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SAFETY RELATED OPERATOR ACTION

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ABSTRACT

The history and content of the N660 criteria and the need for data to guide the time criteria are examined. A program for collecting such data is outlined. This program includes field data collection, field calibration of simulator experiments and a simulator testing program. The results of an initial study of the availability of and techniques for collecting field data are presented. The data do exist and can be collected by a concentrated effort including NRC docket searching and site visits. Results of operator surveys concerning event stress levels and diagnosis difficulty are presented. Statistical analyses of data on time to reset inadvertent safety injection are presented. Conclusions and recommendations for future work based on this initial study are included.

I. INTRODUCTION

There is increasing recognition on the part of reactor safety analysts of the need to include in system reliability and safety studies the effects of human interaction. The desire is to quantify the impact of the operator on system performance. The major obstacle to further quantification is the lack of a comprehensive, objective data base. Currently used data on human performance in nuclear power plant operations are based primarily on information available from studies of personnel in tasks other than nuclear-power-related operation (e.g., aviation or military operations) or from subjective observation (i.e., expert opinion) from nuclear industry personnel. Data that do exist almost exclusively are related to operation under routine conditions. There is virtually no objective data base for nuclear-power-plant-operator behavior under severe accident conditions, i.e., for safety-related operator action.

The reasons for the lack of data seem to be threefold:

(1) The exemplary safety record of the nuclear power industry has produced so few incidents of major consequence that it is impossible (and presumably will remain impossible) to construct a statistical data base totally from actual experience.

(2) Most of the safety-system actions required for response to potentially serious accidents are automated. Consequently the primary focus of designers and safety analysts has been the quantification and improvement of safety-system-equipment reliability and availability.

(3) Measurement (quantification) of human response, particularly response under the stress of severe accident conditions, is extremely difficult. Human variability and the very aspects of human behavior that make the operator a desirable element for response to extreme events - adaptability, learning capability, capacity for performing multipurpose tasks, ability to self-monitor and correct errors - make systematic analysis and prediction of response much more difficult than for machines.

In most technical areas development of codes, standards and regulations usually lags the development of technology. In this particular area, safety-related operator action in nuclear power plants, the opposite appears to be true. There is a well-developed effort that has been in progress for a number of years to establish industry design standards for safety-related operator action. Currently there is still very little research (at least non-proprietary studies) specifically for or directly applicable to the purpose of assessing operator response to severe accident events. The standards effort referred to is the American Nuclear Society (ANS) Writing Group 58.8 (formerly 51.4) work toward development of an American National Standards Institute (ANSI) standard entitled "Criteria for Safety-Related Operator Actions," and designated ANSI N660.[1] The criteria recognize the lack of an adequate data base, but also recognize the need to set guidelines for designers and regulators to determine when certain required safety actions can be initiated by operator action, as opposed to automated action. The writing group has developed interim criteria based on operator "response time." The first draft of the proposed criteria was released in 1973, and work has continued since then to review and improve the criteria and, more importantly, to focus research toward developing the necessary data base.

The need for data to support development of the N660 criteria provided the primary impetus for undertaking this study. The USNRC Office of Nuclear Reactor Regulation, which has a representative on the N660 writing group, requested research support from the Reactor Safety Research Division (RSR) of the Office of Nuclear Regulatory Research. RSR then requested ORNL to examine the problem and suggest an overall approach to accumulating the necessary data, especially data that includes actual operating experience.

It was generally agreed that only a comprehensive, long-range human factors study which examined both the human and the machine aspects of the "man-machine interface" would yield the ultimately desired solution of capitalizing on the advantages of both human and automated action to optimize reliability and availability of safety systems. With regard to interim criteria, the primary conclusions from the initial examination were that:

(1) because of the relatively few events that have been experienced the data base on operator response to design basis events could not be based exclusively on operating experience, but would have to rely heavily on simulator experience as well, (2) the major problem with using simulator results is verification of their applicability to "real" reactor accident conditions and (3) that there may well exist enough data from operating experience to provide a "calibration" of simulator results, i.e., some way of correlating simulator data to actual operating data. ORNL proposed a program that started with a preliminary assessment of the availability of applicable historical (field) data and development of procedures to collect field data. If the results of the preliminary assessment were positive, the program would then proceed in three phases:

(1) Collection of field data.

(2) Development of a correlation between simulator experiments and field data.

(3) Development of a data base using correlated simulator experiments.

This report summarizes the results and conclusions of the preliminary assessment, a six-month (approximately one-half person-year) study. The primary goals of the study were to:

(1) Independently review the proposed N660 criteria and pertinent background material.

(2) Assess the availability of applicable field data.

(3) Devise systematic procedures to collect field data.

(4) Assess preliminary data collected.

(5) If apparent availability of data warranted, outline a program for collection of data and development of the desired data base.

The study concentrated on collecting data on applicable events which have occurred at five different operating nuclear power plant sites (ten units) at which management had volunteered cooperation. All available NRC docket information and site records pertaining to the events were examined, and a written operator opinion survey form was developed and administered. Following the study at the five plants, NRC docket files were searched for information on any occurrences of a few selected events at all operating U.S. PWRs and BWRs. In addition, a broad survey of the psychological and sociological literature was made by consultants with experience in industrial psychology for possible information from studies performed on non-nuclear personnel which might include applicable data.

This report discusses the study of the N660 criteria, the approach taken in collecting field data, the results of the data collection, and conclusions and recommendations for further work.

II. REVIEW OF N660 CRITERIA

The background and contents of ANSI N660 criteria currently proposed were reviewed in order to gain an understanding of the specific needs for data and the general approach taken in developing the proposed standard for safety-related operator action. Table 1 highlights some of the history of the development of the current draft standards. Since the potential impact on future designs, and possibly on existing designs if "backfitting" were to be required, could be considerable, there has been a good deal of controversy surrounding this standard.

The proposed criteria are an extension of an approach which has gained some acceptance through use in design and regulatory processes which simply designates a certain time margin immediately after initiation of a design basis event during which there may be no reliance upon the operator to complete required safety system actions. That is, there are no "required operator actions" within a specified time after initiation of a design basis event. Since the time margin used as a guideline has often been ten minutes, this approach has been referred to as the "ten-minute rule," though there is no written statement or basis for the approach, and frequently other time margins have been used. In fact, the lack of any consistent basis for assessing designs and establishing time margins is a major reason for development of guidelines and criteria.

It is important to understand what is meant by *required operator actions*. As defined in the draft N660 criteria, these are actions which are part of the plant design basis and are used to initiate or adjust safety system equipment. Specifically they are "actions which require manual manipulation of equipment during the course of design basis events (those examined in Chapter 15 of the Safety Analysis Report) to enable the safety systems to provide the minimum acceptable performance that will prevent violation of the design requirements for the particular event category."[1] The criteria also define *optional* and *unplanned* operator actions, but do not

TABLE 1. Highlights of History of N660 Development

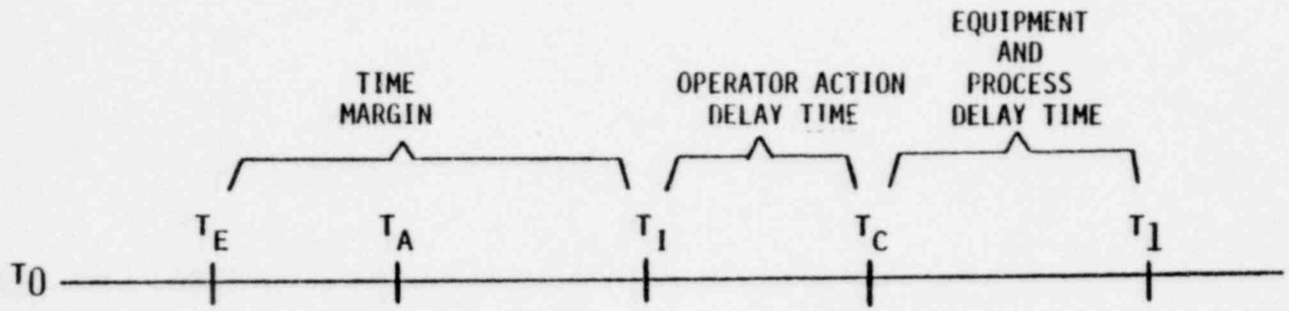
Date	Action
1973	Effort initiated by ANS-51; first draft released based on "10-minute rule"; scope was "all accidents".
4/26/74	AEC cast "negative with comment" ballot on first draft; favored more automation; suggested very long time margins, e.g., one hour.
Late 1974-Early 1975	Draft rewritten, primarily by chairman of writing group; form similar to current draft; several revisions before approval by ANS-51 PWR committee.
2/75	ANS-50 ballot on draft number 4; affirmative vote, but many negative comments.
1975-76	Several revisions of draft to accommodate comments.
11/76	Draft released for trial use and comment (TUC).
6/76	Writing group reorganized with representatives from utilities, vendors, architect engineers and NRC.
Currently	Trying to resolve issues raised during TUC period and subsequent reviews, plus incorporate preliminary results of research at EPRI, Westinghouse and ORNL.

attempt to specify criteria for those.* Optional operator actions are those "...Not required following a design basis event but may be performed by the operator to improve safety system(s) performance over the acceptable minimum."[1] Unplanned operator actions are those which "...May be necessary or useful as corrective measures after an unforeseen event for which the safety systems do not provide the minimum acceptable performance and the event may exceed the design requirements."[1] Note that to make the problem tractable, negative optional or unplanned operator actions are not considered explicitly. That is, the criteria specify time margins for <u>correct</u> completion of a required operator action without attempting to address possible "irreversible" consequences of incorrect operator actions.

The time intervals addressed by the criteria are illustrated in Fig. 1. Given that an event occurs for which there is a required operator action at time t_o and that an event alarm annunciates its occurrence at time t_e, a time margin t_i-t_e is specified as the minimum time which must be allowed for the operator to <u>initiate</u> the required action. (An additional alarm specifically directing the operator to take action may or may not occur at t_a.) The "operator action delay time" is the time necessary for the operator to complete the required action (which may, in fact, consist of a number of discreet manual actions such as flipping a switch, adjusting a dial, etc.). Finally, there must be an allowance for "equipment and process time delay." For example, starting a standby pump to provide cooling water to a heat exchanger may be the appropriate corrective action completed at time t_c, but there is a finite delay time before cooling water supplied by that pump can be effective in restoring the system temperature of interest to the level necessary to avoid exceeding a design limit.

The designer would use the time tests prescribed by the criteria to determine whether operator initiation of the required action is acceptable. Each event for which it is proposed to rely upon a required operator action would be examined to determine the time t_l by which the protective function

*An appendix to the criteria do suggest guidelines for considering optional and unplanned actions in system design.



T_0 = EVENT INITIATION

T_E = EVENT ALARM

T_A = OPERATOR ACTION ALARM

T_I = INITIATE OPERATOR ACTION

T_C = COMPLETE PROTECTIVE ACTION

T_1 = COMPLETE PROTECTIVE FUNCTION: REACH DESIGN REQUIREMENT LIMIT

Fig. 1. Time Intervals Addressed by the Draft N660 Criteria.

must be completed. The equipment and process time delay would be subtracted from that time, and then certain requirements (designated "time test 2") for the operator action delay time would be applied to determine the time t_i by which operator action would have to be initiated. The value of t_i must be greater than minimum time specified by additional requirements in the criteria ("time test 1"), or else the designer could not use operator action to initiate that particular protective action. He would have to either automate initiation of that action or alter the system in some other way to reduce the different time intervals such that time test 1 would be satisfied.

Much of the effort and the controversy associated with the criteria development, and most of the work in this study, have centered on the time margin t_i-t_e, in time test 1. The time margin is seen as necessary to permit the operator to "(1) recover from his initial stress, (2) diagnose the event that has occurred, and (3) plan his action."[1] In addition, the time margin allows the operator to "(1) assure that proper automatic protective actions have occurred, (2) initiate manual backups to automatic protective actions, and (3) monitor the correct accomplishment of automatic protective functions."[1]

The approach has been to specify a time margin which represents an appropriately conservative design value and to increase the time margin as (1) the severity of the event increases, (2) the frequency of occurrence decreases, and (3) the familiarity of the operator with the event decreases. In practice, there are three values specified, depending on whether the event is a Condition II, III, or IV event as defined by ANSI Standard N18.2 for PWRs and N212/ANS-52.1 for BWRs. (Condition IV events are the most severe, least frequent design basis events such as LOCA, and Conditions III and II are designated as less severe, more frequent.) In the draft standard released for trial-use-and-comment in 1976, the time margins specified were 10, 20 and 30 minutes, respectively for Conditions II, III, and IV events. However, the approach now being suggested by the ANS 58.8 writing group is to use the existing framework for specifying time criteria, but to specify the values only after there is some basis in data resulting from ongoing and/or planned research projects.

In developing the criteria and releasing the draft version for trial use and comment, the writing group recognized that the use of this time margin as a criteria, and especially specifying values for the time margins could not be justified by available quantitative data. It was felt, however, that the strong need to "provide a consistent basis for design," and to stabilize the licensing process "by reducing the case-by-case resolution of design assumptions" demanded development of interim criteria on a high priority basis. A number of specific research needs are listed in the draft criteria. At least two different studies, both using simulator experiments, are currently attempting to gather data in support of the criteria development. This study at ORNL is different in that it emphasizes the use of data from actual operating experience.

III. RESULTS OF THE PRELIMINARY ASSESSMENT

The preliminary assessment was a six-month study that was divided into four subtasks:

(1) Background study

(2) Examination of documented events

(3) Initial site visits

(4) Presentation of results.

The goals of the study were to review the proposed N660 criteria and the broader scope of the problem of defining criteria for safety-related operator action, assess the availability of applicable data from operational experience, devise procedures for collection of applicable data, investigate the possibility of gathering applicable data from non-nuclear sources, summarize initial data collected during the preliminary assessment, and (if results of the study were positive) outline a program for continued development of a data base which includes data from operating experience. The previous two sections of the report provided a summary of the N660 criteria and the background of the overall problem. This section summarizes the approaches used to assess data availability and gather data, and presents initial data collected.

III.A. Selection of Events, Operator Action and Cooperative Sites

After reviewing the draft of the N660 criteria it was necessary to identify (1) the design basis events in Chapter 15 of the SAR that currently include required operator action, (2) the specific actions required, and (3) nuclear power plants that would cooperate. Since there were utility representatives on the N660 writing group, item (3) was probably not as difficult for this stage as it will be for future work, or for this type of study in general. However, even though these utilities had already shown a strong interest and cooperative spirit by participating in the N660 effort, there was an understandable reluctance on the part of some of the utility management to contribute the considerable time and energy of their staff and operators necessary to complete the on-site portion of this study. The number of studies such as this one requesting site visits is apparently increasing, and even administrative time to arrange visits is not insignificant. Also, the current "political" environment that gives rise to increasing public criticism of the nuclear power industry does not encourage utilities and operators to invite public discussion of past accident events. These comments on the difficulty of securing site management cooperation are made neither as a criticism nor a defense of utility management. They are intended to point out that obtaining cooperation, which we feel is vital to any project of this type, is not a trivial portion of the effort, and that there are good reasons for the reluctance of some utilities to participate. The comments should also indicate further that we sincerely appreciate the cooperation we have received during the preliminary assessment.

Items (1) and (2) above, identification of the events and operator action of interest for data collection, also were not straightforward. Examination of "standard" PSARs and (presumably) typical SARs for current-design BWRs and PWRs revealed that very few of the design basis events include required operator action as strictly defined in the N660 criteria. That is, operator action was not used as part of the design basis for licensing. Since mechanistic analysis of many of the events in Chapter 15 of an SAR are carried out to only a very limited time into the accident sequence, it was not clear that required operator action might not exist at some time later in the sequence. Furthermore, without being familiar with the details of the accident analysis it is not easy to determine whether some of the specified operator actions really are required to avoid violation of a design basis. The statements in the SARs, particularly for PWRs, indicate very few required operator actions exist. Examination of several emergency/abnormal operating procedures for both BWRs and PWRs did not help much to clarify exactly which of the actions specified by procedures or SARs were "required" as defined by the N660 criteria. The procedures for a particular event of course vary from plant to plant because of differences in design and operational factors. Also, abnormal operating procedures may include actions that are safety-related but are primarily included to prevent equipment damage.

Those few events and actions which were identified as directly applicable to the N660 criteria are listed in Table 2. In order to increase the potential data base, it was decided to include all design events which had specific manual operator action prescribed by the SAR or operating procedures. Since the specific actions were in a general sense "required" in that the operators training, instructions and written procedures prescribed that they be performed, and since they were performed under stress of a design basis event, data on the performance of those actions are considered applicable to N660. A list of the Chapter 15 events selected for compilation of data during this preliminary assessment are listed in Table 3. Future work should include a more thorough examination of specific operator actions that can be measured and can be assumed to be applicable for the data base. Assistance from persons thoroughly familiar with analysis of Chapter 15 events and persons with operating experience should be obtained.

III.B. Docket Searches for Events at Five Selected Sites

Once the events and actions of interest and the cooperative sites had been identified, the next step was to identify occurrences of the events at the sites. Since the five sites were selected on the basis of availability rather than frequency of occurrence of events, the number of event occurrences may not be typical of that which can be expected at other sites if the data collection continues.

The principal source of information for identifying event occurrences was the library of Safety-Related Occurrences maintained by the Nuclear

TABLE 2. Events Considered Directly Applicable to N660

EVENT	ACTION	TIME REQUIREMENTS
	PWR	
S/G Tube Rupture	Initiate RCS Cooldown and Depressurization	9 Minutes after Low Pressure Trip
Boron Dilution	Terminate Manually	~ 45 Minutes
Loss of A/C Power	Place DHRS in Service (May Be Required after Diesels Come On Line)	None Specified
Loss of Service Water	Reduce Power and Remove RCP from Service	At Least 10 Minutes
	BWR	
Relief Valve Opens Inadvertently and Will Not Reclose	Try to Reclose Valve; If Not Successful, Shutdown the Reactor, Initiate Torus Cooling	ASAP
Rupture of Primary Instrument Line	Shutdown and Isolate Leak	$\leq$ 10 Minutes
Rupture of Off-Gas System	Clear Area of Personnel, Isolate Affected System	Initiate by 1 Minute
Loss of A/C Power	Maintain Pressure and Water Levels by Manual Operation of Relief Valves and RCIC	As Required

TABLE 3. Chapter-15 Events Examined During Preliminary Assessment

BWR INCIDENTS	PWR INCIDENTS
LOAD REJECTION	BORON DILUTION ACCIDENT
TURBINE TRIP	LOSS OF FEEDWATER
MSIV CLOSURE	LOSS OF ALL A.C. POWER
RECIRCULATION PUMP TRIP	FIRE IN THE PLANT
LOCA	LOCA
LOSS OF FEEDWATER	MAIN STEAM LINE RUPTURE
INSTRUMENT LINE RUPTURE	S/G TUBE RUPTURE
FAILURE OF MAIN CONDENSOR OFF-GAS SYSTEM	LOSS OF CONTROL ROOM
PLANT FIRE	OVER PRESSURIZATION
LOSS OF CONTROL ROOM	INADVERTENT SAFETY INJECTION
RELIEF VALVE STUCK OPEN	
LOSS OF A.C. POWER	

Safety Information Center (NSIC) at ORNL. This library includes a variety of docket material - Licensee Event Reports (LERs), special reports, correspondence between the utility and NRC, etc. These items are abstracted and keyworded to allow automated searching of the collection. An example of this type of search using "safety injection" as a keyword is shown in Appendix A.

When a potentially applicable event was identified from the abstract, the report number, docket number and report date were noted. With this information the Power Reactor Docket Information indices at NSIC were used to locate the actual microfiche copy of the docket. This was usually a trial and error process with an estimated 80-90% success rate. There is no consistent method for absolutely identifying or automatically retrieving the correct microfiche directly from the NSIC computerized searches. This is because the complete document is filed according to a single number which is usually assigned by NRC after the abstract is prepared. Once the full microfiche docket entry was available, the applicability of the event to the N660 data base was assessed.

Sometimes the docket entry did not contain any usable data, but did point to other potentially more useful data (e.g., a similar or related event or a different source of information). Several searches through the NSIC files were necessary to exhaust all possible data sources on any given event. After the NSIC search, however, sufficient information was usually available to assess the applicability of the N660 criteria to the event, pinpoint the date and time of occurrence, and generally describe the sequence of events. A small amount of usable time response data was also available in the docket material. An example of a docket entry is shown in Appendix A.

III.C. Data Collection from Site Records

The work conducted at the sites consisted primarily of searching site records and conducting a written survey of operator opinion. The latter is discussed in Section III.F. An attempt was made to examine all available site records which were applicable to the occurrences identified from the NSIC docket searches or which might give information on other occurrences that had not shown up in the NSIC search (e.g., recently filed LERs that had not yet appeared in the NSIC abstracts). The purpose of the records search was to find as much quantitative information as possible about the time required for correct completion of specific operator actions. Additional qualitative information on operator response (error or corrective action) to emergency events inevitably presented itself, and probably more could be extracted; but that was not the primary purpose of this initial work.

The records storage and retrieval facilities at the sites visited (presumably typical) were generally quite effective, i.e., a high percentage of the desired documents were retrieved. All of the sites visited used microfilm systems, automated to varying degrees. These expedite the search procedure but present the normal problem of occasionally illegible copies, as well as the difficulties of transcribing data. As much as possible, hard-copies of key documents were made for analysis later at ORNL. The assistance of site records staff was necessary for instruction on the particular procedures used at that site, but after a period of adjustment, we were able to perform most of the work ourselves with only occasional assistance. The use of records personnel must also be considered in carrying out further records searching. The fact that this effort is tedious and time-consuming also cannot be overlooked.

Table 4 summarizes the main types of records examined at the sites visited which contained some data or have some potential for future work. Each site has a long list of types of records covering a wide range of information. At each site, the catalog of record types was examined and both engineering staff and records staff were queried as to types of

TABLE 4. Summary of Site Document Search

SITE	NO. OF EVENTS	LOGS CHECKED	NO. OF OPERATOR SURVEYS	SPECIAL REPORTS AVAILABLE
~~1~~ A	25	Reactor Operator, Shift Supervisor, Control Room, Computer Printout	21	---
~~2~~ B	8	Reactor Operator, Computer Printout	9	Plant Information Report
~~3~~ C	11	Reactor Operator, Shift Supervisor, Computer Printout	4	Plant Upset Report
~~4~~ D	9	Reactor Operator, Shift Engineer, Control Room	6	Deviation Report
~~5~~ E	8	Reactor Operator, Shift Engineer, Control Room	4	Deviation Report

records that might contain operat~~or~~ response-time data. The broad categories listed in Table 4 are the only ones identified with significant potential for data. Examples of the different types of plant documents are presented in Appendix A.

The first category included hand-written logs. These logs could usually be classified as "reactor operation," "supervisory," or "auxiliary operator" logs. The first two logs often contained some detailed time information, while the latter was more often a radwaste log with little or no information pertinent to operator action. The time data in these logs ranged from very complete to nonexistent. Even in the best of circumstances the data extracted must be recognized as approximate, however, since the times were usually not written down in the log itself until sometime after the event was "under control," and there certainly must have been some loss of accuracy. Finally, it should be noted that usually only major actions or general tasks, which might consist of several subtasks, are noted. For example, placing the Residual Heat Removal (RHR) system on torus cooling may involve several distinct manual actions, but it would typically be written as "placed RHR on torus cooling."

The second category of records includes computer trend, event and alarm printouts. Examination of typical computer output revealed that operator action is seldom explicitly identified. Rather, it is inferred from plant parameter trends or alarm annunciation. Reconstruction of the event chronology from computer output requires someone who is intimately familiar with the plant and its operation. Thus it is not generally an effective means of data collection for an outside party. However, plant personnel have in the past been successful in extracting useful information from these computer output, and there may be specific instances in which they could be used in further work in this study, especially when no other data are available. For instance, in the case of a reactor relief valve stuck open in a BWR plant, the operator is required to manually scram the reactor when the torus water temperature reaches 110°F (43.3°C). The time for the operator to take this action might be determined by noting when the torus temperature reaches 110°F on the computer trend chart, and noting the time of scram on the alarm chart.

A third category of on-site records which contained some time response data were special reports and internal documents written by and for the site staff. These frequently contained rather detailed analyses and chronologies of the events. These reports were compiled by on-site engineers who had access to log books and computer printouts and used personal interviews with the operators who had experienced the event to supplement written records. These records, when they exist, are some of the best sources available for time-response data on the reported event. Unfortunately, such reports do not exist for many events of interest.

III.D. Combining Data from Site Records and NRC Docket Files

In the process of data collection it became clear that there is no single source of data that can be consistently used to collect information on events and operator action. Information has to be pieced together from all available sources. Even then there will usually be some judgement and approximation involved in extracting data, and rarely will information be complete for any one event. The best that can be hoped for is that by careful compilation and pooling of all sources of information about an event occurrence, a reasonably accurate, detailed, complete and documentable chronology can be established from which operator response times to initiate or complete specific actions can be derived.

An example of how information from a number of sources was combined is shown in Fig. 2 which is a copy of a raw data sheet for the event "inadvertent relief-valve lifting." The "required" (not necessarily as defined by N660) operator actions and the sequence of their occurrence was determined by review of BWR PSARs and plant adnormal/emergency operating procedures. There were frequently inconsistencies in reported times among the various sources. In this example, the plant upset report indicates the relief valve lifted at 2200 hours while the supervisor log places the time

SITE

EVENT Relief Valve Opens

ACTION	TIME	REACTOR LOG	SUPERVISOR LOG	PLANT UPSET REPORT	DOCKET
1. Recognize Symptoms verify relief valve open	8 min.	"E" ADS Open	2208 "E" opened Lost 70 MWe	2200 "E" failed open. Lost 70 MWe. Hi Temp Alarm	"E" Relief Valve opened
2. Place Torus cooling in service (RHR pumps)	22 min	—	2222 RHR & HPSW on Torus Cooling	Torus cooling placed in service	—
3. Reduce Rx pressure and attempt to close relief valve			Reduced pressure to reseat valve		—
4. If relief valve will not close or Torus water temp. reaches 110°F, Shutdown		—	—	when relief valve confirmed open 2219 Tripped Recirc pumps	
a) Recirc. Runback (< 1 min)					Runback
b) Trip pumps	1.9 min				Tripped
c) Scram Rx by placing mode sw in S/D	20 min	2220 Manual Scram per OT-35	2220 Manual scram per OT-35	2220 Manual scram (12 sec after recirc. trip)	t = 20 min, Scram
5. Trip T/G			2220 Turbine Trip	2220 Main turbine tripped (23 sec. after scram)	
6. Maintain Rx vessel level		—	—	—	—
7. Monitor cooldown		—	—	—	—
8. Recover recirc. flow	65 min	2305- "B" Recirc pump on	—	—	—
9. Relief valve seated	~ 30 min	—	—	2228 - Reclosed at 400 psig	t=30 min Reclosed at 350 psig
10. Inspect torus		—	—	No damage (Date of inspection)	

Fig. 2. Raw Data Sheet for Event "Relief Valve Lifting."

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at 2208 hours. Each time a significant discrepancy such as this exists, it is necessary to make some judgement, based on all the information available, as to the actual timing of events and operator actions. In this case, examination of the available documents led to the conclusion that the time of 2200 was more nearly correct, and 2208 was probably the time at which the operator recognized what had happened. Whenever such a judgement had to be made and some question remained, there was every effort made to err in the conservative direction, i.e., to overestimate rather than underestimate the time required if an error was unavoidable.

A summary of availability of data on the selected events at the five sites visited collected from all available sources is provided by the tables in Appendix B. Pooling data from site documents appears to be a necessary element in gathering the desired operational-experience-based data. This is illustrated by Fig. 3, which shows the relative increase in quantitative data on specific actions taken during a number of occurrences of the event "relief valve stuck open" when site data are available. The significant addition of data from site documents is typical, though there is no question that a great deal of variability exists in completeness of any type of document from site-to-site and event-to-event.

Even with data pooling we did not feel that the amount of data collected from the few sites visited was sufficient to provide a sound basis for numerical estimates of time required for completion of specific operator actions. The amount of data available from the five plants does suggest, however, that it will be possible to gather sufficient data on a relatively small number of specific operator actions for use in calibrating simulator experiments. Additional data obtained by a more extensive search of NRC docket information (covering all operating BWRs and PWRs) for only a few selected events supports this conclusion. This more extensive docket search on "key events" is discussed in Section III.E.

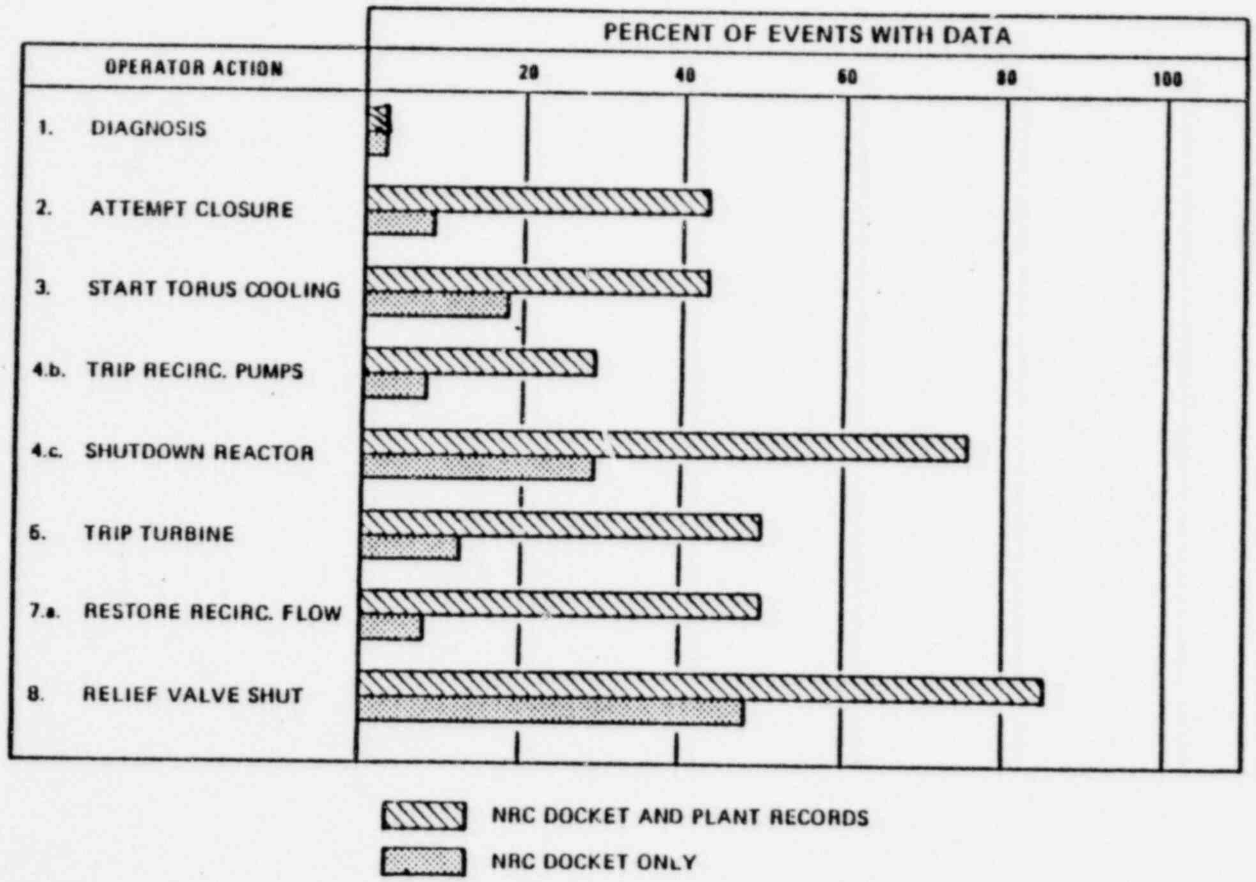

Fig. 3. Relative Effectiveness of NRC Dockets and Site Documents.

One additional factor should be considered when evaluating the availability of data on the basis of the present five-plant survey. In a number of instances one, or perhaps two, plants seem to have experienced a major portion of the total number of occurrences of a particular event. For example, Plant A in this study has experienced more than 20 inadvertent safety injections, approximately 40% of the total of 56 that were identified from docket searches for all U.S. PWRs. A recent survey of event occurrences at U.S. PWRs performed by EPRI[2] suggests this is not uncommon (e.g., 7 of 23 loss of one RCS loop events occurred in one plant, 53 of 95 CRDM/rod drop problems occurred at one plant, 39 of 88 partial feedwater flow losses occurred at two plants, etc.). The reasons for this behavior are not clearly defined, though one logical explanation is plant design differences. The implication for this study is that in a number of cases the bulk of data on one type of event may be gathered from just a few sites. The success with collecting data on inadvertent safety injections at Plant A suggests that a data base adequate to support the proposed simulator calibration study can be developed if the data-collection effort is extended to all operating reactors. The possibility for a data base dependent solely on operating experience is more questionable.

III.E. Docket Searches on Key Events

As noted above, the results of the preliminary study at the five plants gave a positive indication on the availability of data, but did not produce enough data on different operator actions to warrant making numerical estimates of operator response times. Since such estimates, if founded on a reasonable data base, are desired by the N660 writing group as soon as possible, it was decided to include in the preliminary assessment a more exhaustive search of NRC docket files. The additional time available for this aspect of the project was very limited so the search was restricted to a few events and specific operator actions; but it covered all operating PWRs and BWRs. The "key events" selected and the relative success in extracting quantitative data on each of the events are shown in Table 5.

was limited by date (thru 1977)

TABLE 5. Summary of Key Event Search

	EVENT	NO. IDENTIFIED	NO. FOUND	NO. WITH DATA	NO. WITH SITE DATA
BWR	Off-Gas Explosion	17	16	7	6
	Relief Valve Stuck Open	42	45	8	4
	Loss of AC Power	12	9	0	0
PWR	Inadvertent SI	56	49	44	18
	Loss of Service Water	3	3	1	0
	Loss of AC Power	16	14	2	2
	S/G Tube Leak	20	17	1	0

For three events - inadvertent safety injection, off-gas system rupture, and relief valve opening - it was possible to identify and accumulate sufficient data on specific operator actions to make initial numerical estimates of operator response times. The actions and estimated mean-times-to-respond are listed in Table 6.

For the safety injection event, the quantity of data was sufficient to perform a graphical analysis using probability plotting. The plot of response time vs. the cumulative probability that the response was performed within that time is shown in Fig. 4. The fact that the plot is nearly linear on logarithmic probability paper indicates that a log-normal distribution is a reasonable model. The median and standard deviation were estimated directly from the plot.

For the other two events, data were not sufficient for graphical analysis. A log-normal distribution was assumed, but confidence intervals were not estimated. The value of $\bar{T}$, the mean time to response for each action was estimated frcm:

$$\bar{T} = \left(\prod_{i=1}^{n} Ti\right)^{1/n}$$

where Ti = ith response time

n = number of response times in the sample.

The assumption of a log-normal distribution was based on three factors:

(1) The "intuitive" feeling that the distribution of operator response time would tend to cluster around a relatively low value and, because of the various event-dependent factors, spread to include (relatively) long times.

(2) The data collected on the safety-injection event suggested a log-normal distribution.

TABLE 6. Summary of Data Collected on Key Event Actions

EVENT	ACTION REQUIRED	FSAR	DATA		SURVEY	
			MEDIAN	90^{th} PERCENTILE	MEDIAN	90^{th} PERCENTILE
Inadvertent safety injection	Determine cause and secure S.I.	ASAP	1.7 min. (2.0* min.)	6.0 min.	.3 min.	1.25 min.
Off-gas system rupture	Clear area of personnel	< 2 min.	6.3* min.	NA	NA	NA
	Isolate affected SJAE	ASAP	13.6* min.	NA	NA	NA
Relief valve sticks open	Recognize problem and attempt to close valve	ASAP	5.1* min.	NA	.35 min.	1.0 min.
	Initiate torus cooling	ASAP	9.2* min.	NA	NA	NA

*Mean-time assuming log-normal response time distribution.

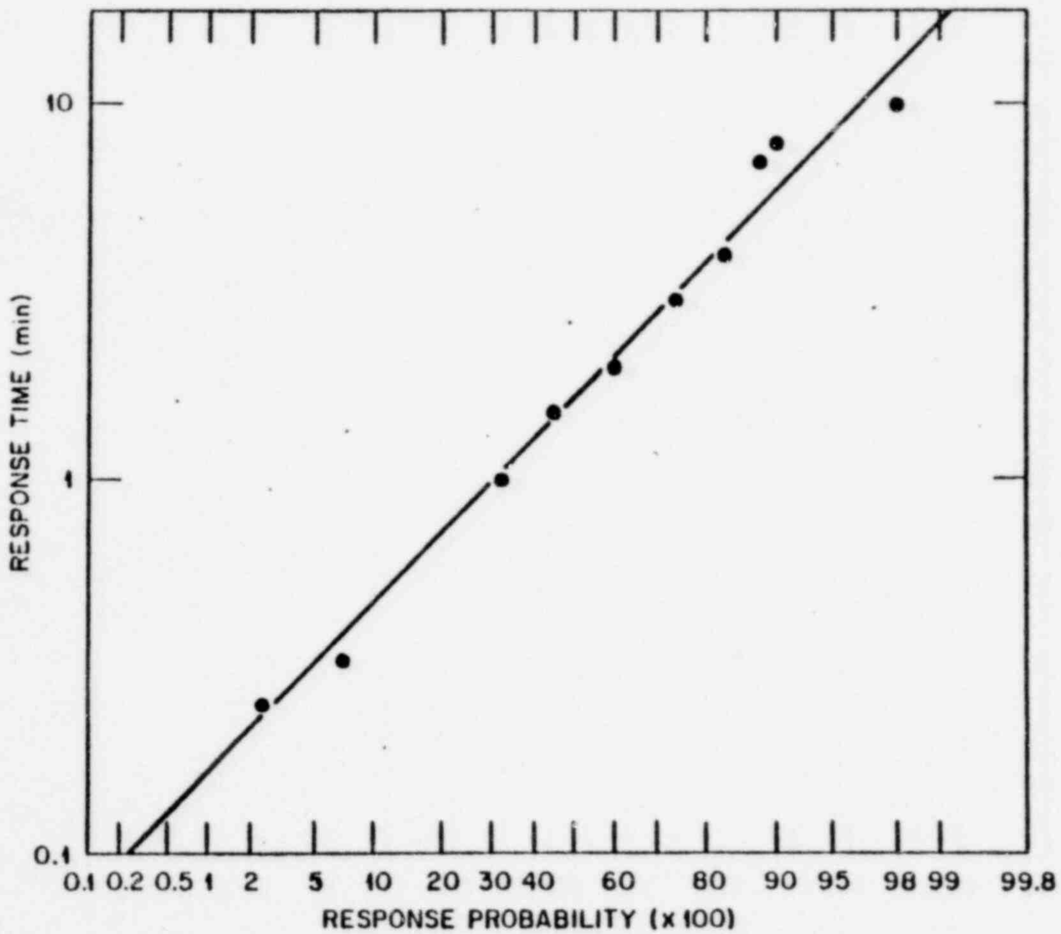

Fig. 4. Inadvertent Safety Injection Probability Plot.

(3) The response of operators to survey questions regarding the distribution of response times suggested a log-normal distribution. (Results of Operator Opinion Surveys are discussed in Section III.F.)

Although the confidence placed in these preliminary estimates should not be too great because of the few data points, especially for the latter two events, it is encouraging to see that initial data suggest mean times to respond that are within the range of values proposed in the draft of N660 released for trial use and comment. In particular, for the one event for which there is sufficient data to provide at least a moderate level of confidence, the data suggest a median time well within the proposed values, and a 90% confidence interval also within the suggested values. Further accumulation of data on these and other events, especially when combined with data from site visits should provide a firm basis for simulator calibration studies as well as some preliminary guidance for the N660 writing group and NRC licensing assessments.

III.F. Operator Opinion Surveys

In addition to an examination of site records, the site visits included administering a written survey form to operators in order to gather "expert opinion" on qualitative and quantitative aspects of operator behavior under the stress of severe accident conditions. Although consultant expertise in the areas of psychology and human factors was not available in time to assist in developing the form, informal reviews by qualified personnel have not indicated any major problems with the approach taken or the limited conclusions drawn from the survey. The goals of the survey during the preliminary assessment were more to establish the feasibility of using this approach to gathering useful information and to gain experience in developing and executing the survey than to extract data. Completion of the written survey was not a required portion of the contract. However, a good deal of qualitative information, and some quantitative information was produced by conducting the survey. Future work by trained psychologists could probably use a similar approach to add considerably to the understanding of operator response.

It should be noted that administration of the survey form did require a significant amount of time and coordination. Typically, the time required for an operator to complete the form was about one hour. It is difficult for an operating power plant to make available more than one or two licensed operators for an hour. Usually, it was possible to complete only a few surveys during a one or two day visit. At two plants, arrangements were made to administer the survey to a group of operators as part of their normal in-service training. This proved to be much more effective than individual interviews (for the written survey). At each of the five sites visited agreement was made with cooperating site management to complete additional surveys after the ORNL visit and forward them to ORNL. However, only in one case were a significant number forwarded. The reasons for the poor follow-up are not entirely clear; no doubt there were several factors involved. The point is that future work should recognize the difficulty of coordinating a voluntary survey of plant operators and plan in advance to secure the necessary strong cooperation of key site personnel, especially operating supervision, training directors, union representatives (if applicable) and operators. As much as possible, the surveys should be conducted by the party requesting it, during the site visit.

The form, which actually developed over a period of time during the study, consisted of some general background information on the respondent and then five questions. A complete copy of a blank form for PWR sites is included in Appendix C. Reference to Appendix C during reading the following paragraphs would be helpful.

III.F.1. <u>The General Model of Operator Response.</u> The first question outlined the general "model" of operator behavior which we interpreted as the basis for the time criteria proposed for N660, that is, that there are four distinct phases of behavior:

(1) Shock - initial period of reaction to a highly stressful situation during which no positive action is taken.

(2) Diagnosis - operator assesses available information, identifies event that has occurred and plans his corrective actions.

(3) Immediate Action - first corrective action taken as soon as possible after initiation of the event.

(4) Subsequent Action - additional corrective action taken over a longer period of time, presumably under a reduced stress level because immediate corrective action has brought the reactor to a recognizably safe condition.

The question then asked for a "yes" or "no" answer as to whether the model was a "reasonably accurate general description of operator response," and for the respondent to comment if desired. The response was to the written question was overwhelmingly "Yes." Out of 13 survey forms completed by BWR operators, 12 indicated "Yes" and one did not answer. Only one written comment was received - that there may in some cases be a second diagnostic phase between immediate and subsequent action. Out of the 30 PWR operators completing the survey, 25 responded "Yes," 2 responded "No" and 3 did not answer. Both of the negative responses made the same two points: (1) that the initial "shock" was nonexistent or so small as to be insignificant and that, (2) the diagnosis and immediate action occurs essentially simultaneously. Similar comments were made by three of the respondents giving "Yes" answers:

(1) *"The diagnosis and immediate action phases are not necessarily distinct from one another."*

(2) *"Surprise or realization is a more accurate term than shock for phase a."*

(3) *"Diagnosis and immediate action for most events" are pretty automatic due to training.*

These comments and a good number of verbal responses during discussions with the operator have led to three tentative conclusions on the overall model which might be pursued further in subsequent study:

(1) The initial period of shock immediately following annunciation of a serious event is felt by some operators to be nonexistant or negligible. However, (in responses to this and other survey questions) most of the operators recognized that the response time should be expected to be longer for events that were perceived as more stressful.

(2) In some cases, the diagnostic phase and immediate action phase are concurrent. The conceptual model of an operator assimilating all available information, identifying the event and organizing a plan of action prior to initiating action is not a realistic model for all cases. Discussion with a number of operators indicated that often the operator's corrective action is taken in response to a specific symptom - a low level, a high pressure - without waiting for complete and detailed information as to exactly what event has occurred, what the underlying cause of the event is or what step-by-step action is specified by written procedures. This "symptomatic response" is especially likely for events that are first annunciated by symptoms which obviously demand immediate action or for events in which symptoms develop over a period of time. In the latter case, it is more likely, according to a number of operators, that some response would be made to major symptoms essentially immediately, regardless of whether or not the complete information was available. The point was made that complete information is often not necessary to initiate the "proper" corrective action.

(3) There is an element of operator behavior that can "override" the orderly, step-by-step response suggested by the model during very serious events. Some operators indicated that they felt there are a few fundamental requirements that must be met to assure that the most severe consequence - a core meltdown - does not occur. Regard-

less of the initiating event, or the prescribed procedures, if the event has progressed to a very serious point, assuring these few fundamental requirements - primarily that rods are inserted and the core is covered with water - is the overriding objective. Actions will be taken to meet those requirements by whatever means are available. This "mini-max" principle, which says there are a few very fundamental actions/requirements that can be taken to limit the maximum consequences of any event, apparently results from extensive and intensive training emphasizing the potentially severe consequences of very unlikely events such as the LOCA. The implication is that if this is indeed a "universal" behavior, then the actions taken in response to very severe events may not correspond to specific procedures or patterns than can be readily prescribed by a general model or general criteria. On the more positive side, if indeed there are only a very few fundamental requirements to avoid maximum consequences, and these are deeply ingrained in operators during training, then the likelihood for their being incapable of taking the corrective action is rather low.

III.F.2. Stress, Difficulty of Diagnosis, and their Effects on Operator Response Times. The second question (paragraph IV) of the survey dealt with the psychological stress of emergency/abnormal events, the difficulty of diagnosing events, and the quantitative estimate by operators of the effects of these factors on time to respond. The approach used was strongly influenced by the critical incident technique described by Flanagan[3], though this preliminary work is far from a complete analysis. The survey form listed twenty different abnormal/emergency events (a different set for PWRs than for BWRs) which were felt by us to vary considerably in their likelihood for producing stress and in their difficulty of diagnosis. The operators were asked to first circle the events they had personally experienced. This was done so that in later analysis responses could be separated according to the level of experience or operator opinion could be "weighted" by personal experience if desired.

III.F.2.a. Stress and Difficulty of Diagnosis - Selection of Events. The operators were asked (questions IV.A and IV.B) to select the three events that in their opinion were most likely to induce psychological stress and explain why. Then they were to select the three events that they felt were least likely to induce stress, and to explain why. Similarly they were asked to select the three events that were most difficult and the three that were least difficult to diagnose and to give reasons why. The intent of this type of questioning is to extract from the operator, by focusing on specific incidents, what factors tend to cause stress and what factors make diagnosis more difficult.

If the technique were applied further and more rigorously, follow-up discussions on specific events within the operator's personal experience and more quantitative procedures might be applied to begin to quantify the effects of stress and to specifically identify design, human engineering or environmental factors which could be improved to reduce the difficulty of proper and prompt event diagnosis. In this initial work the goal was to establish only generally the key factors and their relative effect on behavior, and this seems to have been accomplished. Conclusions unfortunately were limited by the relatively few number of respondents, particularly from the three BWR plants. The discussions that follow present results from the thirty surveys completed at the two PWR plants visited. The twenty PWR events listed on the form are shown in Table 7. The events selected by the operators and their reasons given (when a reason was given) are summarized in Tables 8, 9, and 10. A more detailed listing of the responses is presented in Appendix D. Since the reasons and comments were solicited in free form, i.e., in the operator's own words, there had to be some interpretation and a concomittant loss of objectivity in grouping the responses into five general categories. However, the categories were defined from the responses of the operator, not a priori, and the responses did seem to group themselves into general categories.

The major points to note are that the primary reason given for stress is the fear of potentially severe consequences, either in terms of public hazard or plant outage, and the primary reason for difficulty of diagnosis is the occurrence of similar symptoms for different events. Other factors

TABLE 7. List of Events on PWR Survey

1. Loss of Feedwater Flow
2. Loss of Condensor Vacuum
3. Emergency Boration
4. Loss All AC Power
5. LOCA
6. Reactor Trip
7. Loss of Condensate/Cond. Booster Pump
8. Fuel Handling Emergency
9. Reactor Coolant Pump Vibration
10. High Coolant Activity
11. Main Steam Line Break
12. Steam Generator Tube Failure
13. Excessive Primary Plant Leakage
14. Reactor Coolant Pump Trip
15. Loss of Component Cooling
16. Continuous Rod Withdrawal
17. Loss of RHR System
18. Safety Injection
19. Rad. Mon. System-High Activity Alarm
20. Loss of Service Water

TABLE 8. Results of PWR "Critical Incident" Questions

NOTE: 30 RESPONDENTS AT TWO SITES

	EVENT	NO. OF CHECKS	PRIMARY REASON(S)*
MOST STRESSFUL	1. LOCA	25	#1 - Consequences
	2. Main Steam Line Break	21	#1 - Consequences
	3. Station Blackout	13	None (#3 Uncertainty)
LEAST STRESSFUL	1. High Activity Alarm Rad. Mon.	13	None #1 Consequences #2 Control #4 Demands
	2. High Coolant Activity	10	None (#2,#4)
	3a. Loss Condensate/Cond. Booster	9	None (#2,#4)
	b. Reactor Trip	9	Frequent Occurrence, Experience
MOST DIFFICULT	1. S/G Tube Failure	20	None #2 - Similar #1 - Inadequate
	2. Excessive Primary System Leakage	16	None (#2,#1)
	3. Main Steam Line Break	11	None (#2)
LEAST DIFFICULT	1. Reactor Trip	22	#1 - Adequate (First Out)
	2. Loss FW Pump	15	None (#1)
	3. RCP Trip	13	None (#1)

*See Tables 9 and 10 for further explanation of reasons. When the reason was most often left blank, the most frequently cited reason is shown in parenthesis.

TABLE 9. "Reason" Codes - Stress

	MOST STRESSFUL	LEAST STRESSFUL
1. Consequence	Potentially Major; Hazard to Self, Plant Personnel, Public; Potential Major Plant Damage (18)	No Major Consequences (7)
2. Plant Control	Fear of "Losing Control of Plant"; Related to Both Consequences and Performance and Uncertainty (3)	Plant Under Control, Not Likely to Get Out of Control (5)
3. Uncertainty	Not Clear What is Happening, What Action to Take or What Sequence of Events Will Follow (6)	Event Readily Identified, Subsequent Action Events Clearly Understood and Anticipated (0)
4. Performance Demands	Fear of Performing Poorly, e.g., Allowing Plant to Scram when Prompt, Accurate Action Could Prevent Scram (4)	Demands on Operator Performance Not Severe, No Fear of Failure (13)
5. Other	Inexperience with Event; Event Cannot be Affected By Operator Action (5)	Experience with Event; Backup Systems (4)

NOTE: The numbers in parenthesis indicate the number of times this general category of reason was given by the operators.

TABLE 10. "Reason" Codes - Diagnosis

	MOST DIFFICULT	LEAST DIFFICULT
1. Annunciation Adequacy	Inadequate for Prompt Detection (i.e., No Immediate Alarm or Alarm Not Directly Related to Event) (4)	Immediate, Direct Annunciation (14)
2. Discrimination of Symptoms	Symptoms Well Annunciated, but More Than One Event has Similar Symptoms; Cannot Distinguish which is Occurring (12)	Symptoms Clearly Delineate which Event; Multiple Indications of which Event (6)
3. Information Precision	Adequate Annunciate; Symptoms Clearly Indicate Event, but Details not Sufficient to to Indicate Proper Action (5)	Detail of Information Either Adequate or Totally Unnecessary (1)
4. Event Time	Event Time is Long; Symptoms Take Time to Develop After Alarm; Tension Builds (2)	Event Time Short, Diagnosis either Immediate or Not Required (0)
5. Other	Inexperience with Event; Rely on Someone Else (2)	Experience and Training Give Great Confidence Can Recognize (1)

NOTE: The numbers in parenthesis indicate the number of times this general category of reason was given by the operators.

expressed as contributing to stress are the fear of poor performance (as perceived by peers or supervisors) as well as feelings of "uncertainty" - either in what the proper action is or in a more general sense of not being in control of the situation. The difficulty in diagnosis, according to these responses, is almost always related to adequacy of annunciation - either in the ability to distinguish among events, or the promptness, clarity and completeness of annunciation. A few comments noted an increased difficulty of diagnosis if the event involved information that could only be obtained outside the control room, e.g., in the case of an off-gas explosion. There was little direct indication that lack of familiarity with the event either in actual experience or training was a major concern in diagnosing the event, nor was it a major reason listed as causing stress. However unfamiliarity with a particular type of event is certainly indirectly involved with comments on uncertainty as to what action to take.

As noted previously, further work could be directed toward more specific identification and greater quantification of the factors causing stress and difficulty of diagnosis. For example, techniques such as "paired comparisons" commonly used by psychologists to establish a quantitative expression of expert opinion might be applied. Another area of interest for further investigation, if the indications from this preliminary survey are shown to be valid, is development of improved capability for diagnosing the causal event. Such development might include simply improved annunciation or perhaps computer assistance in diagnosis.

III.F.2.b. Effects of Stress and Difficulty of Diagnosis on Time to Respond. A second portion of the survey question on stress and difficulty of diagnosis (Question ~~IV.C~~ IV.C.1 and IV.C.2) asked the operators to consider the combined effects of both stress and diagnosis problems and to select the three events which they felt would take the most time for operator response and the three events which would take the least time. "Operator response" was defined as the time required for the operator to recover from initial shock, correctly diagnose the event, plan his action and initiate the required immediate action. It was pointed out to the operators that the selection of events for this response may or may not correspond to their previous selections of most (least) stressful or most (least) difficult-to-diagnose events. The operators were instructed to indicate for each of the events selected whether they felt the event would cause "moderate," "high," or "severe" stress. Finally they were asked to estimate the time required for them to respond to one of the events requiring the most time, and to one requiring the least time. This was the first attempt to gather quantitative "expert opinion" on the required time for responses.

Responses from the PWR operators are shown in detail in Appendix D and are summarized in Table 11. The selection of events suggests that both stress and difficulty of diagnosis are perceived as contributors to increasing the time required for operator response with perhaps stress being more important. The two events selected as requiring the most time were also selected as the most stressful, while the third, steam-generator tube failure, had been selected as one of three most difficult to diagnose. The events selected as requiring more time were generally noted as causing higher stress levels than the "least-time" events. The operator estimates of their response time averaged from about one-half minute to a minute for the least time to about a minute and a half to two minutes for the most time The times listed as "average of all time estimates" is the numerical average of all the response given for each category.* The correlation noted in Table 11 is the number of total responses, i.e., event numbers selected as requiring either "most time" or "least time," which were also selected by that operator as most (least) stressful or most (least) difficult to diagnose. The "correlation" merely gives further indication of the relative emphasis the operators place on the two contributing factors.

III.F.3. <u>Operator Estimation of the Response-Time Distribution.</u> Question IV.C.3 ~~IV.C~~ of the survey was an attempt to get the operators to estimate the distribution in operator response time. The operators were asked to assume that a large number of typical operators independently experienced an event

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*Note in Appendix D that operator estimates were not always precise and some interpretation was required. A conservative approach was taken. For example, an estimate of 1 to 2 minutes was taken as 2 minutes, a response of "less than 30 seconds" was taken as 30 seconds.

TABLE 11. Results of PWR Operator Survey - Time Estimates

EVENTS REQUIRING THE MOST TIME FOR RESPONSE

EVENT	CHECKS	STRESS LEVEL*	AVERAGE TIME (SEC.)
1. LOCA	21	High (2.05)	122
2. Main Steam Line Break	20	High (1.89)	107
3. S/G Tube Failure	10	High (1.78)	98

Average of All Time Estimates = 111 Sec.

Correlation: 45 Out of 84 Corresponded to "Most Stressful".
31 Out of 84 Corresponded to "Most Difficult".

EVENTS REQUIRING THE LEAST TIME FOR RESPONSE

EVENT	CHECKS	STRESS LEVEL	AVERAGE TIME (SEC.)
1. Reactor Trip	18	Moderate (1.38)	32
2. Loss FW Pump	14	Moderate (1.58)	24
3. Continuous Rod Withdrawal	8	Moderate (1.25)	36
4. Loss Condensate/Cond. Booster Pump	8	Moderate (1.0)	67

Average of All Time Estimates = 35

Correlation: 31 Out of 81 Corresponded to "Least Stressful".
35 Out of 81 Corresponded to "Least Difficult".

*The numbers in parenthesis under the column "Stress Level" are to be used simply as a rough guide in quantifying the operator response. They are the numerical average of the operators response obtained by assigning a value of 1.0 for "moderate," 2.0 for "high" and 3.0 for "severe."

which had been selected in Question IV.C.1 ~~IV.8~~ as one of those requiring the most time, and that all equipment performed as designed. A list of times ranging from less than 30 seconds to 30 minutes was provided, and the operators were asked to specify opposite those times, what would be the cumulative percentage of operators who had responded by that time. A distribution of response time for occurrence of one of the "least-time" events was requested in a similar manner.

Results of the PWR responses for the "most-time" events are illustrated in Fig. 5. The estimates of percent that would complete response within a specified time were averaged and plotted as a function of time. The error bars indicate one standard deviation on the average value. Note that the variation in operator estimates is rather large (numerically as great or greater than the value of the average itself) for time periods below about one minute, and the "confidence interval" narrows with increasing time. The shape of the cumulative distribution in Fig. 5 suggests a log-normal distribution, and Fig. 6, which is a probability plot of the logarithm of the time vs. the estimated percent responding, confirms that a log-normal distribution is a reasonable model. The median and standard deviation were estimated from the plot in Fig. 6. Values of the median time and times corresponding to several different confidence levels are shown in Fig. 6.

Question V in the survey attempted to gain similar quantitative information using a slightly different approach. Five specific events (different events for PWRs and BWRs) were specified for the operator and he was asked to provide an estimate of the mean and 90% confidence interval of the estimated distribution of response times. The wording of the question (see Appendix C) and verbal instructions explained the concepts of a distribution, a mean and a 90% confidence interval. Table 12 lists the five events selected for the PWR survey and the mean (arithmetic average) $\bar{x}$ and standard deviation σ of the operator estimates. Note again the rather wide dispersion in the values of the estimates.

In Fig. 7, operator estimates for time to respond to the event "inadvertent safety injection" are compared to data collected from operating

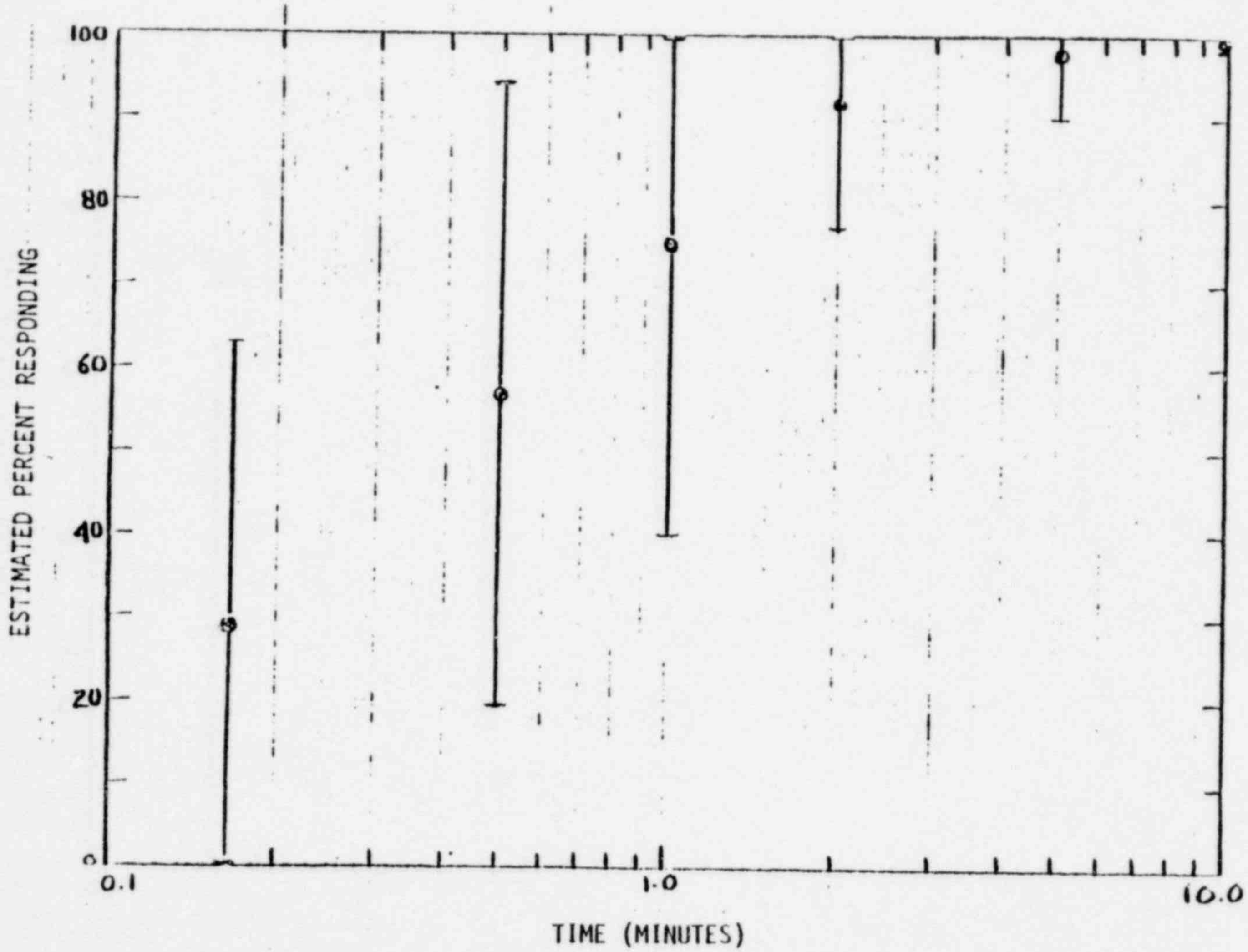

Fig. 5. Operator Estimates of Cumulative Distribution of Response Time for "Most-Time" Events.

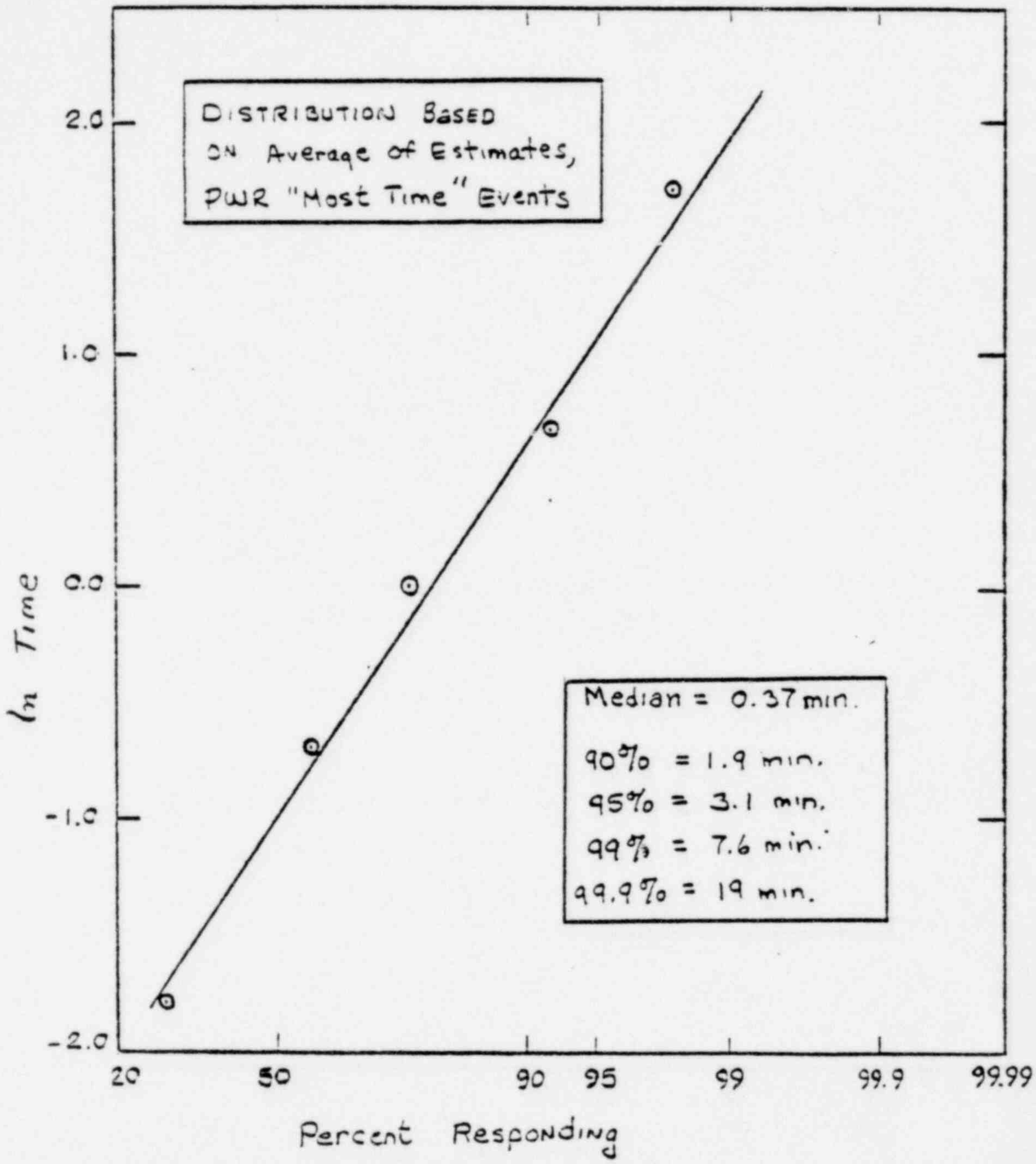

Fig. 6. Probability Plot of Operators Estimate of Cumulative Distribution of Response Time to "Most-Time" Events.

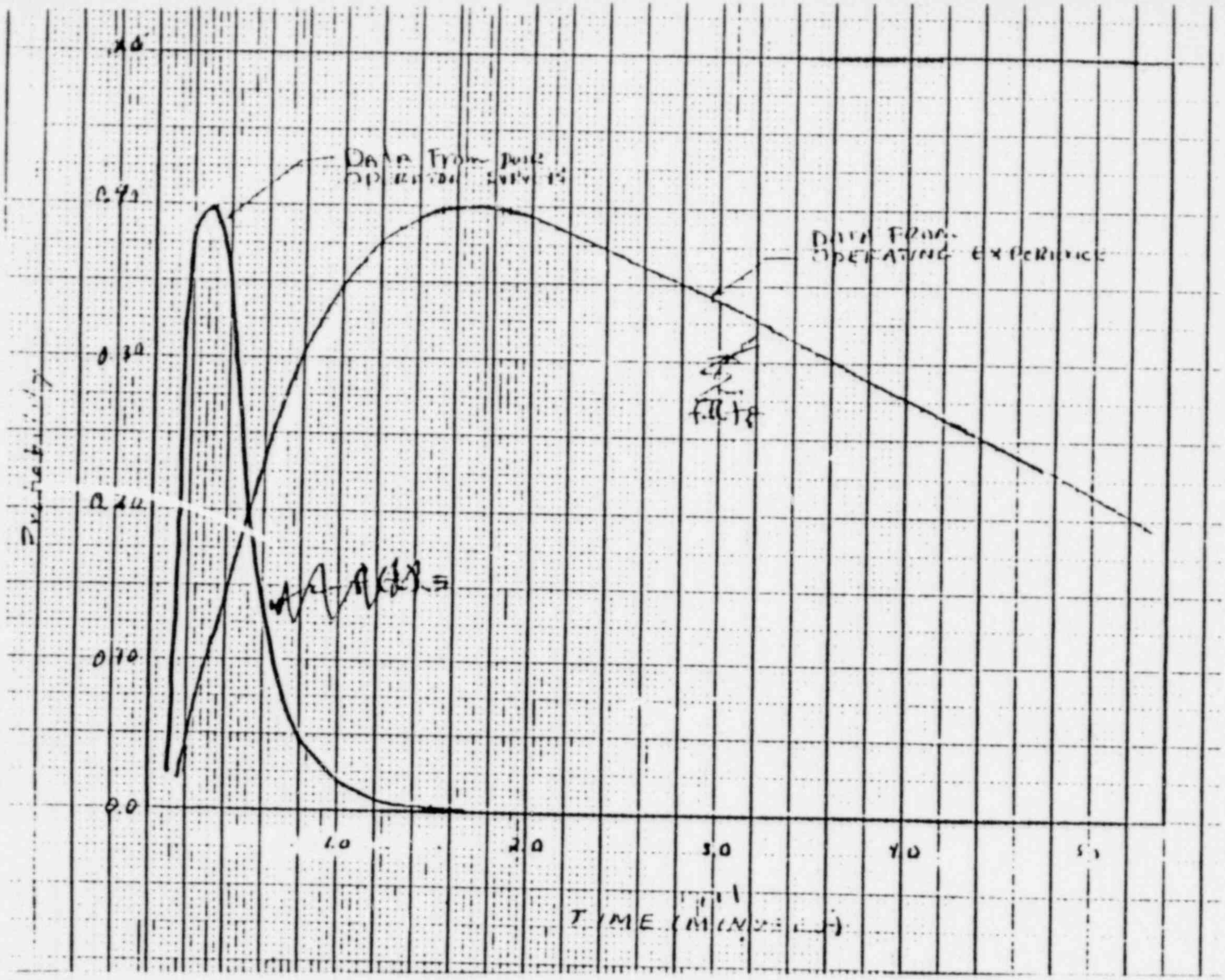

Fig. 7. Probability Density Functions for Response Time Estimated from Operator Surveys and from Operating Experience.

TABLE 12. Question V - PWR Operator Survey Results

EVENT	50th PERCENTILE $\bar{x}$ (sec)	50th PERCENTILE s (sec)	90th PERCENTILE $\bar{x}$ (sec)	90th PERCENTILE s (sec)
Steam Generator Tube Rupture	67	71	287	670
Automatic Control Rod Withdrawal	21	17	49	63
Station Blackout	42	84	93	137
Automatic Safety Injection	34	48	75	109
High Primary Coolant Leak Rate (< 100 gpm)	125	190	413	774

experience which were presented previously in Fig. 4. The plots in Fig. 7 are probability density functions for the (assumed) log-normal distributions. For the case of the actual data, the parameters of the distribution were estimated from the plot in Fig. 4. For the case of the operator survey, the median of the assumed log-normal distribution was taken as the average value $\bar{x}_{50}$ = 34 sec from the operator estimates, and the shape parameter σ was calculated from:

$$\sigma = 0.78 \log \frac{\bar{x}_{90}}{\bar{x}_{50}}$$

where $\bar{x}_{90}$ = the average of operator estimates for the 90 percentile (75 sec). The comparison shows that for this event the operator estimates of their response time distribution is qualitatively similar to the data, but quantitatively underestimates the time.

Although the analysis was not pursued further in this preliminary phase, the work accomplished illustrates that operator opinion could be gathered and used to provide qualitative and quantitative information on the expected response time of operators. The qualitative agreement of the operator estimates with the data from actual occurrences provides some indication that operator opinion could be used successfully to add to the quantitative data base if necessary. Specifically, a rigorous Bayesian analysis could use operator opinion to develop a prior distribution which could then be combined with relatively sparse data from operating experience, if desired.

IV. CONCLUSIONS AND RECOMMENDATIONS

IV.A. Conclusions and Recommendations on the Draft N660 Criteria

The conclusions from our review of the draft N660 criteria are:

(1) The conceptual model of operator response assumed as a basis for the criteria is reasonable for many events, but may not be generally applicable. For some events, especially when symptoms are manifested over a period of time, diagnosis and action may be a concurrent process.

(2) The use of a time margin for operator response is a reasonable approach for interim criteria, i.e., until a comprehensive human factors study is completed. Ultimately, criteria should be based on performance of tasks according to capability. That is, operators should perform those tasks for which it's determined human action is more reliable than automatic action.

(3) Increasing the time margin with stress and difficulty of diagnosis is appropriate. However, quantification of stress and difficulty of diagnosis is extremely difficult. We have no basis for judging whether the current basis for extending the time limit, i.e., expected frequency of occurrence of the event, is adequate. Professionally administered and analyzed psychological surveys of operator opinion should be very useful in quantifying these two factors.

(4) This preliminary assessment did not provide enough data to make a judgement concerning the quantitative values for time margins that have been suggested at different times during the development of the draft criteria. However, data collected to date do not support the use of times much greater (e.g., one hour) than were included in the draft released for trial use and comment.

Our recommendation is to proceed with development of interim criteria based on the best available information, and to modify time margins as data from this and/or other programs accumulate. The validity and significance of the behavior patterns which we have designated "symptomatic response" and "minimum-maximum" response should be investigated further, and if necessary, accommodated within interim criteria.

IV.B. Conclusions and Recommendations on Data Availability

The conclusion on availability of data is that there probably are sufficient data to provide a data base for calibrating simulator results. There probably are not enough applicable operating data to develop an adequate quantitative data base solely from operating experience. Data collection is a tedious, labor intensive, frustrating task, and a number of problems should be recognized:

(1) The potential for a "pure" data base, i.e., one that includes purely objective measurement of operator action strictly applicable to N660 criteria, is small. Judgement, inference and extrapolation will be necessary.

(2) Specific definition of actions and events is made difficult because there is great variability from event-to-event and site-to-site.

(3) Searching docket information is a non-trivial task which is not fully automated. There is no single source that is 100% complete and there are some problems with retrieving desired documents.

(4) Interpretation of site records, especially computer output, requires skilled site personnel.

(5) Coordination of the site visits, operator interviews and records searching requires a considerable effort and a significant contribution of site staff time; the availability of site personnel is limited.

(6) Operator survey and other voluntary follow-up work by site personnel will probably not be effective.

Our recommendations are that a decision by NRC should be made as to the relative benefit of continued work to support interim criteria vs. initiation of a more comprehensive human factors study. If the decision is to continue this work then it is recommended to:

(1) Proceed with data collection as outlined in this study for the purpose of gathering enough data to develop a correlation between simulator results and operating experience. The data collection procedure is outlined in Fig. 8.

(2) Use carefully constructed, professionally administered operator surveys for qualitative information and to help quantify perceived stress levels and difficulty of diagnosis.

(3) Gather and assess all available data on operator response under stress from other industries such as aviation, fossil-fuel power plants, or chemical process plants which could be applied to nuclear operator response.

It should be noted that the work proposed for continuation of this study would be an integral part a comprehensive human factors evaluation.

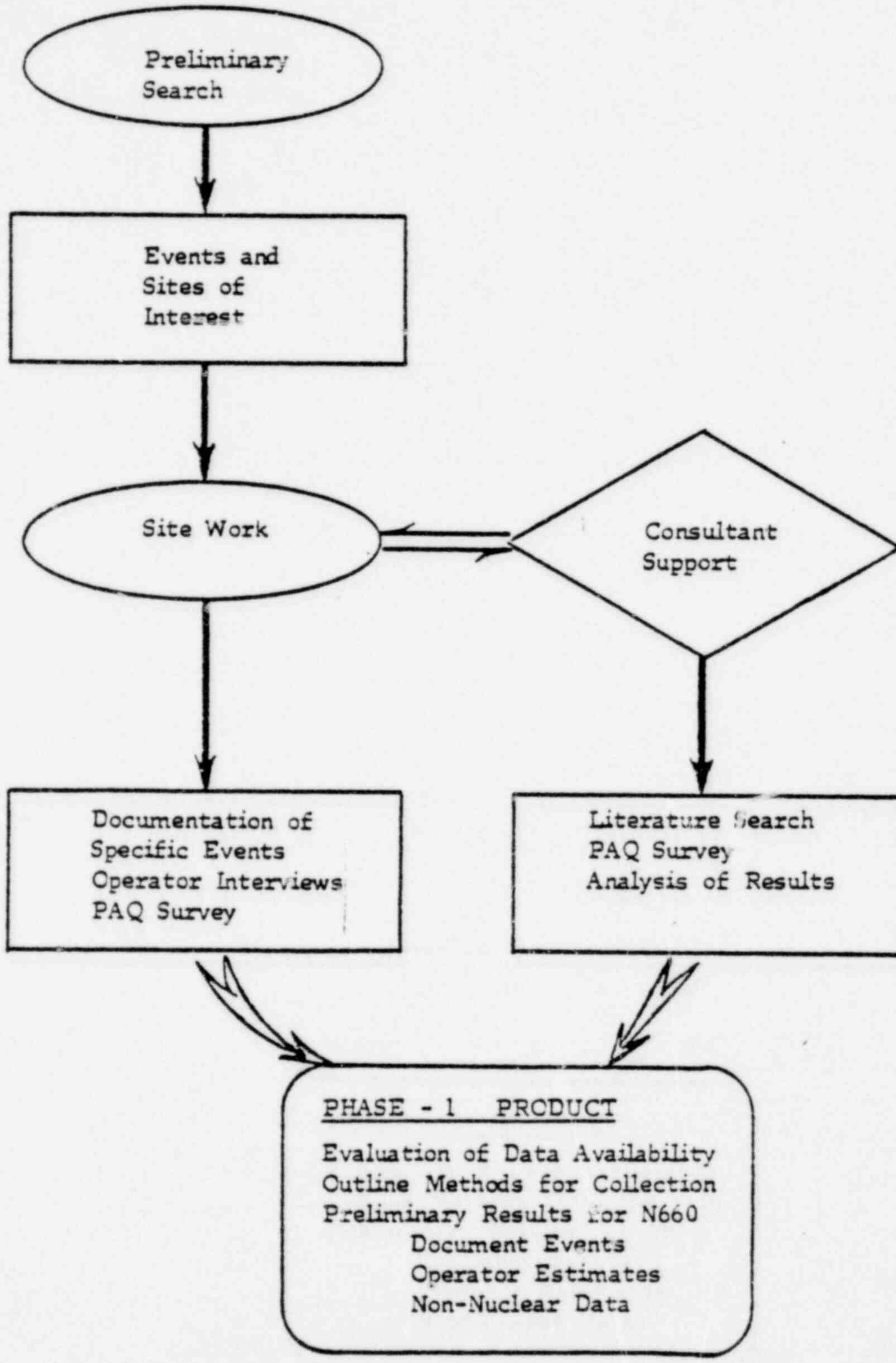

Fig. 8. Outline of Procedure for Collection of Data from Nuclear Plant Operating Experience.

REFERENCES

1. "Criteria for Safety Related Operator Actions," ANSI N660, ANS-51.4, Draft 12, Rev. 1, November 1976.

2. R. C. Erdmann, et al., "Probabilistic Safety Analysis III," EPRI NP-749, April 1978.

3. J. C. Flanagan, "The Critical Incident Technique," Psychological Bulletin 51, 327-358, July 1954.

APPENDIX A

Examples of Raw Data Sources

113542
SWITCHYARD BATTERIES NOT TESTED ON SCHEDULE AT MILLSTONE 1
NORTHEAST NUCLEAR ENERGY CO., HARTFORD, CT
3 PGS. LTR W/RO-76-14/3L TO NRC OFFICE OF I & E, REGION I, APRIL 30, 1976, DOCKET 50-245, TYPE--BWR, MFG--G.E., AE--EBASCO

CAUSE - ADMINISTRATIVE CONTROL. WHILE OPERATING AT 40% POWER, IT WAS FOUND THAT THE SWITCHYARD BATTERIES WERE TESTED ONE WEEK BEYOND THE LIMIT. THE CAUSE WAS AN ADMINISTRATIVE ERROR ON THE PART OF THE INDIVIDUAL RESPONSIBLE FOR THE TEST. NO DAMAGE OCCURRED.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545 (10 CENTS/PAGE -- MINIMUM CHARGE $2.00)

TESTING + FAILURE, ADMINISTRATIVE CONTROL + TEST INTERVAL + MILLSTONE 1 (BWR) + BATTERIES & CHARGERS

113541
PRIMARY CONTAINMENT ISOLATION VALVE FAILS TO CLOSE COMPLETELY AT MILLSTONE 1
NORTHEAST NUCLEAR ENERGY CO., HARTFORD, CT
3 PGS. LTR W/RO-76-16/1T TO NRC OFFICE OF I & E, REGION I, MAY 4, 1976, DOCKET 50-245, TYPE--BWR, MFG--G.E., AE--EBASCO

CAUSE - INTERNAL BINDING. DURING AN INVESTIGATION OF THE DRYWELL VENT VALVE BYPASS VALVE FOR LOSS OF CLOSED INDICATION WITH THE REACTOR AT 100% POWER, IT WAS FOUND THAT THE VALVE WOULD NOT GO FULLY CLOSED EVERY TIME IT WAS CYCLED. FAILURE APPEARS TO BE ATTRIBUTABLE TO INTERNAL BINDING. THE VALVE WAS MANUALLY CLOSED. IT WILL BE REPAIRED AT THE NEXT PLANT OUTAGE.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545 (10 CENTS/PAGE -- MINIMUM CHARGE $2.00)

CONTAINMENT + FAILURE, EQUIPMENT + CONTAINMENT ISOLATION + MILLSTONE 1 (BWR) + VALVES

113538
LEVELS OF ACTIVITY IN OYSTERS EXCEED 10 TIME CONTROL AT MILLSTONE 1
NORTHEAST NUCLEAR ENERGY CO., HARTFORD, CT
3 PGS. LTR W/RO-76-19/40 TO NRC OFFICE OF I & E, REGION I, MAY 7, 1976, DOCKET 50-245, TYPE--BWR, MFG--G.E., AE--EBASCO

CAUSE - WASTE DISPOSAL. OYSTERS GATHERED AT THE COLLECTION POINT WITHIN 500 FEET OF THE DISCHARGE IN FEBRUARY 1976 WERE FOUND TO HAVE LEVELS OF ACTIVITY THAT EXCEEDED THE CONTROL STATION VALUES BY OVER A FACTOR OF TEN. THERE APPEARS TO BE A DECREASING TREND SINCE EARLY 1975. THE AUGMENTED LIQUID RADWASTE TREATMENT SYSTEM FOR UNIT 1 HAS BEEN IN OPERATION SINCE MARCH 15, 1976.

AVAILABILITY - NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20545 (10 CENTS/PAGE -- MINIMUM CHARGE $2.00)

WASTE DISPOSAL + WASTE DISPOSAL, LIQUID + RADIOISOTOPE + MONITORING PROGRAM, ENVIRONMENTAL + MILLSTONE 1 (BWR) + RADIONUCLIDE UPTAKE + ANIMAL, INVERTEBRATE + CONCENTRATION

Fig. A.1 Example of NSIC Keyworded Abstract

Description of Occurrence:

At approximately 35 power with steady plant conditions, the "J" relief valve opened and closed dropping plant electrical load by about 70 Mw. This occurred twice within a few minutes.

Designation of Apparent Cause of Occurrence:

Ground on the control cable to the "J" relief valve manual actuation solenoid.

Analysis of Occurrence:

Because this valve is not part of the Automatic Depressurization System and because the failure and subsequent lifting of leads affected only the capability of manually operating the relief valve from the control room, the safety implications are minimal.

Corrective Action:

Investigation showed that one side of the relief valve manual actuation solenoid was solidly grounded and that there was a concurrent intermittent ground on the d.c. system supplying control power to this valve. This combination of grounds was responsible for the unexpected operation of the relief valve solenoid. The leads were lifted to the manual actuation solenoid, removing only the manual actuation capability. The ground will be further investigated and corrected when practical.

Failure Data:

No previous failures of this type.

Very truly yours,

POOR ORIGINAL

Ass't Gen'l Superintendent
Generation Division

cc: Mr. J. P. O'Reilly,
Director, Region I
United States Atomic Energy Commission
631 Park Avenue
King of Prussia, PA 19406

Fig A.2 Example of NRC Docket File (Abnormal Occurrence Report)

3:30 P to 11:30 P 3/18/74

- Rtr. Power 47.6% 491 Mwe
- LPRM's 08-17(B) & 08-25(A) went high and then cleared
- Several LPRM's spiking
- Turb Stop Valve Surv. completed
- Could not complete Cont. Valve fast closure Surv. due to power spikes affecting fuel preconditioning.

7:25 P Reactor scram from MSIV closure - Lab. doing surv. on LIS 2-2-72B. When valving instr. back in service LIS 2-2-58A & 58B tripped causing MSIV closure and A & B Recirc Pp. trip. - Tripped Turbine

- Opened 71.D Relief Valve to lower Rtr. press. (~1080#). 71D would not reclose Tried several times unsuccessfully. Could not reopen A & B Recirc Pp Disch Valves to ~~start~~ run Pumps - Had to take

Fig. A.3 Example of Operator's Log

POOR ORIGINAL

287

11³⁰/P TO 7³⁰/A MONDAY Nov. 4, 1974

#2 REACTOR POWER @ 100%, 1091 MWE & 3293 MWT

#3 REACTOR POWER @ 48.7%, 500 MWE & 1603 MWT – "C" ROD SWAP IN PROGRESS

2⁴⁰/A #2 RELIEF VALVE 71L LIFTED AND WOULD NOT RESEAT

2⁴²/A 2A & 2C RHR AND HPSW PUMPS ON – TORUS COOLING MODE

2⁴⁷/A #2 RTR. PRESS. WAS LOWERED TO 955 PSIG IN AN ATTEMPT TO CLOSE 71L RELIEF – STILL WOULD NOT RESEAT

2⁵¹/A #2 TRIPPED BOTH RECIRC PUMPS MANUALLY

2⁵³/A #2 REACTOR MANUALLY SCRAMED – TORUS TEMP ~ 90°F

2⁵⁴/A #2 TURBINE MANUAL TRIP

2⁵⁵/A #2 B & C CIRC WTR PUMPS OFF

3⁰⁰/A #2 RTR RECIRC PUMPS RESTARTED

3⁰²/A #2 RWCU SYS ISOL. RESET AND IN SERVICE

3⁰⁵/A #2 SJAE & RECOMB OFF, MACH. VAC PUMP ON

3¹⁰/A #2 D/W PURGE ON TO DE-INERT

3³⁰/A 2G3 DISC. SW. OPEN AND 215 CB & 225 CB RECLOSED

3⁴⁵/A #2 RELIEF VALVE 71L CLOSED @ ~207 PSIG, TORUS TEMP ~110°F

6²⁰/A #2 C RHR & HPSW PUMPS OFF

6⁴⁰/A #2 TURB STM SEALS ON AUX STM.

Fig. A.9 Example of Supervisory Log

POOR ORIGINAL

ALARMS: H_2 SPLY LINE LOW PRESSURE
RADWASTE BLD LOW D.P.
[illegible] TURB RM 2/3 AIR FILTER RUNOUT
DRYWELL II H_2 LINE PRESS HIGH
RECIRC MTR/MW II CLR RETURN AIR HI TEMP

[illegible] UNIT II ON SYSTEM
[illegible] "B" MAX RECYCLE NEUT; 95%, PH 7.5 CL 20
[illegible] SECURED STANBY GAS SYS, UNIT II H.P.C.I [illegible]
AND U II DIESEL GEN COOLING WTR [illegible] OFF
17.9 2/3 "E" LIFT [illegible] OFF
1745 TRANSFERED UNIT II AUX POWER
1[illegible] OPENED CCB 2-3 & 2-7, FOR U II SYNC
1[illegible] CLOSED TRAN 2 SYS K.V DISCS, & CHECKED CLOSE
2[illegible] LAKE LEVEL [illegible], [illegible] FLOW [illegible], [illegible] FLOW [illegible],
GATES: 55[illegible],
2[illegible] "B" WASTE SAMPLE TK ON RECIRC FOR SAMPLE
2130 2/3 "E" LIFT [illegible] ON
2[illegible] "B" WASTE SAMPLE TK SAMPLED
2[illegible] "B" WASTE SAMPLE TK → 2/3 "A" STORAGE

Fig. A.5 Example of Central Desk Log

POOR ORIGINAL

PLANT UPSET NO. ______________

Unit 2 was operating at 100% power, when the E.T.S. fluid line to the #4 control valve sheared. The resulting rapid pressure fluctuations produced repeated half scrams from the control value fast closure pressure switch. The continuing oil loss resulted in a low EHC fluid reservoir level and an overall system pressure reduction at the control values.

At this point, the control valve fast closure pressure switches produced a full scram signal andthe turbine control values closed due to low oil pressure to their disk dump mechanisms. The combination of scram and control valve closure resulted in a reactor water level shrink that initiated a group I,II, and III isolation. The reactor vessel was now isolated and automatically controlling level and pressure via relief valves, HPIC and RCIC.

The turbine generator output rapidly fell to zero, and beyond (no steam supply). The EHC accumulators held the pressure high enough, for long enough at the EHC low pressure turbine trip switches, so that the turbine trip was initiated by the generator reverse power relay. (The reverse power relay was found in the tripped condition along with the 386F lockout relay that it operates).

The reverse power relay trip and its association generator lockout relay produced the standard turbine generator trip and shutdown actions, ie, generator output breaker trip, unit aux bus fast transfer, Recirc M.G. set trip and turbine trip.

With the turbine generator tripped and reactor water level and pressure under control by the manual application of the HPCI and relief valve systems, repairs to the ETS fluid line were initiated.

Plant transient effects were minimal, since the repair work was simple and quickly accomplished. This allowed the isolation to be reset, and the MSIV's to be re-opened with in a half hour of the scram at which time, pressure control was transferred to the turbine bypass values with the main condenser as the heat sink. All reactor systems were returned to normal and a stable hot standby condition was achieved within one hour of the scram.

An inspection of the ETS tubing to the control valves did not reveal any further defects and a plant start up was commenced within 2 hours of the scram.

Fig. A.6 Example of Special Report

```
120015 21 D535 RTR AUTO  SCRAM  B    TRIP
120015 30 D544 TURB CV   CLOSE  D    NORM
120015 43 D544 TURB CV   CLOSE  D    TRIP
120015 52 D544 TURB CV   CLOSE  D    NORM
120016 40 D544 TURB CV   CLOSE  D    TRIP
120016 55 D544 TURB CV   CLOSE  D    NORM
120018 53 D535 RTR AUTO  SCRAM  B    NORM
120019  3 D544 TURB CV   CLOSE  D    TRIP
120019  3 D535 RTR AUTO  SCRAM  B    TRIP
120019 11 D535 RTR AUTO  SCRAM  B    NORM
120019 11 D544 TURB CV   CLOSE  D    NORM
120019 35 D544 TURB CV   CLOSE  D    TRIP
120019 36 D535 RTR AUTO  SCRAM  B    TRIP
120019 44 D535 RTR AUTO  SCRAM  B    NORM
120019 44 D544 TURB CV   CLOSE  D    NORM
120020 12 D544 TURB CV   CLOSE  D    TRIP
120020 12 D535 RTR AUTO  SCRAM  B    TRIP
120020 21 D544 TURB CV   CLOSE  D    NORM
120020 32 D544 TURB CV   CLOSE  D    TRIP
120020 41 D544 TURB CV   CLOSE  D    NORM
120020 45 D544 TURB CV   CLOSE  D    TRIP
120417 58   98.25  497.9 29.872 107.33  1069.  3298.  100.1  196.8
120020 54 D544 TURB CV   CLOSE  D    NORM
120021 32 D544 TURB CV   CLOSE  D    TRIP
120021 40 D544 TURB CV   CLOSE  D    NORM
120022  7 D544 TURB CV   CLOSE  D    TRIP
120022 15 D544 TURB CV   CLOSE  D    NORM
120023 23 D544 TURB CV   CLOSE  D    TRIP
120023 31 D544 TURB CV   CLOSE  D    NORM
120023 32 D544 TURB CV   CLOSE  D    TRIP
120023 41 D544 TURB CV   CLOSE  D    NORM
120024  2 D544 TURB CV   CLOSE  D    TRIP
120024 11 D544 TURB CV   CLOSE  D    NORM
120024 31 D544 TURB CV   CLOSE  D    TRIP
120024 40 D544 TURB CV   CLOSE  D    NORM
120024 43 D544 TURB CV   CLOSE  D    TRIP
120024 52 D544 TURB CV   CLOSE  D    NORM
120025 15 D544 TURB CV   CLOSE  D    TRIP
120025 24 D544 TURB CV   CLOSE  D    NORM
120025 33 D544 TURB CV   CLOSE  D    TRIP
120025 42 D544 TURB CV   CLOSE  D    NORM
120025 46 D544 TURB CV   CLOSE  D    TRIP
120025 54 D544 TURB CV   CLOSE  D    NORM
120026  1 D544 TURB CV   CLOSE  D    TRIP
120026  9 D544 TURB CV   CLOSE  D    NORM
120026 25 D544 TURB CV   CLOSE  D    TRIP
120026 34 D544 TURB CV   CLOSE  D    NORM
120548 30 D500 CRD VOL   HI LVL A    HIGH
120548 30 D501 CRD VOL   HI LVL B    HIGH
120548 30 D502 CRD VOL   HI LVL C    HIGH
```

Fig. A.7 Example of Plant Computer Output

POOR ORIGINAL

APPENDIX B

Summary of Availability of Data on Occurrences at the Five Sites Visited During the Preliminary Assessment

TABLE B.1. Data Availability for Site A (2-Unit PWR) Events

Event	Docket	Reactor Log	Supervisory Log	Computer Output
1. Automatic rod withdrawal	No time-response data	Good qualitative description; no time-response data	No time-response data	Not located
2. Reactor coolant leak	Detailed time-response data	Some time-response data; good qualitative description	Some time-response data; good qualitative description	No time-response data
3. Loss of feed-water	Not located	No time-response data	Good qualitative description	Not located
4. Loss of Pressurizer level	Detailed time-response data	Some time-response data	Good qualitative description	Not located
5. Boron dilution	Little time-response data	Little time-response data	Little time-response data	No time-response data
6. Inadvertent safety injection (18 events)	Some time-response data for 17 events	Some time-response data for 12 events (one illegible)	Some time-response data for 12 events	No data extractable

TABLE B.2. Data Availability for Site B (2-Unit BWR) Events

Event	Docket	Reactor Log	Supervisory Log	Special Reports	Computer Output
1. Relief valve stuck open (7 events)	Some time-response data for 3 events	Some time-response data for 5 events; one log not located	Some time-response data for 4 events; two logs not located	Time-response data for 4 events	Not examined
2. Loss of off-site power (2 events)	Not located	Not located	Not located	Some time-response data for both events	Not examined
3. Reactor scram, low level	Not located	Not located	Not located	Some time-response data	Not examined
4. MSIV closure	Not located	Not located	Not located	Some time-response data	Time-response data on one action identified

TABLE B.3. Data Availability for Site C (2-Unit BWR) Events

Event	Docket	Reactor Log	Supervisory Log	Central Desk Log	Special Reports
1. Reactor scram, low reactor level	No time-response data	No time-response data	No time-response data	No time-response data	No time-response data
2. Low flow feed-water line severed	No time-response data	Illegible	Not located	No time-response data	No time-response data
3. Condenser circulating water system rupture	No time-response data	Some data on subsequent action	Good time-response data on many actions	Data on personnel muster	No time-response data
4. Cable-tray fire	Time of discovery only	Good qualitative description of action	Good qualitative description of action	Some time-response data	No time-response data
5. Off-gas explosion	Good time-response data	Good time-response data	Supplements docket and reactor log	No time-response data	None
6. Off-gas explosion	Good time-response data	Some data in addition to docket	No additional data	Some additional data	None
7. Reactor scram, APRM high flux	No time-response data	Not located	Not located	Not located	Not located

TABLE B.4. Data Availability for Site D (1-Unit PWR) Events

Event	Docket	Reactor Log	Special Reports	Computer Output
1. Reactor coolant Leakage	Good time-response data	Good time-response data	Some time-response data	No time-response data
2. Loss of circ. pumps	No useable data	No data of interest	No useable data	No time-response data
3. Loss of AC power	No time-response data	No time-response data	No useable data	None available
4. Loss of RCP	Good time-response data	Some time-response data	No useable data	None available
5. High steam flow trip	No time-response data	Some time-response data	No useable data	None available
6. Loss of AC power	Little useable data	Little time-response data	No useable data	None available
7. Loss of feed pump during startup	Some time-response data	No time-response data	No useable data	None available

TABLE B.5. Data Availability for Site E (3-Unit BWR) Events

Event	Docket	Reactor Log	Shift Engineers Log	Control Room Log
1. Drywell pressurization	Special report detailed chronology	No useable time-response data	Fairly extensive detail on events, but not a single time listed	Not available
2. Off-gas "overpressurization"	Less data than S.E. log; story easier to read	Not available	General time data available; good qualitative description	No useful data; mostly radwaste information
3. Safety valve failure following FW transient	Detailed chronology and follow-up reports	Sparse time-response data	Qualitative description o.k.; no time-response data	Verify time of reactor scram
4. Containment pressurization; MSIV closure	Special report; reasonably good time data	Minimal time-response data	Not available	Minimal time-response data
5. Off-gas explosion	No significant time data; explosion time only	Only time of explosion; no further data	Only time of explosion	Time of explosion only
6. Off-gas explosion	Good qualitative description; no time-response data	Some time-response data	Time-response data on one action	No time-response data
7. Off-gas explosion	Minimal time-response data	Reasonably complete time-response data	Minimal time-response data	No indication event had occurred
8. Drywell pressurization	No time-response data	Minimal time-response data	Not available	No useable data
9. Off-gas "overpressurization"	No useable data	Minimal time-response data	Minimal information	No indication event had occurred

APPENDIX C

PWR Operator Survey Form

SAFETY-RELATED OPERATOR ACTIONS

A SURVEY OF OPERATOR OPINION

CONDUCTED BY OAK RIDGE NATIONAL LABORATORY

____
Oper. No.
(ORNL Use Only)

I. Survey Purpose, Goals and Procedures

This survey is being performed as part of a research program which will help supply data necessary to establish industry (American Nuclear Society) design criteria for automation of safety-related actions. The central question addressed by the criteria is whether certain key safety-related actions should be performed by the licensed operator or whether they should be performed by automatic action, with verification by the operator.

It is also hoped that the research will ultimately help identify and suggest ways to improve any conditions that tend to decrease the likelihood of reliable operator performance.

II. Operator Personal Data (NOTE: All Information (Raw Data) Collected in this Survey is Confidential)

AGE: ______________________

CURRENT POSITION: ______________________

TIME IN CURRENT POSITION (Years): ______________________

PREVIOUS EXPERIENCE (Directly Applicable to Nuclear Power Plant Operation):

Employment: Position | Years Experience

______________________ ______________________

______________________ ______________________

______________________ ______________________

Training: ______________________

____

Education: ______________________

____

TIME SINCE MOST RECENT TRAINING/REQUALIFICATION EXAM: ______________________

____

____
Oper. No.
(ORNL Use Only)

III. General Model of Operator Response

We are assuming that response of an operator following annunciation of an emergency/abnormal event can be generally categorized into four phases as follows:

a. Initial Shock - a very brief period immediately following the event alarm during which the operator is surprised or alerted and is essentially inactivated (except perhaps for acknowledging the alarm).

b. Diagnosis - the operator evaluates information from alarms, annunciators and indicators, identifies what event has occurred, plans his action and initiates required immediate action.

c. Immediate Action - operator carries out immediate actions, both verification of automatic responses and manual actions required to bring the plant to a safe condition.

d. Subsequent Action - operator carries out over a longer time period subsequent actions necessary to maintain the plant in a safe condition, prevent further damage or release of radioactivity, etc.

In your opinion, is this a reasonably accurate general description of operator response? ☐ Yes ☐ No

Please comment if desired: ______________________________

____

____

IV. There are twenty potential emergency/abnormal events listed below. Please circle the numbers of the events you have experienced during plant operation. Then answer the questions A, B, C and D regarding these twenty events.

1. Loss of Feedwater Flow
2. Loss of Condensor Vacuum
3. Emergency Boration
4. Loss All AC Power
5. LOCA
6. Reactor Trip
7. Loss of Condensate/Cond. Booster Pump
8. Fuel Handling Emergency
9. Reactor Coolant Pump Vibration
10. High Coolant Activity
11. Main Steam Line Break
12. Steam Generator Tube Failure
13. Excessive Primary Plant Leakage
14. Reactor Coolant Pump Trip
15. Loss of Component Cooling
16. Continuous Rod Withdrawal
17. Loss of RHR System
18. Safety Injection
19. Rad. Mon. System-High Activity Alarm
20. Loss of Service Water

Oper. No.
(ORNL Use Only)

IV.A. Shock (Psychological Stress)

1. Of the events listed, select three which in your opinion are most likely to result in psychological stress (tension), as evidenced by tightening of stomach muscles, excessive sweating, dryness of mouth, pounding of the heart, or other typical anxiety symptoms.

   Event Numbers: ________ ________ ________

   Please explain briefly what factors (for example, potential harm to yourself or to the public, potential damage to equipment) make these events more stressful than others.

   ____
   ____
   ____

2. Of the events listed, select the three which in your opinion are the least likely to result in stress, and explain why they are not as stressful.

   Event Numbers: ________ ________ ________

   Explanation: ______________________________
   ____
   ____

IV.B. Diagnosis

1. Of the events listed, select three which in your opinion are the most difficult to diagnose.

   Event Numbers: ________ ________ ________

   Please explain briefly what causes the difficulty; for example, symptoms are similar to other events, symptoms are not annunciated quickly or precisely enough, etc.

   Explanation: ______________________________
   ____
   ____

NOTE: Use Reverse Side as Desired for Additional Comments/Explanation.

____
Oper. No.
(ORNL Use Only)

IV.B. Diagnosis (Continued)

2. Of the events listed, select the three which in your opinion are the least difficult to diagnose, and explain what factors make them easier to diagnose.

   Event Numbers: ________ ________ ________

   Explanation: ____

   ____

   ____

IV.C. Time Response

1. Considering both stress level and difficulty of diagnosis, select three of the events listed which you feel would require the most time for you to recover from initial shock, correctly diagnose the event, plan your action and initiate required immediate action.

   Indicate the level of stress (moderate, high or severe) you would anticipate for these events. (NOTE: These three events may or may not be three that were selected in Items A and B.)

   Event Numbers: ________ ________ ________

   Stress Levels: ________ ________ ________

   Estimate the time you think would be adequate for you to respond to one of these events. ____________________________

2. Considering stress level and difficulty, select three of the events listed which you feel would require the least time for you to recover, diagnose, plan and initiate immediate action. Indicate the stress level you would anticipate for these events.

   Event Numbers: ________ ________ ________

   Stress Levels: ________ ________ ________

   Estimate the time you think would be adequate for you to respond to one of these events. ____________________________

NOTE: Use Reverse Side as Desired for Additional Comments/Explanation.

____
Oper. No.
(ORNL Use Only)

IV.C. Time Response (Continued)

3. Consider the three events you have selected as requiring the most time for response. Suppose that a large number of licensed operators experienced these events independently. For each time that is listed, estimate the percentage (%) of operators you think would have completed "response" (that is, recovery from shock, diagnosis, planning, and initiation of required action) by that time or before. Assume that all symptoms are indicated as described in emergency/abnormal procedures and that all safety equipment performs as designed. (NOTE: Each entry should be a cumulative percentage, that is, it should include the total percentage you think would have responded by that time.) Make similar estimates for the events you selected as requiring the least time for response.

	Number of Operators Responding Correctly in Specified Time	
Time	"Most Time" Events	"Least Time" Events
10 sec.		
30 sec.		
1 min.		
2 min.		
5 min.		
10 min.		
20 min.		
30 min.		

V. Specific Events

There are five specific events listed on the following page. If each of these events occurred many times at different plants and with different operators, there would be a "spread" or distribution in response times because of the variability in operators and specific circumstances of the event. For each of the events, estimate the "mean" and "maximum" times to respond. The "mean time" is the time within which you would expect the response to be completed in 50% or more of the occurrences. The "maximum time" is the time in which you feel the response would be completed "9 times out of 10", i.e., in 90% of the occurrences.

Recall that "response time" refers to the time required to recover from initial shock, diagnose the event, plan action and initiate required immediate action.

____
Oper. No.
(ORNL Use Only)

Event	Mean Time (50%)	Maximum Time (90%)
1. Steam-generator tube rupture	________	________
2. Continuous rod withdrawal in auto	________	________
3. Loss of all AC power	________	________
4. Automatic safety injection	________	________
5. High coolant leak	________	________

VI. Discreet Manual Actions

For most of the events described in the emergency/abnormal operating procedures the operator is required to perform one or more manual actions. In our model of operator response we are attempting to separate these discreet manual actions from other tasks such as diagnosis, verification of correct operation of automatic systems, etc. These actions are best explained by example. The actions taken after a LOCA may include tripping the reactor coolant pump for the affected loop or loops. Tripping the pump for one loop is one "discreet manual action". Manual reactor or turbine trip, energizing an electric relief valve, activating a stop valve, or closing a breaker are other examples. As for question V, estimate a "mean" and "maximum" time for completing a discreet manual action such as these.

Mean Time (50%): ________ ____ Maximum Time (90%): ______________

Further Discussion

In the future, it may be desirable to discuss in more detail with station operators specific experiences with abnormal events that have occurred. If duty time were available would you like to participate in such interviews? ☐ Yes ☐ No

If "Yes", please print your name below:

____
(Name)

APPENDIX D

Data Sheets for PWR Operator Response to "Critical Incident" Type Questions

OPER. No.	EVENT NUMBER 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	REASON 1	2	3	4	5
1					X						X					X					X	X	X		
2					X						X								X		X				
3	X			X	X																		X		
4				X	X													X							
5				X	X							X													
6					X						X					X					X				X
7				X	X							X													X
8	X				X						X										X			X	
9	X				X						X										X			X	
10				X	X						X										X				X
11				X	X															X					
12					X						X							X			X		X		
13					X			X			X										X		X		
14					X						X	X													
15				X							X					X									
16				X	X						X														
17				X	X						X										X		X		
18				X	X						X														
19					X						X	X									X				
20	X													X				X			X			X	X
21				X	X						X														
22					X						X	X									X				
23				X	X						X										X				X
24						X				X						X						X		X	
25	X																			X	X	X			
26					X						X	X									X				
27					X			X			X										X				
28					X						X										X				
29				X		X							X										X		
30					X						X	X									X				

Fig. D.1. Data Sheet for Operator Selection of Events Most Likely to Result in Psychological Stress.

POOR ORIGINAL

OPER. No.	EVENT NUMBER																				REASON				
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	1	2	3	4	5
1							X		X										X		X			X	
2	X	X					X																	X	
3							X			X									X						
4		X					X								X										
5										X				X		X									
6							X			X									X					X	
7						X								X				X						X	
8									X	X			X											X	
9									X	X									X						
10													X		X	X						X		X	
11	X						X					X											X	X	
12									X	X								X							
13		X					X												X						
14									X	X						X									X
15															X			X		X					X
16	X					X	X																		
17			X			X			X												X	X		X	
18	X	X	X																						
19						X										X			X						
20		X	X			X																X		X	X
21		X							X																
22								X	X										X						
23								X		X									X						
24					X			X												X	X				
25						X				X									X		X	X			
26						X											X		X		X				
27		X								X									X		X			X	
28		X				X													X			X		X	
29				X		X																			
30		X													X					X				X	

Fig. D.2. Data Sheet for Operator Selection of Events Least Likely to Result in Psychological Stress.

POOR ORIGINAL

OPER. No.	EVENT NUMBER																				REASON				
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	1	2	3	4	5
1											X	X	X									X			
2									X			X	X												
3											X	X	X												
4												X	X												
5											X				X		X						X		
6										X			X					X						X	
7											X	X	X												
8													X		X					X	X				
9					X						X						X								
10					X						X	X										X			X
11								X		X								X			X	X			
12										X		X							X			X		X	
13					X						X	X													
14																									
15										X			X						X						
16					X						X							X							
17					X						X	X									X	X			
18												X	X						X						
19					X						X	X										X			
20											X							X					X		
21											X	X													
22					X			X			X											X			X
23												X	X		X						X	X			
24												X				X							X		
25		X											X						X			X			
26												X	X									X		X	
27												X	X		X								X		
28									X			X													
29					X						X	X										X			
30					X							X	X									X	X		

Fig. D.3. Data Sheet for Operator Selection of Events that are Most Difficult to Diagnose.

POOR ORIGINAL

OPER. No.	EVENT NUMBER 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	REASON 1	2	3	4	5
1						X								X		X					X				
2				X		X											X								
3						X										X		X			X				
4		X					X													X					
5	X							X						X											
6				X	X	X																X			
7	X	X															X								
8						X										X		X			X				
9	X			X		X																			
10	X					X								X							X	X			
11	X					X	X														X	X			
12	X					X										X									
13							X							X		X									
14																									
15						X								X		X									
16	X					X								X											
17	X					X								X											
18	X					X								X											
19	X					X								X											
20	X					X													X		X				
21	X				X	X																			
22			X	X									X								X	X	X		
23	X					X								X							X	X			X
24				X							X			X							X				
25	X											X						X			X	X			
26				X		X										X					X				
27				X	X	X															X				
28				X		X					X														
29				X		X								X							X				
30	X					X								X							X				

Fig. D.4. Data Sheet for Operator Selection of Events Least Difficult to Diagnose.

OPER.	EVENT NUMBER																				TIME	CORREL.	
No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	(SEC)	S	D
1											M	H						M			120	1	2
2				M	H						S										120	1	0
3																							
4					M						M						M				30	1	1
5											S	M					H				60	1	2
6	M			H	H																30-60	1	0
7										X	X	X									<60	1	2
8		H						H			S								M		10-30	1	0
9												H	M				H				60	0	1
10					H						H						M				60	2	2
11				S	S						H										60	2	0
12										M			M						M		—	0	3
13					S						S										120	3	1
14					H						H	H									300	3	0
15				H	H												H				—	2	0
16					X						H							X			60	2	3
17				H	H						M										600	3	2
18					S						H						M				10-15	2	0
19				H											H					H	120	0	0
20				H	H						H										60-120	0	0
21																							
22					H						H	H									5-10	3	2
23				H	M										M						10-30	2	1
24					H			H				H									120-180	2	1
25					S															S	120-300	1	1
26					M						H	M									120-180	3	1
27					H			M			M										60-120	3	0
28					H						H										30-60	2	0
29					H						H	H									<5	0	3
30					H						H	H									5-10	3	2

Fig. D.5. Data Sheet for Operator Selection of Events Requiring the Most Time for Response.

NOTES: M = Event selected was judged as producing "moderate" stress.
H = Event selected was judged as producing "high" stress.
S = Event selected was judged as producing "severe" stress.
X = Event was selected, but no stress level was assigned.

Under the column labeled "CORREL." (correlation), the values under "S" are the number of events selected here which were also selected by this operator as one of the "most stressful" events. The values under "D" are the number of events selected which were also selected as "most difficult to diagnose".

OPER. No.	EVENT NUMBER 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	TIME (SEC)	CORREL. S	D
1							M									H			M		120	2	1
2		M				M	M														–	2	1
3																					–	–	–
4							M									M				M	30	1	2
5	H	M		H																	15	0	1
6						M				M									M		10-15	2	1
7	X					X								X							<30	2	1
8	S				S	S															~0	0	1
9	S				S	H															15-30	0	2
10	M					M				M											30	1	2
11	M						M									M					<10	2	2
12	H				S						S										–	0	1
13		M				H	M														30	2	1
14		M				M	M														180	0	0
15							M								M					M	–	2	0
16	X					X	M														<30	3	2
17	M					M								M							<30	1	3
18	H					S								M							<10	1	3
19	M					M										M					30	2	2
20						M									M					M	<30	1	1
21																					–	–	–
22						M				M				M							3-5	1	1
23	M		M							M									M		2-5	2	1
24						M										M					30	1	0
25			M	M									M								30	0	0
26				M		M										M			M	M	30-60	1	3
27	M																				60	1	0
28																					30-60	–	–
29						M								M		H					<5	1	2
30	M					M										M					2	0	2

Fig. D.6. Data Sheet for Operator Selection of Events Requiring the Least Time for Response.

NOTES: M = Event selected was judged as producing "moderate" stress.
H = Event selected was judged as producing "high" stress.
S = Event selected was judged as producing "severe" stress.
X = Event was selected, but no stress level was assigned.

Under the column labeled "CORREL." (correlation), the values under "S" are the number of events selected here which were also selected by this operator as one of the "least stressful" events. The values under "D" are the number of events selected which were also selected as "least difficult to diagnose".

POOR ORIGINAL

APPENDIX E

Summary of Literature Search for Relevant Sources of Data from Non-Nuclear Industries

by

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Safety Related Operator Actions

A very limited literature regarding operator related responses and safety exists, apparently, at least in the usual library contained sources. Even an extensive 200-item search of computer indexed citations in each of two systems, DIALOG file 7, Social Sisearch and DIALOG file 11, Psych. Abstracts, identified very few documents with contents directly relevant to the areas of operator responses, stress, transfer of training, and safety. The research content reported in those sources identified through computer search, library literature search and documents supplied by ORNL and others is reviewed herein. The complete reference list is appended for use by investigators who may wish to use the materials cited.

Generator Performance Variables

Brigham and Bohr (1978) analyzed the switching errors occurring in five and ten kilovolt electricity distribution systems in a city in West Germany. Two types of error were noted: 1) logical error or memory lapse by the switching engineer in the planning stage, and 2) an unadvertent activation of the wrong switch during execution by the operator. The paper contains a critical discussion of error prevention, possibly of interest to nuclear operator safety response investigators.

In 1975 (Swain & Guttman) an attempt was made to provide reliability estimates for reactor plant operators utilizing known critical incident rates for various stressed stressful situations (i.e., Strategic Air Command in flight emergencies, Army recruits under simulated (but perceived as real) mortar fire). Berkun (1964) provided the basis for the estimates. Error rates for reactor plant operators were then estimated under loss of coolant (LOCA) stress. A model was developed for predicting the rate of human errors and evaluating the degradation of the man-machine system. No real data for operators is included.

A. D. Swain and a group at SANDIA Corporation have produced many reports, papers, and publications on stress, reliability and performance over the past 20 years. Many of these are listed in the reference list but many of the publications and papers have not been available to us. The reviews and summaries we have found suggest that the work of A. D. Swain and his associates has been most valuable but that much remains to be done with the nuclear plant operator domain.

In a study of airborne pilotage error (Roscoe, 1974; see also Kraus and Roscoe, 1972) it as suggested that a blunder occurs because the demands (requirements) for perceptual, judgmental and motor capacities exceed the person's momentary capacity. Using three independent variables: 1) usual type of manual control flown by pilot subjects, 2) storage capacity of the simulated computing system, and 3) level of side-task loading to which pilot subjects were subjected. A four-to-one ratio of residual attention was demonstrated among professional pilots but well designed systems approached freedom from blunder-prone behavior. Pilot's residual attention, as measured by the way subject pilots could cope with information-processing side tasks varied in a sensitive, orderly and statistically reliable manner with each change in equipment characteristics. Such results suggest that pilots (operators) might reasonably be required to demonstrate a specific (minimum level)of blunder-free residual attention before or during training (for certification). This possibility may lend itself to direct evaluation in training module reactor studies.

Laios (1976) used three levels of uncertainty as the independent variable in a decision-performance task (none, medium and high uncertainty). He found that uncertainty decreased decision performance but that there was no difference in decision performance between the two (medium and high) levels of uncertainty. Kennedy, Coulter, and Xenia (1975) used a one-channel or a three-channel vigilance task combined with no threat of shock or threat of shock with students learning to fly. Greater absolute decrement in performance with threat of shock was noted for the three-channel vigilance task than for the one-channel task but the relative decrements were equivalent. Non-stresssed subjects monitored better on one channel than on the three channel task while stressed subjects performed better than non-stressed subjects, with the improvement greater for the three-channel task. From these results it may be suggested cautiously that realistic expectation of shock (danger) heightens performance for active vigilance tasks. Rigby and Edeman (1968) used a questionnaire to gather data to estimate multieng.ne pilot's errors under mild stress and high stress. The low error rate under mild stress was estimated at from .01 to .10 and under urder high stress from .15 to .30. Brigham and Laios (1975) in a study of operator performance in a laboratory process plant found superior performance by subjects who were given intermediate information over those who were given none and those watching automatic controls. The intermediate information subjects developed an "Anticipatory" approach or "control operator system" and reduced errors.

In a study of the effects of threat induced stress on tracking performance, Bergstroem (1970) used army conscripts divided into three matched groups (training 1, 3, 13 hours on a tracking task). Contrary to expectations, under short term stress induced by electric shock, no significant differences were shown by the three groups in task decrements. About 25% of the subjects showed no decrement under shock.

Parasuraman (1976) found two types of ability requirements related to vigilance performance: 1) perceptual speed, and 2) flexibility of closure. Individual differences were highly consistent for subjects requiring the abilities even though the displays were markedly changed but significantly less consistent for tasks classified differently. Perhaps a classification approach for performance on different monitoring tasks is reasonable.

Paternotte's (1978) study of distilling process operators showed that operators controlling the same process differ widely in control goals due to lack of information concerning technical and economic aspects of the process. These findings, based upon an extremely small sample, suggest the need for carefully defined control goals. The operators tend to be conservative, to take "small actions" or tentative actions at first because: 1) the precise effects of the control actions on the output valves are unknown, 2) it was not possible to exercise control action as precisely as desired, 3) lack of immediate, reliable feedback, and 4) the consequences of control action could not be assessed for 40 to 120 minutes.

Haas (1977) noted that high steel construction ironworkers developed a "norm" to refuse to talk about the dangers or risks involved in the occupation. Later, as a working member of the group, he had the opportunity to discuss the deepseated fear and anxieties of the workman regarding "going in the hole" (falling) from 500 feet up due to a mistake by themselves or others. It was clear from his report that anxious and fearful workers were, in turn, feared by others. Although nuclear plant operators are not likely to "go in the hole" the levels or hidden anxiety and stress may be similar.

As shown by the results of a study of experimentally induced stress and performance of power station and distribution operators (Bures and Buresova , 1974) subjects under stressful conditions exhibited one of three behavior patterns: aggression, withdrawal, or reflective responses. These results were obtained through analyses of behavioral observations during a stressed-performance (with levels of training and task modifications) task study.

Although the results and interpretations included above lack any semblance of completeness, it could be suggested that understanding of nuclear plant operator responses, especially with respect to safety responses, can be studied directly through laboratory studies. Design and use of simulation systems wherever stress can be induced and where the complexity of monitoring,decision making and control responses appear possible and necessary. Electric shock has been used with reasonable success and other modes of stress or threat of punishment could be developed. Such equipment could be used for selection, possibly measuring the individual differences noted above as being associated with better or poorer performance. In addition, experienced operators could be interviewed by skilled interviewers (psychologists) for determining something about the norms, perceived stress and danger level and fears and anxieties of nuclear reactor plant operators, both new and experienced. Also, as noted by Finley, Webster and Swain (1974) reduction in human errors in use of equipment can be substantially reduced by using human factors participation in the design phase of equipment, real or simulatory.

Job Analysis and the Nuclear Operator Job(s)

In a study for test validation purposes at the Tennessee Valley Authority (TVA) conducted in 1975 and 1976 by O. Spurlin, R. Ridley and J. Lounsbury, the Position Analysis Questionnaire (PAQ) was utilized for purposes of job analysis. For the purposes of validation of a selection program and training of Student Generating Plant Operators (SGPO) and Assistant Unit Operators (AUO) the PAQ results were analyzed. McCormick (1972, 1973) and his associates have developed 27 dimensions from the PAQ focusing on worker-oriented activities and behaviors. Two PAQ's were completed at each training site. For each position an incumbent AUO filled out the PAQ jointly with the supervisor. The full 27 PAW dimensions were generated for each position. Dimensions with mean scores between 1.5 and 2.5 can be considered dimensions of average importance for the dimension. Dimension scores were obtained by taking a weighted average of the item scores.

"The operator's job is primarily centered around the observation and manipulation of instruments and controls, as indicated by high scores on the general dimensions 1, 2, 5, 9, and 10 . . ." (Spurlin et al, 1976).

Dimension 1	Watching devices/materials for information
Dimension 2	Interpreting what is seen or heard
Dimension 5	Evaluating information from things
Dimension 9	Processing information
Dimension 10	Controlling machine processes

Other high-score dimensions were:

Dimension 3	Using data originating with people
Dimension 8	Making decisions
Dimension 15	Using fingers vs. general body movement
Dimension 19	Contacting supervisor or subordinates
Dimension 25	Being alert to detail/changing conditions

While some of those which might be expected to be higher:

Dimension 22	Being in a hazardous/unpleasant environment
Dimension 23	Engaging in personally demanding situations

were not. However, in the job performance rating scales developed by Spurlin et al (1976), ability to work in hazardous or unpleasant surroundings and ability to work in personally demanding situations were included as Scales 10 and 19.

The characteristics of the sample of SGPO's and AUO's were probably important. Most of the sample personnel were connected with fossil fuel generating stations due to short term experience or emergencies at TVA nuclear stations.

In another study utilizing the PAQ Thompson (1977) attempted to determine the characteristics of the nuclear reactor operator job. Rather than using the conventional dimensions developed by the authors of the PAQ (McCormick, Jeanneret and Mechan 1969, 1972), the Thompson research involved a principal component factor analysis with a varimax rotation (Dixon, 1974) of the responses from 371 nuclear plant operators to the PAQ. Thompson (1977), using his criteria as described in the unpublished dissertation, identified 17 factors, called dimensions by Thompson, see attached copy of Thompson's Table 6). Although the table provides information on the number of items of the PAQ loading on a given factor, no information is supplied regarding the identification of the items.

The 17 dimensions identified by Johnson include one related to possible physical hazards and dangers of the job. However,

> "The idea of permanent physical disability or impairment due to the nature of the job is minimal in the opinion of the operators. The safety record of all aspects of the nuclear industry in one of the best of any in the country." (Thompson, 1977, p. 58.)

However, when Thompson introduces the formulation of a description or summary of the job, the emphasis on the stressful aspects of the operator's job is more evident.

> "A nuclear reactor operator's job is one of extremes. The operator must be able to handle precision instruments and heavy equipment. He must be able to endure long periods of physical inactivity spent monitoring the control room panel and, at a moment's notice, endure intense mental pressure with the realization of possible consequences of malfunction or improper procedures. The operator is one who must have the skills of a manual laborer and the intelligence of a college graduate. A list . . ." (Johnson, 1977, p. 55).

Following through on the description of the 17 dimensions as provided by Johnson, one can "pick out" stressor elements:

Dimension 1	The need to be prepared to handle infrequently performed or irregularly performed nuclear-radioactive related tasks with very small to very heavy equipment.
Dimension 5	Required to estimate speeds of ongoing critical processes (nuclear reactions, timing of reactor periods, startups, shutdowns, power level changes).
Dimension 7	Responsibility for safety of others, etc. general public, and billions of dollars of material things. Errors easily visible.
Dimension 9	Urgent deadlines and precision responses require diligent attention to details.
Dimension 12	Responsible for behavior and responses of others, some as trainees.
Dimension 13	Solving a wide range of intellectual and practical problems.

The fact that the level of stress, danger, consequences of actions and other risks are not clearly identified or emphasized in the two studies cited suggests that efforts to utilize the PAQ to derive a stress index for the nuclear operator job(s) may have a low level of probable utility.

Transfer of Training

Valverde (1973) reviewed the transfer of training effect of flight simulation to flight performance. It was concluded, based upon weak designs, little attention to subject assignment (randomness) and no systematic study of instructors, that simulation aids in some learning. Roscoe (1971) suggested that cost effectiveness of simulators be carefully examined. Blaines, Puig and Regan (1973) also raise questions about the measures of effectiveness of transfer of training from simulators. The best summary is contained in an article by C. O. Hopkins (1976) in which he asks questions about how much one should "pay for that box." He argues that many claims of cost benefits for aircraft simulators are equivocal and that most of the effectiveness of simulation is dependent upon the training procedures used. Other simulation factors, he reports, vary in their demonstrable importance. Cost effectiveness has not been demonstrated for many interesting and attractive features of flight simulators (motion systems, simulator fidelity, etc.).

Summary

Although, and perhaps because, this literature related to the subject areas appears to be so limited, more research is indicated. Time and effort devoted to ferreting out the obscure national and international literature on operator studies can be worthwhile. Many other types of operator jobs, distilling, petroleum cracking, electrical distribution system controllers, Federal Aviation Authority aircraft controllers, chemical plant process operators, sintering plant operators and others can provide useful data from operator error-performance records and through studies. The apparently unique and stressful combination of operation decisions and managerial decisions in the nuclear plant operator's job suggests that, in spite of results of PAQ studies being less useful than anticipated, thorough job analysis, job descriptions and specifications should be developed for the job(s). There is enough information in the fragments in the literature to point to the value of selection of qualified individuals for operator responsibilities.

Although those qualifications are not clearly and completely evident, careful selection studies, after extremely careful job analyses and the gathering of stress data, could increase the quality of operator safety responses under stress.

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