would benefit from further development of HRA methods and means to use resultant HRA/PRA data is the Probabilistic Safety Analysis (PSA) effort (NUREG/CR-2815, 1985). A principal objective of the PSA effort is to make risk assessments of nuclear power plants comparable to each other as well as more useful in addressing issues related to retrofit requirements. Another PSA objective is to allow for risk assessments to be useful in assessing new information on risk reduction as it becomes available.

Work continues in this effort to develop ways of using HRA/PRA data more effectively in addressing human risk issues of concern to NRC. The products of this research will be documented in the next report in this series and will be useful in efforts, by both NRC and industry, to address a much broader range of issues than is currently the case.

5.3 Recommendations

The objective of making risk assessments more useful is the essence of many PRA-related efforts in both industry and NRC. The research program described in this report is aimed at supporting those efforts. Full consideration should be given to better documenting the consideration of HRA in PRAs. Fuller consideration will necessitate more use of qualified HRA specialists with the training and background necessary to document the relevant elements of human errors and report information necessary to use HRA/PRA data in addressing issues of concern to NRC. This study has lead to the following recommendations:

- The HRA segment of the PRA process should be improved so that it considers and presents both quantitative and qualitative data directed toward both risk qualification and risk reduction at the plant level.

- The HRA segment of a PRA process should be documented so that it can be used as a technical basis to address a broader range of human risk issues of immediate and long-term concern to NRC.

- HRA information and data should be presented and formated so that it can be systematically used along with information and data from non-PRA sources to address a broader range of human risk issues of immediate and long-term concern to NRC.

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APPENDIX A

GENERIC SAFETY ISSUES AS THEY RELATE TO WORKING LEVEL ISSUES AND DATA ELEMENTS

In this appendix, each Generic Safety Issue is listed in Column 1. Corresponding final working level issue identifiers are listed in Column 2. The complete list of final working level issues identified in Column 2 is presented in Appendix B. Corresponding data elements required to address the Generic Safety Issues are listed in Column 3-7. An entry "ALL" in Columns 3-7 means the data on all of the corresponding entries in that category of Table A.1 are required to satisfy Generic Safety Issues data needs.

Table A.1 Categories and Elements for Developing Data Elements From Working Level Issues.

Individuals and Groups	Performance Shaping Factors	Actions	Situations	Systems
Plant Manager	Equipment Design	Testing	Loss of Coolant Accident	Safety-related Systems
Shift Supervisor	Workplace Layout	Operating	Loss of Off-site Power	Structures
Shift Tech. Advisor	Habitability	Monitoring	Other Transients	Non-safety Systems
Senior Reactor Op.	Time Available	Inspecting	System Isolation	
Reactor Operator	Staffing	Checking	Normal Operation	
Auxiliary Operator	Organizational Climate	Deciding	External Event	
Maintenance Mech.	Job-related Training	Managing	Outage	
I&C Tech.	Information Feedback	Communicating		
Chemistry Tech.	Task Complexity	Calibrating		
Health Physics Tech.	Regulations	Responding		
Engineers	Stress	Maintaining		
Security Guard	Fatigue			
QA/QC Tech.	Attitude			
Contractor Personnel	Job-related Experience			
Operations Org.	Fitness for Duty			
Maintenance Org.	Perceived Risks			
I&C Org.	Procedures			
Chemistry Org.				
Health Physics Org.				
Engineering Org.				
Plant Management				
QA/QC Org.				
Security Org.				
Off-site Response Personnel				

Appendix A

Generic Safety Issues, Working Level Issues, and Data Element Tables.

Table 1

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
IA1 - Operating Personnel and Staffing	1.15, 1.19-1.22, 1.26-1.28, 2.1, 2.3, 2.5, 2.7, 2.9, 2.16, 3.1, 3.3-3.5, 3.7, 3.10-3.13, 3.15, 6.1, 6.7, 4.2, 7.9	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	Staffing Job-related Training Job-related Experience Fitness for Duty	Operating Deciding Communicating	All	All
IA2 - Training and Qualifications of Operating Personnel	1.21, 1.22, 2.1-2.10, 2.14-2.19, 3.1, 3.4, 3.5, 3.7, 3.9-3.15, 4.11, 4.12, 4.18, 6.1, 9.10, 9.14	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	Job-related Training Task Complexity Job-related Experience	All Except Calibrating Maintaining	All	All
IA3 - Licensing and Requalification of Operating Personnel	1.1-1.3, 1.5, 1.7, 1.9-1.11, 1.16, 1.19, 1.21, 1.22, 1.24-1.27, 2.1, 2.3, 2.7, 2.9, 2.10, 2.16, 3.3-3.5, 3.7, 3.9-3.16, 6.9, 7.2, 7.3, 7.8, 7.10, 8.5-8.7, 8.10	All	Staffing Organizational Climate Job-related Training Regulations Attitude Job-related Experience Fitness for Duty	All	All	All
IA4 - Simulator Use and Development	1.9, 1.11, 1.27, 1.28, 2.1, 2.7-2.9, 2.18, 3.1-3.4, 3.7, 3.9, 3.11, 3.14, 5.1, 5.2, 5.7-5.10, 5.14, 5.19, 5.20, 6.18	Senior Reactor Op. Reactor Operator	All	All Except Calibrating Maintaining	All	All
IB1 - Management for Operations	1.2-1.4, 1.7, 1.8, 1.15, 1.16, 1.19-1.22, 2.12, 2.13, 2.16, 2.19, 3.3, 3.6, 4.1, 4.4-4.6, 4.8, 4.9, 4.12, 4.14, 4.17-4.19, 6.10, 7.2, 7.3, 7.5, 7.8, 7.12, 8.11, 8.12, 9.6, 9.7, 9.9, 9.11, 9.13	All	All Except Equipment Design Workplace Layout Habitability	All	All	All
IB2 - Inspection of Operating Reactors	1.19, 2.10, 2.14, 4.11-4.14, 5.12, 5.13, 6.8-6.10, 7.3, 7.5, 7.8, 7.9, 9.1-9.3, 9.5-9.13, 9.15	All	Regulations	All	All	All

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
IC - Operating Procedures	1.9, 1.10, 1.19, 1.20, 1.23, 2.5, 2.7, 3.6, 3.7, 4.1-4.15, 4.17, 5.6, 5.8, 5.9, 5.16, 6.1, 6.4, 6.8, 6.14, 7.12, 8.1, 8.8, 8.9, 9.13	All	Procedures	All	All	All
ID - Control Room Design	1.9, 2.20, 3.7, 4.3, 4.15, 4.19, 5.1-5.3, 5.5, 5.7-5.12, 5.14, 5.15, 5.19, 5.20, 6.4, 6.6, 9.6, 9.7, 9.9, 9.13	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	Equipment Design Workplace Layout Habitability Job-related Training Information Feedback Task Complexity Procedures	All Except Calibrating Maintaining	All	All
IE - Analysis and Dissemination of Operating Experience	1.9, 1.12, 1.14,1.24, 3.8, 5.7, 6.1, 6.2, 6.5, 6.6, 6.8-6.12, 6.14, 6.16, 7.1, 7.7, 8.3, 9.4, 9.7	All	All	All	All	All
IF - Quality Assurance	1.16, 1.29, 2.21, 4.11, 4.20, 7.2, 7.13	QA/QC Tech. QA/QC Org.	All	Inspecting Managing Communicating	System Isolation Normal Operation Outage	All
IG - Preoperational and Low-Power Testing	1.23, 2.14, 2.22, 4.21	All	Time Available Staffing Job-related Training Information Feedback Regulations Procedures	All	System Isolation Normal Operation Outage	All
IIB - Consideration of Degraded or Melted Cores in Safety Review	2.1, 2.5, 2.16, 3.14, 5.21	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org.	Procedures	Operating Deciding Responding	All Except System Isolation Normal Operation Outage	All
IID - Reactor Coolant System Relief and Safety Valves	5.22, 9.13	Senior Reactor Op. Reactor Operator Operations Org.	Equipment Design Workplace Layout Habitability Procedures	Testing	System Isolation Normal Operation Outage	Safety-related Systems

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
IIE - System Design	2.1, 3.1, 3.4, 3.14, 4.2, 4.3, 4.15, 4.16, 4.19, 5.8, 9.6, 9.16	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. I&C Tech. QA/QC Tech. Operations Org. Maintenance Org. I&C Org. QA/QC Org.	Equipment Design Information Feedback Task Complexity Regulations Procedures	Testing Operating Maintaining	All	All
IIF - Instrumentation and Controls	1.11, 2.1, 3.2, 4.2, 5.1, 5.2, 5.7-5.9, 5.14, 5.15, 6.2, 6.6, 9.16	Shift Supervisor Senior Reactor Op. Reactor Operator I&C Tech. Operations Org. I&C Org.	Equipment Design Information Feedback Task Complexity	All	All	All
IIH - TMI-2 Cleanup and Examination	7.7	All	All	Operating Monitoring Inspecting Managing	Outage	All
IIJ - General Implications of TMI for Design and Construction Activities	7.7, 7.10	Plant Management	Information Feedback	Deciding	Outage	All
IIIA - Emergency Preparedness and Radiation Effects	1.15, 2.5, 2.13, 2.14, 2.19, 3.6, 5.9, 5.19, 5.21, 7.12, 8.11, 8.12	Plant Management Off-site Response Personnel	Equipment Design Staffing Organizational Climate Regulations	Monitoring Deciding Responding	All Except System Isolation Normal Operation Outage	All
IIIB - Emergency Preparedness of State and Local Governments	2.13	Off-site Response Personnel	All	Responding	All Except System Isolation Normal Operation Outage	All
IIIC - Public Information	2.4, 2.19	Plant Management	Information Feedback	Managing	All	All

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa- tions	Systems
IIID - Radiation Protection	2.4, 2.13, 2.19, 4.16, 9.10	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Plant Management	All	Testing Operating Monitoring Responding	All	All
IVB - Issuance of Instructions and Information to Licensees	7.7	Plant Management	Information Feedback	Monitoring	All	All
IVC - Extend Lessons Learned to Licensed Activities Other than Power Reactors	3.8					
IVD - NRC St[illegible] Training	1.13, 2.8, 2.17	All	Job-related Training	Monitoring	All	All
IVF - Financi[illegible] Disincentiv[illegible] Safety	2.22, 7.8, 7.10	Plant Management	Organizational Climate	Managing	Normal Operation	All
A-3 - Westinghouse Steam Generator Tube Integrity	1.16, 1.24, 2.4, 4.2, 4.16, 6.2, 6.14, 9.1, 9.6, 9.8	Chemistry Org. Engineering Org. Plant Management QA/QC Org.	All	Monitoring Inspecting	All Except System Isolation	All
A-4 - CE Steam Generator Tube Integrity	1.16, 1.24, 2.4, 4.2, 4.16, 6.2, 6.14, 9.1, 9.6, 9.8	Chemistry Org. Engineering Org. Plant Management QA/QC Org.	All	Monitoring Inspecting	All Except System Isolation	All
A-5 - B&W Steam Generator Tube Integrity	1.16, 1.24, 2.4, 4.2, 4.16, 6.2, 6.14, 9.1, 9.6, 9.8	Chemistry Org. Engineering Org. Plant Management QA/QC Org.	All	Monitoring Inspecting	All Except System Isolation	All

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
A-13 - Snubber Operability Assurance	4.16, 4.20, 9.6, 9.7	Maintenance Mech. Maintenance Org. Plant Management	All	Monitoring Inspecting Checking	All	Structures
A-14 - Flaw Detection	9.6	Maintenance Mech. Maintenance Org. Plant Management	All	Inspecting	All	All
A-16 - Steam Effects on BWR Core Spray Distribution	9.6	Operations Org.	All	Testing	All	Safety-related Systems
A-23 - Containment Building Response	9.6	Maintenance Org. Plant Management	All	Testing	Normal Operation Outage	Structures
A-29 - Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	1.26, 4.17, 6.20, 7.11, 7.12, 8.1-8.6, 8.8, 8.9, 8.12	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. I&C Tech. Security Guard Contractor Personnel Operations Org. Maintenance Org. I&C Org. Security Org. Off-site Response Personnel	Workplace Layout Time Available Staffing Organizational Climate Task Complexity Regulations	Inspecting Responding Maintaining	Normal Operation	All
A-30 - Adequacy of Safety-Related DC Power Supplies	4.10, 4.16, 6.12, 6.13, 9.6, 9.9, 9.13	Maintenance Mech. Maintenance Org.	Equipment Design Workplace Layout Procedures	Testing	All	Safety-related Systems
A-34 - Instruments for Monitoring Radiation and Process Variables During Accidents	3.6, 5.9, 5.19	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org.	Equipment Design Information Feedback Task Complexity Procedures	Operating Monitoring Checking Deciding Responding	All Except System Isolation Normal Operation Outage	All

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
A-35 - Adequacy of Offsite Power Systems	4.16, 9.6	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org.	All	Testing	All Except System Isolation Normal Operation Outage	All
A-40 - Seismic Design Criteria - Short Term Program	2.5, 2.16, 5.19	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Plant Management	All	Operating Responding	External Event	All
A-41 - Long Term Seismic Program	2.5, 2.16, 3.14, 5.19	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Plant Management	Job-related Training	Operating Responding	External Event	All
A-43 - Containment Emergency Sump Performance	1.9, 2.5, 3.7	Shift Supervisor Senior Reactor Op. Reactor Operator	Equipment Design Procedures	Testing Operating Responding	All Except System Isolation Normal Operation Outage	Safety-related Systems
A-44 - Station Blackout	1.9, 2.7, 5.6, 9.6, 9.9, 9.13	Shift Supervisor Senior Reactor Op. Reactor Operator	All	Operating Responding	Loss of Off-site Power Other Transients	All
A-45 - Shutdown Decay Heat Removal Requirements	1.9, 1.20, 2.7, 2.9, 3.7	Shift Supervisor Senior Reactor Op. Reactor Operator	Equipment Design Stress Procedures	Operating Responding	All Except System Isolation Normal Operation Outage	Safety-related Systems
B-4 - ECCS Reliability	4.16, 4.19, 5.22, 9.6, 9.13, 9.16	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Plant Management	Equipment Design Procedures	Testing Inspecting Maintaining	All Except System Isolation Normal Operation Outage	Safety-related Systems
B-7 - Secondary Accident Consequence Modeling	3.14					

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa- tions	Systems
B-8 - Locking Out of ECCS Power Operated Valves	3.2, 5.2, 5.8, 5.9, 5.20	Senior Reactor Op. Reactor Operator Maintenance Mech. I&C Tech. Operations Org. Maintenance Org. I&C Org.	Equipment Design Task Complexity Procedures	Testing Checking	All Except System Isolation Normal Operation Outage	Safety-related Systems
B-11 - Subcompartment Standard Problems	3.14	Senior Reactor Op. Reactor Operator	Equipment Design Workplace Layout	Responding	All Except System Isolation Normal Operation Outage	All
B-17 - Criteria for Safety Related Operator Actions	1.9, 4.2, 4.3, 5.5, 5.9, 5.16, 5.20, 6.2, 6.22	Senior Reactor Op. Reactor Operator	All	Responding	All	All
B-18 - Vortex Suppression Requirements for Containment Sumps	9.6	Senior Reactor Op. Reactor Operator	Equipment Design	Responding	All Except System Isolation Normal Operation Outage	Safety-related Systems
B-23 - LMFBR Fuel	7.7	Senior Reactor Op. Reactor Operator	Equipment Design Procedures	Responding	All Except System Isolation Normal Operation Outage	All
B-34 - Occupational Radiation Exposure Reduction	1.13, 1.16, 1.19, 1.24, 2.2, 2.4, 4.2, 4.16, 6.2, 6.10, 7.2, 9.6	All	Equipment Design Workplace Layout Habitability Time Available Staffing Task Complexity Procedures	Testing Inspecting Checking Calibrating Maintaining	System Isolation Normal Operation Outage	All
B-36 - Develop Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption	4.2, 9.6, 9.13	Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. I&C Tech.	All	Tesing Maintaining	Normal Operation Outage	Safety-related Systems

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
B-42 - Socio-economic Environmental Impacts	1.13, 1.16, 1.18	All	Staffing Organizational Climate Regulations	Managing Communicating	Normal Operation	All
B-47 - Inservice Inspection of Supports-Classes 1, 2, 3, and MC Components	9.6	Maintenance Mech. Maintenance Org. Plant Management	All	Inspecting	Normal Operation Outage	Structures
B-48 - BWR CRD Mechanical Failure (Collet) Housing	9.6	Maintenance Mech. Maintenace Org. Plant Management	All	Inspecting	Normal Operation Outage	Safety-related Systems
B-49 - Inservice Criteria and Corrosion Pre-vention Criteria for Containments	9.6	Maintenance Mech. Maintenance Org. Plant Management	All	Inspecting	Normal Operation Outage	Structures
B-50 - Post-Operating Basis Earthquake Inspection	9.6	Plant Management	Workplace Layout	Inspecting	External Event	All
B-53 - Load Break Switch	9.6	I&C Tech. I&C Org.	All	Testing	Normal Operation Outage	Safety-related Systems
B-55 - Improved Reliability of Target Rock Safety-Relief Valves	4.2, 9.6	Senior Reactor Op. Reactor Operator Maintenance Mech. Operations Org. Maintenance Org.	All	Operating Maintaining	Normal Operation Outage	Safety-related Systems
B-56 - Diesel Reliability	5.6, 9.6, 9.13	Maintenance Mech. I&C Tech Maintenance Org. I&C Org.	All	Testing Inspecting Checking Maintaining	Normal Operation Outage	Safety-related Systems

Table I (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
B-58 - Passive Mechanical Failures	5.2, 5.20, 5.22, 6.6, 9.6	Senior Reactor Op. Reactor Operator Operations Org.	All	Tesing	Normal Operation Outage	Safety-related Systems
B-60 - Loose Parts Monitoring System	5.23	Senior Reactor Op. Reactor Operator Operations Org.	All	Monitoring	Normal Operation	Safety-related Systems
B-61 - Allowable ECCS Equipment Outage Periods	4.16, 9.6, 9.13, 9.16	Senior Reactor Op. Reactor Operator Maintenance Mech. Operations Org. Maintenance Org.	Time Available Staffing Regulations Procedures	Tesing Managing Maintaining	Outage	Safety-related Systems
B-66 - Control Room Infiltration Measurements	2.1, 2.5, 2.9, 2.18, 2.19, 3.6, 5.1, 5.4, 6.9	Plant Manager Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Plant Management	Workplace Layout Habitability Perceived Risk	Operating Responding	All Except System Isolation Normal Operation Outage	All
B-71 - Incident Response	1.15, 1.30, 2.13, 5.21	Plant Management Off-site Response Personnel	Equipment Design Information Feedback	Managing Communicating	All Except System Isolation Normal Operation Outage	All
C-11 - Assessment of Failure and Reliability of Pumps and Valves	9.6	Senior Reactor Op. Reactor Operator Operations Org.	All	Testing Inspecting	System Isolation Normal Operation Outage	Safety-related Systems
D-1 - Advisability of a Seismic Scram	5.19, 6.22	Senior Reactor Op. Reactor Operator	Equipment Design	Responding	External Event	Safety-related Systems

Table 1
(Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
HFO 1.1.1 - NPP Staffing Requirements	1.1, 1.7-1.9, 1.12-1.13 1.15-1.16, 1.18-1.20, 1.23, 1.29-1.30, 2.9, 2.19, 3.3, 4.3-4.4, 4.6, 4.14, 5.1, 5.3-5.4, 5.8-5.9, 5.14, 5.18-5.19, 5.21, 6.1, 6.22, 7.2, 7.6, 7.8, 7.12, 9.10-9.11, 9.13	All	Equipment Design Workplace Layout Staffing Information Feedback Task Complexity	All	All	All
HFO 1.1.2 - NPP Personnel Qualification Requirements	1.1-1.4, 1.12-1.13, 1.17, 1.20-1.23, 1.25-1.26, 1.29-1.30, 2.16-2.17, 2.1, 2.3-2.14, 2.16-2.17, 2.20-2.21, 3.4, 3.9-3.13, 3.15, 4.3, 4.12, 5.8, 5.15, 5.19, 5.21, 6.2, 6.7, 6.22, 7.3, 8.5, 8.8	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	Job-related Training Job-related Experience	Testing Operation Monitoring Inspecting Checking Deciding Managing Communicating Responding	All	All
HFO 1.1.3 - Guidance on Limits and Conditions of Shift Work	1.10, 1.16, 1.19, 1.24, 3.11, 4.8, 5.5, 6.9, 6.21, 7.2, 7.4, 7.6, 9.10	All	Staffing Organizational Climate Fatigue	All	All	All
HFO 1.1.4 - Fitness for Duty	1.5, 6.9, 7.10-7.11, 8.5, 8.7	All	Fitness for Duty	All	All	All
HFO 1.2.1 - Development of Training Regulation and Guidance	1.2-1.3, 1.17, 1.21, 1.23, 2.1-2.19, 3.4-3.5, 3.9, 3.11, 3.12, 3.14, 3.16, 4.12, 5.2, 5.5, 5.15, 6.9, 9.11	All	Organizational Climate Job-related Training Job-related Experience	All	All	All
HFO 1.2.2 - NRC Training Evaluation Program	1.2, 1.21, 2.1-2.19, 3.4, 3.11-3.12, 3.15-3.16, 4.12 4.14, 5.5, 5.9, 5.13, 6.8, 7.6, 9.14	All	Organizational Climate Joo-related Training	All	All	All

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
HFO 1.3.1 - The Examination Content	1.2, 1.11, 1.20-1.21, 1.27-1.28, 2.1, 2.5-2.7, 2.9, 2.16-2.17, 3.1-3.16, 4.3, 4.12, 5.1-5.2, 5.6, 5.8-5.9, 5.15-5.16,5.19, 6.2, 6.8, 6.18, 7.9	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator	Job-related Training	Testing Operating Monitoring Inspecting Checking Deciding Managing Communicating Responding	All	All
HFO 1.3.2 - The Examination Process	1.2, 2.1, 2.3, 2.6-2.7, 2.10, 3.1-3.16, 4.3, 4.12, 5.11, 5.15-5.16, 5.19, 6.2, 6.18	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator	Job-related Training	Testing Operating Monitoring Inspecting Checking Deciding Managing Communicating Responding	All	All
HFO 1.4.1 - Procedures Guidance and Criteria	1.11, 2.16, 4.1-4.19, 6.3-6.4, 6.22, 9.1, 9.3, 9.9, 9.13	All	Procedures	All	All	All
HFO 1.5.1 - MMI Guidance for Existing Designs	3.7, 4.2, 5.1-5.21, 6.4, 6.6, 6.22, 9.6, 9.13, 9.16	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Workplace Layout Habitability Information Feedback	Testing Operating Monitoring Inspecting Checking Deciding Managing Communicating Responding	All	All
HFO 1.5.2 - MMI Guidance for Designs Based on Advanced Technologies	1.14, 2.20, 4.2, 5.1-5.21, 6.4, 6.22, 9.6, 9.13, 9.16	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Workplace Layout Information Feedback	Testing Operating Monitoring Inspecting Checking Deciding Managing Communicating Responding	All	All

Table 1 (Continued)

1	2	3	4	5	6	7
		Data Elements				
Generic Safety Issues TMI Action Item	Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
HFO 1.6.1 - Regulatory Position on Management and Organization at Operating Reactors	1.8, 7.1-7.12	Plant Manager Plant Management	Staffing Organizational Climate Job-related Training Information Feedback Task Complexity Regulations Attitude Job-related Experience Perceived Risks	Checking Deciding Managing Communicating Responding	All	All
HFO 1.6.2 - NRC Managment and Organization Guidelines and Assessment Procedures for Operating License Reviews	1.8, 7.1-7.12	Plant Manager Plant Management	Staffing Organizational Climate Job-related Training Information Feedback Task Complexity Regulations Attitude Job-related Experience Perceived Risks	Checking Deciding Managing Communicating	All	All
HFO 1.7.1 - Human Error Data Acquisition	1.1-1.6, 1.9-1.10, 1.21-1.22, 2.1-2.4, 2.19-2.22, 3.1-3.3, 3.5, 3.7, 3.11-3.12, 4.3, 4.8, 4.12-4.16, 5.1-5.21, 6.1-6.21, 7.3, 7.6, 8.3, 8.7, 8.9, 8.13, 9.6, 9.13	All	All	All	All	All

APPENDIX B

FINAL WORKING LEVEL ISSUES LIST

This appendix contains a list of the 175 working level issues in their final form. These issues reflect the results of a review of Generic Safety Issues, NRC planning documents, and interviews with cognizant NRC staff. Each working level issue is a question which brings rise to specific needs for data information.

After each issue the issue type is designated with an A, B, or C. These issue types are defined as follows:

Type A. Working level issues for which a set of quantitative issue data records can be generated that, if addressed by competent data, provide a complete technical basis for addressing the issue in question.

Type B. Working level issues for which a partial set of quantitative issue data records can be generated, but for which additional data not currently provided in PRAs may be needed to establish a complete technical basis for addressing the issue in question. These issues may require additional information such as operational history data or information on the availability of a sufficient work force.

Type C. Working level issues which require data or information not compatible to the form of data records to provide a technical basis. Instead, these issues typically require information in forms other than the data record format such as data on operational history or information on the availability of a sufficient work force.

1. Staffing and Qualifications - Working Level Issues

1.1 What are the bases on which all job positions can be compared in terms of human performance? (A)

1.2 How does training affect human performance? (A)

1.3 How do qualifications affect human performance? (A)

1.4 How do job performance reviews affect human performance? (B)

1.5 How do fitness-for-duty requirements affect human performance? (A)

1.6 How do behavioral observation programs affect human performance? (B)

1.7 Does the STA job position reduce risk significantly? (B)

1.8 Do senior managers' actions affect risk significantly? (B)

1.9 What human errors are most important during recovery activities? (B)

1.10 How do shiftwork and overtime affect human performance? (A)

1.11 What are the most important occurrences to be aware of during normal operations? (B)

1.12 What is the relationship of staffing and qualifications levels with the operational history of licensees? (C)

1.13 What is the availability of qualified personnel for work in plants? (C)

1.14 What are the longitudinal trends on factors affecting human errors in plants? (B)

1.15 What is the optimal form of emergency staffing? (A)

1.16 What are the present staffing conditions in the industry? (C)

1.17 What are licensees' current personnel selection practices and criteria? (C)

1.18 What job vacancies currently exist in the industry? (C)

1.19 How can alternative crew staffing approaches used by the industry be evaluated? (B)

1.20 What is the contribution of engineering input to recovery activities and accident mitigation? (B)

1.21 How does formal education affect human performance? (A)

1.22 How does job-related experience affect human performance? (A)

1.23 What are the impacts of staffing and qualification requirements on the NTOL licensing process? (B)

1.24 What are the important tasks in various operations, I&C, maintenance, QA/QC, and security jobs? (B)

1.25 Should personnel besides operators be licensed by NRC? (B)

1.26 How should access authorization be related to other personnel qualifications? (B)

1.27 How can simulation experiments be used to support qualification requirements? (B)

1.28 What cognitive skills are required for accident management? (B)

1.29 What are the optimal staffing arrangement and qualification requirements for the QA/QC staff? (A)

1.30 How should an off-site emergency support center be staffed and what personnel qualifications should be required? (A)

2. Training - Working Level Issues

2.1 What is the optimal content for an operator training curriculum? (B)

2.2 What is the optimal content for a maintenance training curriculum? (B)

2.3 What is the optimal content and schedule for refresher training? (B)

2.4 What is the optimal content of training curriculum for other plant personnel? (B)

2.5 To what extent does operator training presently prepare operators for severe accident management? (B)

2.6 What qualifications should be required of training instructors and training program developers? (C)

2.7 How can simulators be used to enhance the training of licensed operators? (B)

2.8 How can simulators be used to enhance the training of plant personnel other than licensed operators? (B)

2.9 What specific normal, off-normal, and emergency conditions should be simulated for operator training? (B)

2.10 On what basis should plant training programs be evaluated? (B)

2.11 How do trainers and trainees perceive that training programs can be improved? (C)

2.12 To what extent does security training prepare personnel for safety-related events, as well as security-related events? (B)

2.13 What training is necessary for off-site response personnel? (B)

2.14 What is the optimal role of drills and other performance oriented training techniques? (B)

2.15 How useful is the ISD approach to training development? (C)

2.16 What training requirements are needed to prepare operators to respond adequately during plant conditions for which there are no procedures? (B)

2.17 What forms of hands-on training are needed? (B)

2.18 How can the fidelity of training simulators be best assured? (B)

2.19 What personnel should receive special training in the use of respirators for radiation protection? (B)

2.20 What types of training should be required for personnel using new job performance aids? (B)

2.21 What role can low power testing play in the process of training plant personnel? (B)

3. Licensing Examinations - Working Level Issues

3.1 What are the important accident sequences for each plant? (C)

3.2 What are the effects of response times available for recovery steps during those sequences? (B)

3.3 What is the impact of team behavior on operator performance? (B)

3.4 What are the most important knowledge, skills, and abilities for operators? (B)

3.5 What cognitive skills should operators be tested on? (B)

3.6 At what point in a sequence should the site be abandoned? (C)

3.7 What are the most important tasks during recovery activities? (B)

3.8 What useful information is available from nonpower reactors? (C)

3.9 What is the appropriate role for plant simulators in the examination process? (B)

3.10 What is the validity of the current licensing exam? (C)

3.11 What are the best methods for testing and measuring operator performance? (B)

3.12 How can cognitive skills be assessed in an examination? (C)

3.13 Should examination cutoff scores be established? (B)

3.14 What are the appropriate engineering models for use in programming simulators? (B)

3.15 What is the best format for requalification examinations? (B)

3.16 What are the optimal qualifications and training for licensing examiners? (C)

4. Procedures - Working Level Issues

4.1 What are the effects of different types of procedures on human performance? (A)

4.2 How should trade-offs between hardware changes and procedural changes be assessed? (B)

4.3 What is the optimal roll artificial intelligence in the control room? (B)

4.4 What is the "social domain" of the personnel using procedures in power plants? (C)

4.5 How often are procedures actually used in plant operation and maintenance? (B)

4.6 What is the impact of new advanced emergency operating procedures on operator performance? (A)

4.7 What is the frequency of events associated with the use of procedures? (C)

4.8 What is the impact of stress on the use of procedures? (A)

4.9 What are the types of procedure-associated errors that most impact risk? (B)

4.10 What are the most important sequential errors that can be avoided using procedures designed for that purpose? (B)

4.11 What procedures should be reviewed at an entire facility or just specific segments of the facility (e.g., only maintenance and operators)? (B)

4.12 What are the trade-offs between training and procedures and how can they be assessed? (B)

4.13 How can NRC foster respect for the value of procedures among licensee employees? (C)

4.14 What are the alternative formats for presenting procedurcs and how an they be evaluated? (B)

4.15 What types of procedures should be used for advanced display systems? (B)

4.16 What type of procedures are optimal for preventive and corrective maintenance? (B)

4.17 How do security procedures affect the ability of the operations staff to safely operate the plant? (A)

4.18 How are upgraded procedures integrated with existing procedures and what are the effects of that integration process? (B)

4.19 What are the optimal procedures to be used to minimize risk while isolating systems? (A)

4.20 What are the optimal form of QA/QC procedures? (A)

4.21 What are the optimal procedures for preoperational and low-power testing? (B)

5. Man-machine Interface - Working Level Issues

5.1 What are the impacts of control room design and modifications on operator performance? (A)

5.2 What is the relative importance of the alarms in control rooms? (B)

5.3 What are optimal review criteria for control room reviews? (B)

5.4 What are the impacts of local control station design and modifications on operator performance? (A)

5.5 Which interfaces are associated with high human error rates? (A)

5.6 What are the important aspects of human actions involving manually operated valves, diesel generators, and communications equipment? (B)

5.7 How should advanced display technologies be used in plant control rooms? (B)

5.8 What are the optimal means of managing information in the control room? (B)

5.9 What are the most important things for the operator to do during accident sequences? (B)

5.10 What are the best types of annunciators to use? (A)

5.11 How should advanced displays be assessed? (B)

5.12 How can NRC verify improvements in safety due to interface design modifications? (B)

5.13 What aspects of the man-machine interface are important in maintenance activities? (A)

5.14 Should plant control rooms be completely overhauled and modernized? (B)

5.15 How can operators best obtain a mental image of the plant's state while in the control room? (B)

5.16 How important are operations performed at local contol stations? (B)

5.17 Should local control stations be alarmed? (B)

5.18 What are the impacts of auxiliary operators on safety? (B)

5.19 What are the control/display requirements for operating crew needs subsequent to severe seismic event? (A)

5.20 What is the optimal functional allocation of alarms? (B)

5.21 What are the optimal roles for various personnel during severe accident management? (B)

5.22 What are the optimal ways of assuring safety valve positions? (B)

5.23 What are the optimal ways of monitoring loose parts in the reactor system? (B)

6. Human Reliability - Working Level Issues

6.1 What are the most important human activities in feed and bleed operations? (B)

6.2 What are the general effects of time available to perform tasks on human reliability? (A)

6.3 What is the utility of cognitive modeling? (B)

6.4 What is the utility of decision aids? (B)

6.5 How should errors of commission be treated in risk assessments and regulation? (C)

6.6 What are the optimal means of identifying valves, switches, meters, and so on. (B)

6.7 What are the relative characteristics of group versus individual errors? (B)

6.8 What human errors can initiate an accident? (C)

6.9 What factors actually affect human performance? (A)

6.10 What are the important errors in maintenance activities? (B)

6.11 What sources of human error data are there? (C)

6.12 What sources of dependency data are there? (C)

6.13 How should dependent errors be treated in risk assessment and regulation? (C)

6.14 How can field data be used to provide feedback to licensees? (B)

6.15 What are the optimal means of using structured expert judgment to estimate human error probabilities? (C)

6.16 What are the optimal means of acquiring, storing, and retrieving human error data? (C)

6.17 How can human performance be more fully integrated into risk assessment? (C)

6.18 Can training simulator data be used in risk assessment? (C)

6.19 How can performance modeling best be used in risk assessment? (C)

6.20 Can sabotage be integrated into existing risk assessment methods? (B)

6.21 Can models of performance be developed for performance shaping factors other than available time? (C)

6.22 What operator actions should be made self actuating rather than operator actuated? (B)

7. Management and Organization - Working Level Issues

7.1 What is the relationship between operational history and safety in plants? (C)

7.2 What is the number of personnel at each plant in each job position? (C)

7.3 How can the quality of management personnel at operating plants best be maintained? (B)

7.4 What are the demographics of personnel working at plants? (C)

7.5 What are the optimal forms of guidance and review criteria for management audits? (B)

7.6 What are the organizational structure and climate at each plant? (C)

7.7 What is the optimal way to provide feedback of experience to all plant managements? (B)

7.8 What major activities by management personnel are the most important in terms of safety? (C)

7.9 What external events are most important at each plant? (B)

7.10 What is the best way to assure the management capabilities of personnel at plants under construction? (C)

7.11 How do attitudes toward security affect site security at plants? (A)

7.12 How should operational, security, and off-site response personnel interact and communicate during an emergency? (B)

7.13 What are the optimal form of reporting requirements for QA/QC activities? (B)

8. Trustworthiness - Working Level Issues

8.1 What are the impacts of safeguards and security activities on risk? (B)

8.2 What are the appropriate uncertainty bounds for the probability of a sabotage event? (C)

8.3 What is the optimal way to acquire, store, retrieve, and report safeguards field data from licensees? (C)

8.4 What is the optimal way to avoid excessive vandalism at sites under construction and operation? (C)

8.5 What is the best way to screen personnel for trustworthiness and reliability? (B)

8.6 How can trustworthiness and reliability be ensured on a continuing basis for employees? (B)

8.7 What is the established worth of employee assistance programs? (B)

8.8 What is the optimal allocation of the functions of access control and access authorization screening? (C)

8.9 What is the optimal designation of vital areas considering both security and safety needs? (B)

8.10 To what extent should NRC become involved in regulation of fitness for duty measures? (C)

8.11 How should access authorization of off-site emergency response personnel best be handled? (B)

8.12 Should trustworthiness measures such as access controls be relaxed during a safety-related event and, if so, to what extent? (C)

8.13 To what extent does a behavioral observation program better assure trustworthiness? (C)

9. Maintenance - Working Level Issues

9.1 What is the optimal role of maintenance in preventing the aging of components? (B)

9.2 What are optimal requirements for assuring an adequate supply of spare parts at plants? (C)

9.3 What maintenance practices correlate with reactor scrams trips and safety system challenges? (C)

9.4 What sources of equipment performance data are available and how useful are they? (C)

9.5 What is the utility of the SALP ratings in evaluating maintenance practices? (C)

9.6 What are the optimal means of conducting test and surveillance activities in terms of risk? (B)

9.7 What components and systems at plants are routinely maintained and how often? (C)

9.8 What are the maintenance practices of other industries with similar needs? (C)

9.9 What are the critical considerations in taking equipment out of service for maintenance? (B)

9.10 How should maintenance tasks be staffed? (A)

9.11 Can organizational and management factors be identified that critically impact maintenance in operating nuclear power plants? (B)

9.12 What are the acceptable maintenance requirements, standards, and criteria for use in evaluating plant-specific maintenance programs and activities? (B)

9.13 What are the appropriate methods for validating and verifying correct performance of maintenance work, authorization, and control when systems or components are taken out of service? (B)

9.14 Can a technical basis be established for the training/qualification of maintenance personnel? (B)

9.15 What is the proper extent of regulatory activitity concerning maintenance of security systems and equipment? (B)

9.16 How can the availability of safety-related systems be improved? (B)

APPENDIX C

DATA ELEMENT TABLES FOR EACH WORKING LEVEL ISSUE

In this appendix, the data elements required to address each Type A and B working level issue identified in Appendix B are listed. An entry "ALL" in the data element tables means the data on all corresponding entries in that category of Table C.1 are required to satisfy that working level issue data needs.

Table C.1 Categories and Elements for Developing Data Elements From Working Level Issues.

Individuals and Groups	Performance Shaping Factors	Actions	Situations	Systems
Plant Manager	Equipment Design	Testing	Loss of Coolant Accident	Safety-related Systems
Shift Supervisor	Workplace Layout	Operating	Loss of Off-site Power	Structures
Shift Tech. Advisor	Habitability	Monitoring	Other Transients	Non-safety Systems
Senior Reactor Op.	Time Available	Inspecting	System Isolation	
Reactor Operator	Staffing	Checking	Normal Operation	
Auxiliary Operator	Organizational Climate	Deciding	External Event	
Maintenance Mech.	Job-related Training	Managing	Outage	
I&C Tech.	Information Feedback	Communicating		
Chemistry Tech.	Task Complexity	Calibrating		
Health Physics Tech.	Regulations	Responding		
Engineers	Stress	Maintaining		
Security Guard	Fatigue			
QA/QC Tech.	Attitude			
Contractor Personnel	Job-related Experience			
Operations Org.	Fitness for Duty			
Maintenance Org.	Perceived Risks			
I&C Org.	Procedures			
Chemistry Org.				
Health Physics Org.				
Engineering Org.				
Plant Management				
QA/QC Org.				
Security Org.				
Off-site Response Personnel				

Appendix C

Data Element Tables for Working Level Issues

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
1.1	All	All	All	All	All
1.2	All	Job-related Training	All	All	All
1.3	All	Job-related Training Job-related Experience	All	All	All
1.4	All	Organizational Climate	All	All	All
1.5	All	Organizational Climate	All	All	All
1.6	All	Organizational Climate	All	All	All
1.7	Shift Tech. Advisor	All	Monitoring Inspecting Checking Communicating Responding	All	All
1.8	Plant Manager	All	Monitoring Inspecting Checking Deciding Managing Communicating Responding	All	All
1.9	All	All	All	All Except System Isolation Normal Operation Outage	All
1.10	All	Staffing Stress Fatigue Attitude	All	All	All

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
1.11	All	All	Monitoring Checking	Normal Operation	All
1.14	All	All	All	All	All
1.15	All	Staffing	All	All Except System Isolation Normal Operation Outage	All
1.19	All	Staffing	All	All	All
1.20	Shift Tech. Advisor Engineers Engineering Org.	All	Deciding Managing Communicating Responding	All Except System Isolation Normal Operation Outage	All
1.21	All	Job-related Training	All	All	All
1.22	All	Job-related Experience	All	All	All
1.23	All	Staffing Job-related Training Job-related Experience	Checking Managing Communicating	All	All
1.24	Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. I&C Tech. Security Guard QA/QC Tech. Operations Org. Maintenance Org. QA/QC Org. Security Org.	All	All	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
1.25	All	Organizational Climate Job-related Training Task Complexity Regulations Attitude Job-related Experience	All	All	All
1.26	All	All	All	All	All
1.27	All	Job-related Training Job-related Experience	All	All	All
1.28	Plant Manager Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org.	Time Available Job-related Training Information Feedback Task Complexity Job-related Experience Procedures	Deciding Communicating Responding	All Except System Isolation Normal Operation Outage	All
1.29	QA/QC Tech. QA/QC Org.	Staffing Job-related Training Task Complexity Regulations Job-related Experience	Testing Monitoring Inspecting Checking Deciding Communicating	System Isolation Normal Operation Outage	All
1.30	Off-site Response Personnel	Staffing Job-related Training Job-related Experience	Deciding Managing Communicating Responding	All Except System Isolation Normal Operation Outage	All
2.1	Senior Reactor Op. Reactor Operator Auxiliary Operator	Job-related Training	All Except Maintaining	All	All
2.2	Maintenance Mech. Maintenance Org.	Job-related Training	Inspecting Checking Maintaining	System Isolation Normal Operation Outage	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
2.3	Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech.	Job-related Training	All	All	All
2.4	All Except Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. Operations Org. Maintenance Org.	Job-related Training	All	All	All
2.5	Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	All	Operating Monitoring Checking Deciding Managing Responding	All Except System Isolation Normal Operation Outage	All
2.7	Senior Reactor Op. Reactor Operator	Job-related Training	Operating Monitoring Inspecting Checking Deciding Communicating Responding	All	All
2.8	All Except Senior Reactor Op. Reactor Operator	All	All Except Operating	All	All
2.9	Senior Reactor Op. Reactor Operator Auxiliary Operator	Job-related Training	Operating Monitoring Checking Deciding Communicating Responding	All	All
2.10	All	Job-related Training	All	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
2.12	All	Job-related Training	All	All Except System Isolation Normal Operation	All
2.13	Off-site Response Personnel	Job-related Training	Monitoring Deciding Managing Communicating Responding	All Except System Isolation Normal Operation Outage	All
2.14	All	Job-related Training	All	All	All
2.16	Senior Reactor Op. Reactor Operator Auxiliary Operator	All	Operating Monitoring Inspecting Checking Deciding Communicating Responding	All Except System Isolation Normal Operation Outage	All
2.17	All	Job-related Training	All	All	All
2.18	Senior Reactor Op. Reactor Operator	Equipment Design Time Available Job-related Training Information Feedback Task Complexity Stress Fatigue Perceived Risks Procedures	Operating Monitoring Checking Deciding Communicating Responding	All	All
2.19	All	Job-related Training	All	All	All
2.20	All	Equipment Design Job-related Training Information Feedback Procedures	All	All	All
2.21	All	Job-related Training	All	Normal Operation	All

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
3.2	Senior Reactor Op. Reactor Operator Operations Org.	Time Available	Operating Deciding Responging	All Except System Isolation Normal Operation Outage	All
3.3	Senior Reactor Op. Reactor Operator Operations Org.	Organization Climate	Operating Monitoring Checking Deciding Communicating Responding	All	All
3.4	Senior Reactor Op. Reactor Operator	Job-related Training Job-related Experience	All Except Calibrating Maintaining	All	All
3.5	Senior Reactor Op. Reactor Operator	Job-related Training Job-related Experience Procedures	Deciding Responding	All	All
3.7	Senior Reactor Op. Reactor Operator	All	All Except Calibrating Maintaining	All Except System Isolation Normal Operation Outage	All
3.9	Senior Reactor Op. Reactor Operator	Job-related Trainig Information Feedback Task Complexity Job-related Experience Procedures	All Except Calibrating Maintaining	All	All
3.11	Senior Reactor Op. Reactor Operator	Job-related Training Job-related Experience	All Except Calibrating Maintaining	All	All
3.13	Senior Reactor Op. Reactor Operator	Job-related Training Job-related Experience	All Except Calibrating Maintaining	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
3.14	Senior Reactor Op. Reactor Operator	Time Available Information Feedback Task Complexity Perceived Risks Procedures	All Except Calibrating Maintaining	All	All
3.15	Senior Reactor Op. Reactor Operator	Job-related Training Job-related Experience	All Except Calibrating Maintaining	All	All
4.1	All	Procedures	All	All	All
4.2	All	Equipment Design Workplace Layout Information Feedback Task Complexity Procedures	All	All	All
4.3	Plant Manager Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Plant Management	Workplace Layout Time Available Information Feedback Task Complexity Procedures	All Except Testing Communicating Calibrating Maintaining	All	All
4.5	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. Operations Org. Maintenance Org.	Procedures	All	All	All
4.6	Senior Reactor Op. Reactor Operator Auxiliary Operator	Procedures	All Except Testing Calibrating Maintaining	All Except System Isolation Normal Operation Outage	All
4.8	All	Stress Procedures	All	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
4.9	All	Procedures	All	All	All
4.10	All	Procedures	All	All	All
4.11	All	Organizational Climate Procedures	All	All	All
4.12	All	Job-related Training Procedures	All	All	All
4.14	All	Equipment Design Information Feedback Task Complexity Procedures	All	All	All
4.15	All	Equipment Design Information Feedback Task Complexity Procedures	All	All	All
4.16	Maintenace Mech. Maintenance Org.	Procedures	Inspecting Checking Maintaining	System Isolation Normal Operation Outage	All
4.17	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org. Plant Management	Procedures	All Except Calibrating Maintaining	All	All
4.19	Senior Reactor Op. Reactor Operator Operations Org.	Procedures	Testing Operating Maintaining	System Isolation	Safety-related Systems
4.20	QA/QC Tech. QA/QC Org.	Procedures	Testing Monitoring Inspecting Checking	Normal Operation Outage	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situations	Systems
4.21	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Procedures	Testing	Outage	All
5.1	Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design	All Except Calibrating Maintaining	All	All
5.2	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org.	Equipment Design Workplace Layout Information Feedback Task Complexity Procedures	Responding	All	All
5.3	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org.	Equipment Design	All Except Communicating Calibrating Maintaining	All	All
5.4	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design	All Except Calibrating Maintaining	All	All
5.5	All	Equipment Design Workplace Layout Habitability Time Available Information Feedback Task Complexity Procedures	All	All	All
5.6	All	All	All	All	All
5.7	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Workplace Layout Information Feedback Task Complexity Procedures	All Except Calibrating Maintaining	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa- tions	Systems
5.8	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Workplace Layout Information Feedback Task Complexity Procedures	Monitroing Deciding Responding	All	All
5.9	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	All	All Except Calibrating Maintaining	All Except System Isolation Normal Operation Outage	All
5.10	Shift Supervisor Senio. Reactor Op. Reactor Operator	Equipment Design	Monitoring Deciding Responding	All	All
5.11	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design	All Except Calibrating Maintaining	All	All
5.12	All	Equipment Design	All	All	All
5.13	Maintenance Mech. Maintenance Org.	Equipment Design Information Feedback Task Complexity	Maintaining	System Isolation Normal Operation Outage	All
5.14	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Workplace Layout Habitability Staffing Information Feedback Task Complexity Regulations Perceived Risks Procedures	All Except Calibrating Maintaining	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa- tions	Systems
5.15	Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Workplace Layout Information Feedback Task Complexity Stress Perceived Risks Procedures	All Except Calibrating Maintaining	All	All
5.16	Senior Reactor Op. Reactor Operator Auxiliary Operator	All	Testing Operating Monitoring	All	All
5.17	Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design	Monitoring	All	All
5.18	Auxiliary Operator	All	All Except Calibrating Maintaining	All	All
5.19	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	Equipment Design Habitability Information Feedback Procedures	All Except Calibrating Maintaining	External Event	All
5.20	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	Equipment Design Information Feedback Procedures	All Except Calibrating Maintaining	All	All
5.21	All	All	All	All Except System Isolation Normal Operation Outage	All

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa- tions	Systems
5.22	Plant Manager Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Chemistry Org.	Equipment Design Information Feedback Task Complexity Procedures	Testing Operating Monitoring Inspecting Checking	System Isolation Normal Operation Outage	All
5.23	Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech.	Equipment Design Information Feedback	Monitoring	Normal Operation	Safety-related Systems
6.1	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator	All	Operating	All Except System Isolation Normal Operation Outage	Safety-related Systems
6.2	All	Time Available	All	All	All
6.3	Shift Supervisor Senior Reactor Op. Reactor Operator	Equipment Design Job-related Training Information Feedback Task Complexity Attitude	Deciding	All	All
6.4	All	Job-related Training Information Feedback Task Complexity Procedures	All	All	All
6.6	Shift Supervisor Shift Tech. Advisor Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. I&C Tech. Maintenance Org. I&C Org.	Equipment Design Information Feedback	All	Normal Operation	All
6.7	All	All	All	All	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
6.9	All	All	All	All	All
6.10	Maintenance Mech. Maintenance Org.	All	Maintaining	System Isolation Normal Operation Outage	All
6.14	Plant Management	Information Feedback	Communicating	All	All
7.3	Plant Manager Plant Management	Staffing Organizational Climate Job-related Training Regulations Attitude	Monitoring Deciding Managing Communicating	All	All
7.5	Plant Manager Plant Management	All	Monitoring Deciding Managing Communicating	All	All
7.7	Plant Manager Plant Management	Information Feedback	Managing	All	All
7.9	All	All	All	External Event	All
7.11	All	Attitude	All	All	All
7.12	Chemistry Org. Plant Management Security Org. Off-site Response Personnel	All	Communicating	All Except System Isolation Normal Operation Outage	All
7.13	QA/QC Tech. QA/QC Org.	Information Feedback Regulations Procedures	Monitoring Inspecting Checking Deciding Managing Communicating	Normal Operation Outage	All

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa-tions	Systems
8.1	Plant Manager Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Maintenance Mech. I&C Tech. Health Physics Tech. Security Guard Contractor Personnel Security Org.	Procedures	All	All	All
8.5	All	Staffing Organizational Climate Regulations Attitude Job-related Experience Fitness for Duty	All	All	All
8.6	All	Staffing Organizational Climate Attitude Fitness for Duty Procedures	All	All	All
8.7	All	Organizational Climate	All	All	All
8.9	All	Workplace Layout	All	All	All
8.11	Off-site Response Personnel	Workplace Layout Staffing Attitude Procedures	Communications Responding	All Except System Isolation Normal Operation Outage	All
9.1	Maintenance Mech. Contractor Personnel Maintenance Org. Plant Management	Equipment Design Workplace Layout Habitability Staffing Job-related Training Information Feedback Task Complexity Regulations	Testing Inspecting Checking Managing Maintaining	Normal Operation	Safety-related Systems

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Indivuals & Groups	PSFs	Actions	Situa-tions	Systems
9.6	Shift Supervisor Senior Reactor Op. Reactor Operator Auxiliary Operator Operations Org.	All	Testing Monitoring	System Isolation Normal Operation Outage	All
9.9	Plant Manager Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Maintenance Org.	All	Testing Monitoring Maintaining	System Isolation	Safety-related Systems
9.10	Maintenance Org.	Staffing	Maintaining	Normal Operation Outage	All
9.11	Maintenance Org. Plant Management	Staffing Organizational Climate Attitude Perceived Risks Procedures	Maintaining	Normal Operation Outage	All
9.12	Maintenance Org. Plant Management	All	Maintaining	Normal Operation Outage	All
9.13	Plant Manager Shift Supervisor Operations Org. Maintenance Org. Plant Management	Information Feedback Procedures	Maintaining	System Isolation	Safety-related Systems
9.14	Maintenance Mech. Maintenance Org.	Staffing Job-related Training Job-related Experience	Maintaining	Normal Operation	All
9.15	Maintenance Mech. Maintenance Org. Security Org.	Regulations	Maintaining	Normal Operation Outage	Non-safety Systems

Data Element Tables (Continued)

1	2	3	4	5	6
	Data Elements				
Working Level Issue Identifiers	Individuals & Groups	PSFs	Actions	Situa- tions	Systems
9.16	Shift Supervisor Senior Reactor Op. Reactor Operator Operations Org. Maintenance Org. Plant Management QA/QC Org.	All	Testing Calibrating Maintaining	System Isolation Normal Operation	Safety-related Systems

APPENDIX D

APPLICATION OF THE DATA ELEMENT AND RECORD METHOD

1.0 EXAMPLES OF TYPES A, B, AND C ISSUES

1.1 Type A Working Level Issue

An example of a Type A issue is:

"How does training affect operator performance during an external event?"

The appropriate elements in each category for this issue are presented in Table D.1.

Table D.1 Data Element Table for the Type A Example

Personnel	Actions	PSFs	Situation	System
Senior Reactor Operator Reactor Operator* Auxiliary Operator	Operating* Monitoring Checking Deciding Communicating	Training*	External Event*	Safety-related Systems*

Using all permutations of these elements results in 15 issue data records (3 personnel x 5 actions x 1 PSF x 1 situation x 1 system = 15 data records).

Using the particular elements which are identified by an asterisk in Table D.1, one issue data record is:

How does training affect operating by a Reactor Operator on a safety-related system during an external event?

Another issue data record is:

How does training affect checking by auxiliary operators on a safety system during an external event?

If data were available to completely and competently addressed all 15 data records, then a complete technical basis for addressing this working level issue would exist. All Type A working level issues could potentially be addressed by data typically used in HRA/PRA calculations.

1.2 Type B Working Level Issues

An example of a Type B issue is:

"What is the optimal role of maintenance in preventing the aging of components?"

A set of issue data records can be generated for this issue, but they will not form a complete technical basis for addressing it. The relevant data elements for each category are listed in Table D.2.

Table D.2 Data Element Table for Type B Example

Personnel	Actions	PSFs	Situation	System
Maintenance Organization*	Maintaining*	Staffing	Normal Operation*	Safety-related Systems*
Plant Management	Testing	Information Feedback		
Contractor Personnel	Inspecting	Task Complexity		
	Checking	Training		
	Managing	Procedures*		
		Regulations		
		Habitability		
		Equipment Design		
		Workplace Layout		

All permutations of these elements generate a total of 135 issue data records (3 personnel x 5 actions x 9 PSFs x 1 situation x 1 system). An example of one issue data record can be made by considering the elements designated with an asterisk in Table D.2:

How do procedures affect maintaining by the maintenance organization on safety-related systems during normal operation?

Of particular importance, however, is the aspect of "aging" in this working level issue. In addition to the data represented by the data records, information on the impact of equipment aging on plant reliability is must be considered. The elements which are available to form data records are not sufficiently detailed to reveal a need for information on equipment aging. However, upon completing the data element, table the need for this specific type of information can be determined by the analyst. This information may be acquired by examining the products of NRC's extensive equipment aging studies presently underway.As a result, for Type B issues will be important to seek additional information or data beyond that represent by the data records including: external standards, literature from other industries, more specific PSFs or Actions than are included in Table 3.1, task analyses, field data, or some form of qualitative data. Some of this information may be available in the PRA or may have to be acquired elsewhere.

1.3 Type C Working Level Issue

An example of a Type C issue is:

"What is the relationship between operational history and safety in plants?

This issue does not lend itself to the form of issue data records and instead requires information which can be collected only from field data on operational history and safety.

Table 2 of Appendix B has separate lists of the Type A, B, or C working level issues. This approach allows for identification of pertinent data for each issue besides that which may be typically used in performing risk assessments.

2.0 APPLICATION OF THE DATA ELEMENT AND RECORD METHOD

The data element method can be applied to any issue or set of issues. It will generate a set of data records which would address the issue or issues and facilitate identification of other types of information or data which may be required. This section describes two applications of the method. In the first, a Generic Safety Issue from which initial human performance regulatory issues were derived and then refined and clarified into final working level issues is used. The final working level issues are then used to develop a data element table which can generate a set of issue data records and used to identify additional types of information and data required to address the working level issues on an agency-wide basis. In the second, a Generic Safety Issue is used directly to generate a data element table for the human performance issues related specifically and only to that Generic Safety Issue.

2.1 Application to Final Working Level Issues Associated with a Generic Safety Issue

This section illustrates how the data element method of identifying needed data can be applied to a set of refining and clarified working level issues initially derived from a Generic Safety Issue. As an example, the TMI Action Plan item II.E, "System Design" is considered, the relevant issues on the final working level issues list identified, and a data element table to generate appropriate issue data records presented. Additional forms of information needed are also identified. This approach is illustrated in Figure D.1.

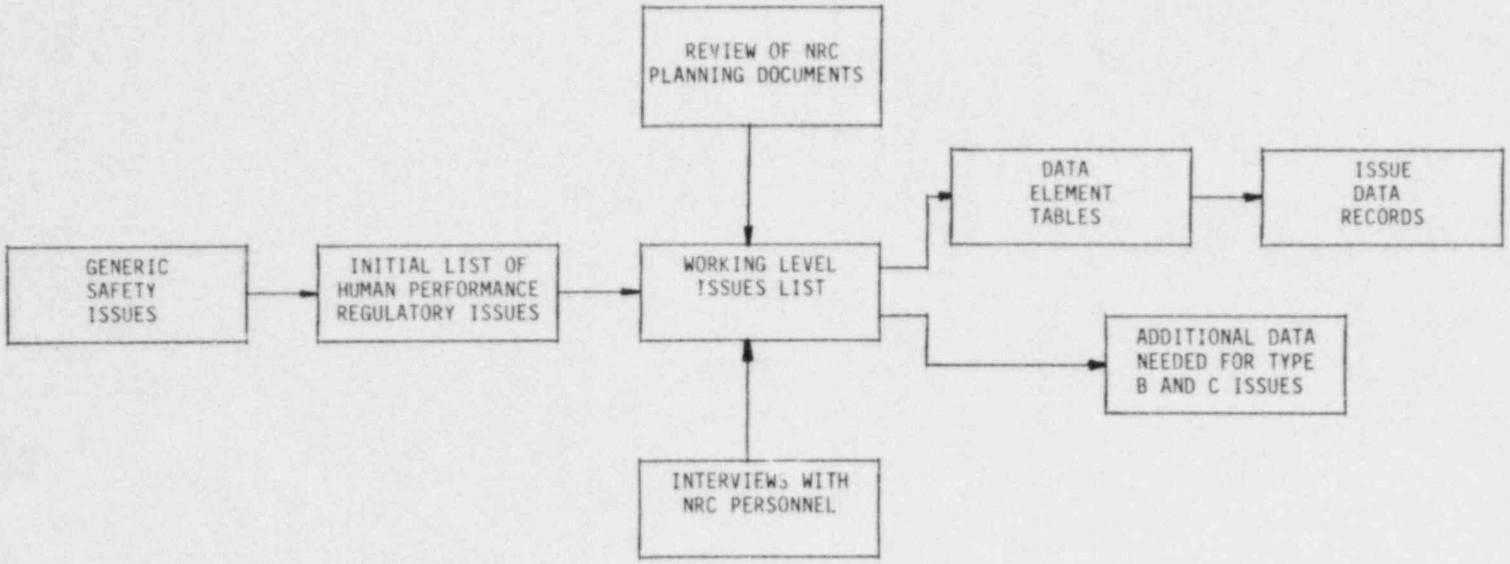

Figure D.1 Method used to identify and list human performance regulatory issues and data needed to address them.

The "System Design" item is a Generic Safety Issue that involves several safety-related systems, structures, and related procedures. These include the Auxiliary Feedwater System, the Emergency Core Cooling System, Decay Heat Removal System, Containment Design, Design Sensitivity of B&W Reactors, and In Situ Testing of Valves. Of particular interest are means to increase the reliability of these systems, improving operator performance using these systems during off-normal situations, development of advanced displays and information handling devices, improved testing and surveillance of systems and structures, and improved maintenance procedures.

The following working level issues were initially deduced from this TMI Action item and subsequantly refined and classified using NRC planning documents and interviews with relevant NRC personnel. They are identified according to their number in the final working level issues list (Appendix B) and the issue type.

(2.1) What is the optimal content for an operator training curriculum? (B)

(3.1) What are the important accident sequences for each plant? (C)

(3.4) What are the most important skills, knowledge, and abilities for operators? (B)

(3.14) What are the appropriate engineering models for use in programming simulators? (A)

(4.2) How should trade-offs between hardware changes and procedural changes be assessed? (A)

(4.3) What is the best use of artificial intelligence in the control room? (B)

(4.15) What types of procedures should be used for advanced display systems? (A)

(4.16) What types of procedures are optimal for preventive and corrective maintenance? (A)

(4.19) What are the optimal procedures to be used to minimize risk in isolating systems? (A)

(5.8) What are the optimal means of managing information in the control room? (A)

(9.6) What are the optimal means of conducting test and surveillance activities in terms of risk? (A)

(9.16) How can the availability of safety-related systems be improved? (B)

A composite data element table (Table D.3) is constructed for all these working level issues. The elements included in this Table reflect the specific personnel, actions, PSFs, situations, and systems pertinent to the working level issues listed above.

This data element table will generate 2700 issue data records which will address item II.E (5 personnel x 6 actions x 9 PSFs x 5 situations x 2 systems = 2700 issue data records.) Some of these issue data records will be unrealistic because some personnel do not perform all of the actions listed. By considering each personnel position separately, a more realistic set of data records for this item can be generated. For instance, plant management does

Table D.3 Data Element for Working Level Issues.

Personnel	Actions	PSFs	Situation	System
Plant Management	Testing	Equipment Design	Normal	Safety-related Systems
Senior Reactor Operator	Operating	Time Available	System Isolation	Plant Structures
Reactor Operator	Monitoring	Training	LOCA	
Auxiliary Operator	Inspecting	Information Feedback	Transient	
Maintenance Organization	Checking	Task Complexity	LOSP	
	Maintaining	Stress		
		Experience		
		Procedures		
		Perceived Risk		

not maintain equipment so that when considering only plant management, the action element "maintaining" can be dropped from consideration. If only realistic permutations are considered, the total number of data records is reduced to 1890.

In addition to the information needed as indicated by these data records, other information is needed. For each Type B and C working level issue, the following is needed:

- For Working Level Issues 2.1 and 3.1 additional information on instructional technology and most important sequences involving safety-related systems is needed.

- For Working Level Issue 3.4 additional information including task analyses on operators and critical errors during important sequences is needed.

- For Working Level Issue 4.3 additional information on the types of artificial intellegence systems available and there applicability to control room operations is needed.

- For Working Level Issue 9.16 additional information on how particular maintenance, calibration, and testing tasks can be made more effective is needed.

2.2 Application Directly to a Generic Safety Issue

The working level issues represent agency-wide human performance regulatory concerns. The data element method can also be applied directly to an individual Generic Safety Issue. While this application does not taken into account important agency-wide concerns, it can be useful in identifying specific data records needed to address a particular Generic Safety Issue.

As an example, the Generic Safety Issue, TMI Action Item I.D.5," Improved Control Room Instrumentation Research" can be used to generate a set of issue data records. This Generic Safety Issue concerns how information or reactor status is presented to operators in the control room. Lights, alarms, annuciators, and other displays are of specific importance in terms of how well they apprise the operators of plant conditions during normal and off-normal situations. Operator actions in question include diagnosis and response as well as routine operation. Table D.4 represents the data element table developed from a close review of this Generic Safety Issue.

Table D.4 Data Element Table Generated Directly from TMI Action Item I.D.5.

Personnel	Actions	PSFs	Situations	Systems
Shift Supervisor Senior Rx. Op. Rx. Op.	Operating Monitoring Checking Deciding Responding	Equipment Design Workplace Layout Info. Feedback Task Complexity Stress Procedures	LOCA Loss of off-site Power Other Transients System Isol. Normal Operation External Event Outage	Safety-related Systems

This data element table will generate a total of 630 issue data records directly pertinent to this Generic Safety Issue (3 personnel x 5 actions x 6 PSFs x 7 situationa dns 1 system = 630 issue data records). The right hand column of Table 1 in Appendix D represents the data element table derived directly from each Generic Safety Issue.

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This report is the first in a series which documents research aimed at improving the usefulness of Probabilistic Risk Assessment (PRA) results in addressing human risk issues. This first report describes the results of an assessment of how well currently available PRA data addresses human risk issues of current concern to NRC.

Findings indicate that PRA data could be far more useful in addressing human risk issues with modification of the development process and documentation structure of PRAs. In addition, information from non-PRA sources could be integrated with PRA data to address many other issues.

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