NUREG/CP-0041
Vol. 3

Proceedings of the U.S. Nuclear Regulatory Commission

Tenth Water Reactor Safety Research Information Meeting

Volume 3
- Human Factors Research
- Instrumentation and Control Research
- Occupational Radiation Protection
- Safety/Safeguards Interaction

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 12-15, 1982

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Printed copy price: $8.50

and

National Technical Information Service
Springfield, VA 22161

NUREG/CP-0041
Vol. 3
RH, RS, RV, RX

Proceedings of the U.S. Nuclear Regulatory Commission

Tenth Water Reactor Safety Research Information Meeting

Volume 3
- Human Factors Research
- Instrumentation and Control Research
- Occupational Radiation Protection
- Safety/Safeguards Interaction

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 12-15, 1982

Date Published: January 1983

Compiled by: Stanley A. Szawlewicz, Consultant

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

ABSTRACT

This report is a compilation of papers which were presented at the Tenth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 12-15, 1982. It consists of six volumes. The papers describe recent results and planning of safety research work sponsored by the Office of Nuclear Regulatory Research, NRC. It also includes a number of invited papers on water reactor safety research prepared by the Electric Power Research Institute and various government and industry organizations from Europe and Japan.

PROCEEDINGS OF THE
TENTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

October 12-15, 1982

PUBLISHED IN SIX VOLUMES

GENERAL INDEX

PROCEEDINGS OF THE
TENTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

held at the

NATIONAL BUREAU OF STANDARDS
GAITHERSBURG, MARYLAND

October 12-15, 1982

TABLE OF CONTENTS - VOLUME 3

HUMAN FACTORS RESEARCH (Cont'd)

INSTRUMENTATION AND CONTROL RESEARCH
Chairman: D. L. Basdekas, NRC

OCCUPATIONAL RADIATION PROTECTION
Chairman: A. K. Roecklein, NRC

SAFETY/SAFEGUARDS INTERACTION
Chairman: J. B. Durst, NRC

PREFACE

This report, published in six volumes, includes 160 papers which were presented at the Tenth Water Reactor Safety Research Information Meeting. The papers are printed in the order of their presentation in each session. The titles of the papers and the names of the authors have been updated and may differ from those which appeared in the Final Agenda for this meeting.

Five papers, which were submitted for presentation at the meeting but could not be scheduled, are also included in this report. They are the following:

Calculations of Pressurized Thermal Shock Problems with the SOLA-PTS Method, B. J. Daly, B. A. Kashiwa, and M. D. Torrey, LANL, (Pages 113-130, Volume 2)

Hydrogen Migration Modeling for the EPRI/HEDL Standard Problems, J. R. Travis, LANL, (Pages 131-144, Volume 2)

Independent Code Assessment at BNL in FY 1982, P. Saha, U. S. Rohatgi, J. H. Jo, L. Neymotin, G. Slivik, and C. Yuelys-Miksis, BNL, (Pages 145-168, Volume 2)

Experimental Evidence for the Depencence of Fuel Relocation upon the Maximum Local Power Attained, D. D. Lanning, PNL, (Pages 285-296, Volume 2)

PRA Has Many Faces - Can the Safety Goal Be Well-Posed? H. Bargmann, Swiss Federal Institute for Reactor Research, (Pages 105-114, Volume 6).

HUMAN FACTORS EXPERIMENT SUPPORT[a]

O. R. Meyer
EG&G Idaho, Inc.

Experiments to evaluate human factors in nuclear reactor power plant operations receive engineering support to ensure that the experiment results have significance to safety-related questions. The experiment design is impacted by engineering support which both initiates the experiment (the safety-related question) and defines the form of results that would be significant.

The safety-related question must be defined so as to relate human behavior to reactor safety and, thus, to public risk in observable, objective terms. The human behavior involved in the safety-related question needs to be identified both in terms of the activities (tasks) and the job positions (individual). The emergency response of the main control room (MCR) crew has current high priority because the study of events at reactor plants in recent years, for example, the accident at Three Mile Island, Unit 2 and the steam generator tube rupture at the Ginna plant, has demonstrated a need to understand and predict the impact on reactor safety of human performance during emergency response.

Reactor safety is a technical abstraction in that it is a condition abstracted from a complex aggregation of physical parameters of the reactor facility combined with the probability of future plant conditions and human behaviors. The objectives for maintaining reactor facility safety can be categorized in a progression from normal operation through increasingly severe stages of departure from normal. Finally, the technical abstraction called "reactor safety", itself, must be transformed into its impact upon

a. Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.

public risk. If certain proposals (see Reference 1) for a safety goal definition are followed, the public risk will be quantified in terms of (a) the probability that (b) a defined population group will (c) receive a defined amount of exposure.

Assuming we have defined a significant, safety-related question relevant to the behavior of the MCR crew engaged in the mitigation of an event that could lead to an accident, we can now approach the design of an experiment. The design of the experiment must include the modeling of the complete MCR cognitive system (see Reference 2) which includes the nuclear power plant, the safety parameter display system, and the MCR crew. The experiment designer must determine which of these elements of the model shall be reproduced in his experiment.

Experiments can be defined in which all elements of the MCR cognitive system are present. The most valid case would be to use the reactor facility, itself, of course, but there are serious economic, safety-related, and institutional constraints on the conduct of human factors experiments using the reactor facility. Training simulators that are only one step removed from the reactor facility MCR are being used for human factors experiments (see Reference 3). Training simulators are limited in their capability to reproduce accident behavior, and are heavily booked for their basic mission of operator training. There exists, therefore, the concept of a research facility specifically designed for the conduct of macro-experiments to study the behavior of MCR cognitive systems (see Reference 4).

The functional model of the macro-experiment facility includes a "driver system" which represents the requirements to reproduce information display to the subject(s) and to reproduce plant behavior response to subject's action that is similar to that which would be effected by the actual reactor plant. The driver system would most commonly be some form and degree of plant simulation, but need not be confined to plant simulators.

The macro-experiment facility could also be used for the performance of part-task experiments. It is expected that such experiments would utilize only a part of the facility's capabilities, and would be directed at operator performance of part of the task engaged in by the MCR cognitive system.

The final form of engineering support of human factors experiments is the determination of the ultimate significance of the results of the experiment. As previously stated, the experiment is designed to measure the effect of human behavior upon public risk. If these data are to impact public risk, that is, are to do more than provide confirmation of apparent adequacy, the options for improving the value impact, or the cost-benefit ratio, should be defined in dollars of cost of the option and person-rem of probable exposure associated with the option. Ideally, the optimum mix of options could be then selected to meet a required safety goal as defined above. This ideal is much simpler to state than to attain.

The effect of human behavior in cognitive systems is more complex than the simple classification of "go/no go" used in reliability analysis of equipment or of humans engaged in sensory-motor tasks (see Reference 5). Objective definitions of human behavior that may be useful in risk assessment are being studied at EG&G Idaho.

References

1. United States Nuclear Regulatory Commission, Office of Policy Evaluation, Safety Goals for Nuclear Power Plants: A Discussion Paper, NUREG-0880, February 1982.

2. E. Hollnagel and D. D. Woods, Cognitive Systems Engineering, RISO-M-2330, RISO National Laboratory, Denmark, February 1982.

3. W. Newell et al., Data Collection Plan, Task Analysis of Nuclear Power Plant Control Room Crews, General Physics Corp., July 12, 1982.

4. D. N. Tillitt, R. J. Peterson, R. L. Smith, <u>Performance and Design Requirements for a Graphics Display Research Facility</u>, NUREG/CR-2711, EGG-2194, June 1982.

5. J. Rasmussen, <u>Human Error Analysis and Prediction</u>, Chapter 3, "Human Interaction With Computers," London: Academic Press, 1980.

Human Factors Experiment Support

O.R. Meyer

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

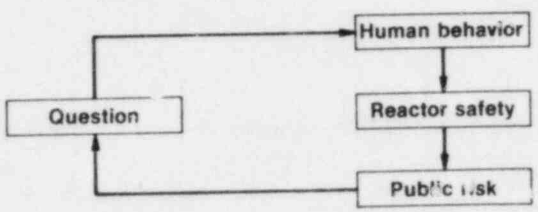
Human Factors Experiment Support

- Definition of safety-related question
- Design of experiment
- Evaluation of significance of results

52 10 226

Human Factors Experiment Support

Definition of Safety-Related Question



52 10 222

Human Factors Experiment Support

Definition of Safety-Related Question

1. Human behavior
   1.1 Activities
      - Normal operation
      - Emergency response
      - Maintenance
      - Management
      - Design, construction, technical support

   1.2 Jobs
      - MCR crew
      - TCS, EOF crew
      - Maintenance
      - Management
      - Engineering and construction

52 10 228

Human Factors Experiment Support
Definition of Safety-Related Question

1. Human behavior
2. Effect upon reactor facility safety (a technical abstraction)
   - 2.1 Containment of radioactive waste
   - 2.2 Prevention of event initiators
   - 2.3 Mitigation of events
   - 2.4 Protection of the public

S2 10 225

Human Factors Experiment Support
Definition of Safety-Related Question

1. Human behavior
2. Effect upon reactor facility safety
3. Effect upon public risk
   - 3.1 Population group
   - 3.2 Probability (frequency per reactor-year)
   - 3.3 Consequence (person-rem of exposure)

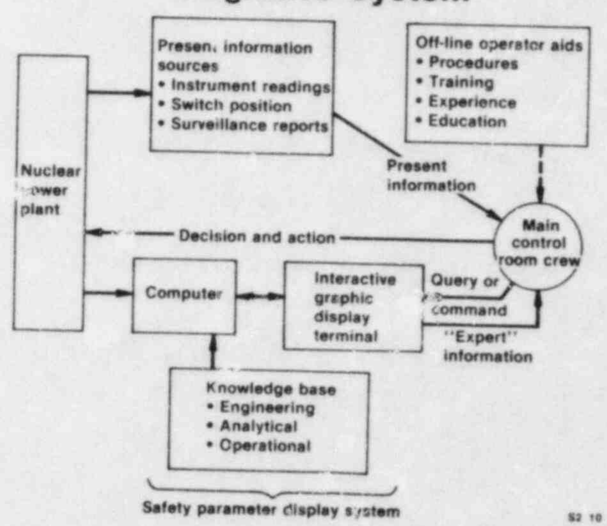
S2 10 227

Human Factors Experiment Support
Experiment Design

1. Subject: Main control room (MCR) crew
2. Activity: Mitigation of events (accident initiators)
3. Modeling: MCR cognitive system

S2 10 224

Modeling of Main Control Room Cognitive System



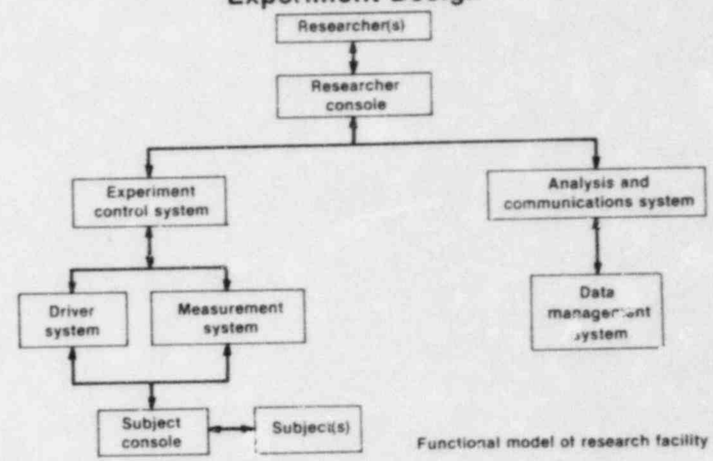
S2 10 221

Human Factors Experiment Support
Experiment Design

1. Subject
2. Activity
3. Modeling
4. Macro-experiments
   4.1 The reactor facility
   4.2 Training simulator
   4.3 Research facility

52 10 229

Human Factors Experiment Support
Experiment Design



Functional model of research facility

52 10 223

Human Factors Experiment Support
Experiment Design

Driver systems

1. Recorded scenarios
2. Specialized (focused) simulation
3. Generic plant simulation
4. Full-scope engineering simulation

52 10 231

Human Factors Experiment Support

Experiment design

1. Subject
2. Activity
3. Modeling
4. Macro-experiments
5. Part-task experiments
   5.1 Detection
   5.2 Recognition
   5.3 Diagnosis
   5.4 Decision-making
   5.5 Feedback

52 10 230

Human Factors Experiment Support

Evaluation of Experiment Results

Human behavior

Reactor safety

Public risk

Options

Ultimate significance of experiment:

- Cost-benefit ratio
- Value impact

S2 3510

Human Factors Research at the OECD Halden Reactor Project

James P. Jenkins
Senior Engineering Psychologist
Human Engineering Section
Human Factors Branch
Division of Facility Operations
Office of Nuclear Regulatory Research

The Halden Reactor Project celebrates 25 years of service to the scientific community in 1983. The Project dates from July 1958 when a BWR, built and owned by the Norwegian Institutt for Atomenergi, became the subject for an international agreement of signatory nations to participate and share in nuclear research. The initial goal was to demonstrate the heavy water moderated reactor concept. The test reactor program has grown from early thermocouple and flow turbine testing into three principal research programs, which are described below:

1. In-core behavior of reactor fuel, particularly reliability and safety aspects, which are studied through irradiation of test fuel elements.
2. Models of fuel and core behavior are developed for prediction, surveillance and control of fuel and core performance.
3. Application of process computers and human factors analysis of power plant control are developed through the use of prototype software systems, display/control hardware and control room manmachine concepts.

Although reports about all the research programs are received by the NRC, this paper will present the major activities concerning the third research area at the Halden Reactor Project.

In the past fiscal year of 1982, the Halden Reactor Project management implemented a series of changes which, when completed in FY 1983, will provide a superior human factors research facility. This facility, called the Halden ManMachine Systems Laboratory or HAMMLAB, is intended to be the main site for systems research on process control and human factors problems. Figure 1 presents the design and laboratory layout used at HAMMLAB. The facility will house the NORS full scale simulator which is a modified copy of the LOVIISA simulator in Finland and will occupy an area of about 3000 sq. ft. (300 m2). Modification to the simulator by Nokia Electronics and the Finnish Technical Research Institute were begun in FY 1982 and presently are about 50% completed. The Finnish utility, IVO, is specifying display formats, operating procedures, and other documentation, for the simulator. This task is more than 90% complete. Other software development is the responsibility of the Project personnel. System testing is planned for a two-month period with a debugged system available in March 1983. A structural overview of program systems and main data areas is shown by Figure 2. The Laboratory's final design was finalized in August 1982 and the design modifications to the existing facilities, and installation of equipment has begun. The layout of physical spaces will be completed by December 1982 and HAMMLAB is scheduled to become operational in the Spring 1983. Notwithstanding the development of HAMMLAB, human factors research activities have been proceeding during FY 1982, and encompasses four projects which are summarized below:

1. First, work on the Handling Alarms with Logic (HALO) system has continued as a part of the plant disturbance handling program. This work is concentrated in two areas: completion of a prototype HALO system for use and experimentation with a small simulator called STUDS and development/ documentation of the HALO system logic. In FY 1982, about 50% of the system was completed. HALO consists of an online alarm processing system which is connected to STUDS and an offline logic generator program. A detailed specification was prepared in FY 1982. Human factors research on the display format is continuing with several display concepts under evaluation. Figure 3 illustrates the primary display which incorporates symbology and intuitive message detection/recognition where alphanumenes will be minimized. Several experiments will be performed to refine and verify the adequacy of these formats.

2. The second project consists of a series of supporting activities and experiments performed with the Finnish utility, IVO, at the LOVIISA simulator. The purpose of the project is to complete development of the Critical Function Monitoring System (CFMS). This multinational research involves not only personnel from the Halden Reactor Project, human factors staff, but also scientists from Combustion Engineering, IVO and the Finnish Technical Research Institute, (VTT). In FY 1982, the CFMS software was mated to hardware at LOVIISA. Twelve crews of operators were trained on the alarm analysis system. Validation experiments began in September and are scheduled to conclude in December 1982. The experiments produced over 200 megabytes of raw data. Additional software to reduce this volume remains to be specified and developed. The Halden staff has developed a methodology called ATEA to provide the theoretical basis for the experimental design, variable definition and selection, hypothesis testing, statistical treatment and data analysis. The experimental plan provides for two different transients of about 30 minutes each to be presented in the simulator. Six of the 12 crews will run one of the transients before receiving training on CFMS and the other six after receiving training. Data will be collected on status of safety parameters, alarm status and operator activities.

3. The third project is a series of human factors experiments involving different display concepts, software, and formats. The overall objective is to define unique display characteristics which influence operator (individual and crew) performance in a control room. Several of these studies are described below:

a. One of the studies analyzes the performance of a two man crew in identifying key information characteristics which must be shared to successfully manage an event. Two operators, each unable to see each other's displays, must coordinate their activities for a successful trial. Different display formats and protocols will be evaluated.

b. Another study involves multinational development of a concept called a "man-machine systems generator". The concept includes a high-level language,(PICASSO), which is user-friendly, intuitive, and self-documenting. PICASSO is the language for the generation of graphic displays, especially a mimic diagram format. It is expected that by the end of 1983 the development of the system will include a compiler, a linker and associated picture libraries and possibly an interactive editor.

4. The fourth project was the Workshop on Human Factors Experiments and Validation of Operator Aids. The Workshop was held at the Halden Reactor Project in March, 1982. A follow on is scheduled for Loen, Norway in May 1983. The March 1983 Workshop was intended to provide the signatory representatives forum to discuss the human factors plans and present their own concerns, plans and requirements. The three principal topics were: first, Halden Reactor Project scientists described their concepts of experimental validation and the design requirements for such experiments; second,the 13 signatory representatives provided the results of their experiences and activities in validation; and third, the Project personnel described HAMMLAB and the display projects which are summarized in this paper.

FIGURE 1 LABORATORY LAYOUT OF HAMMLAB

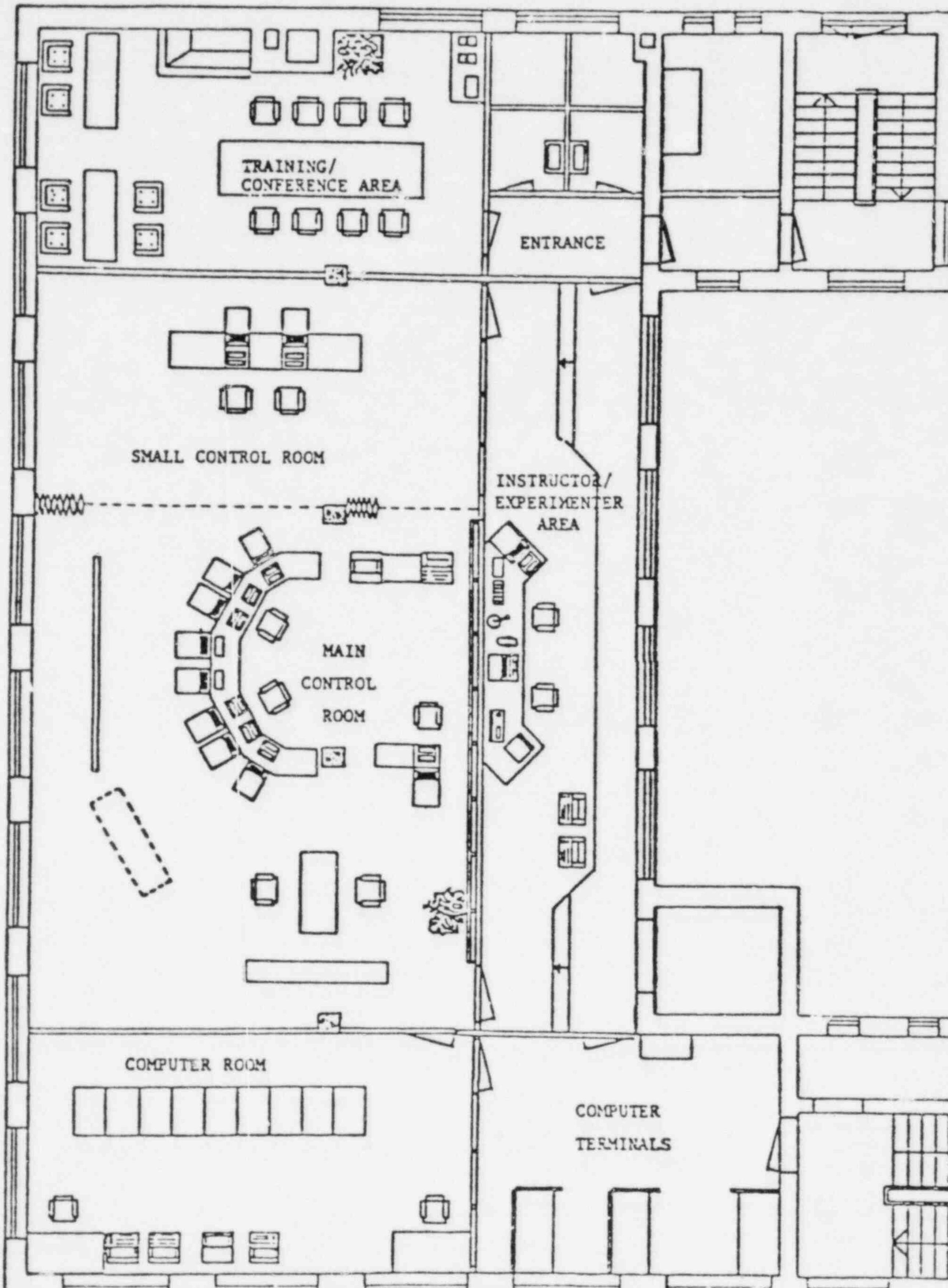
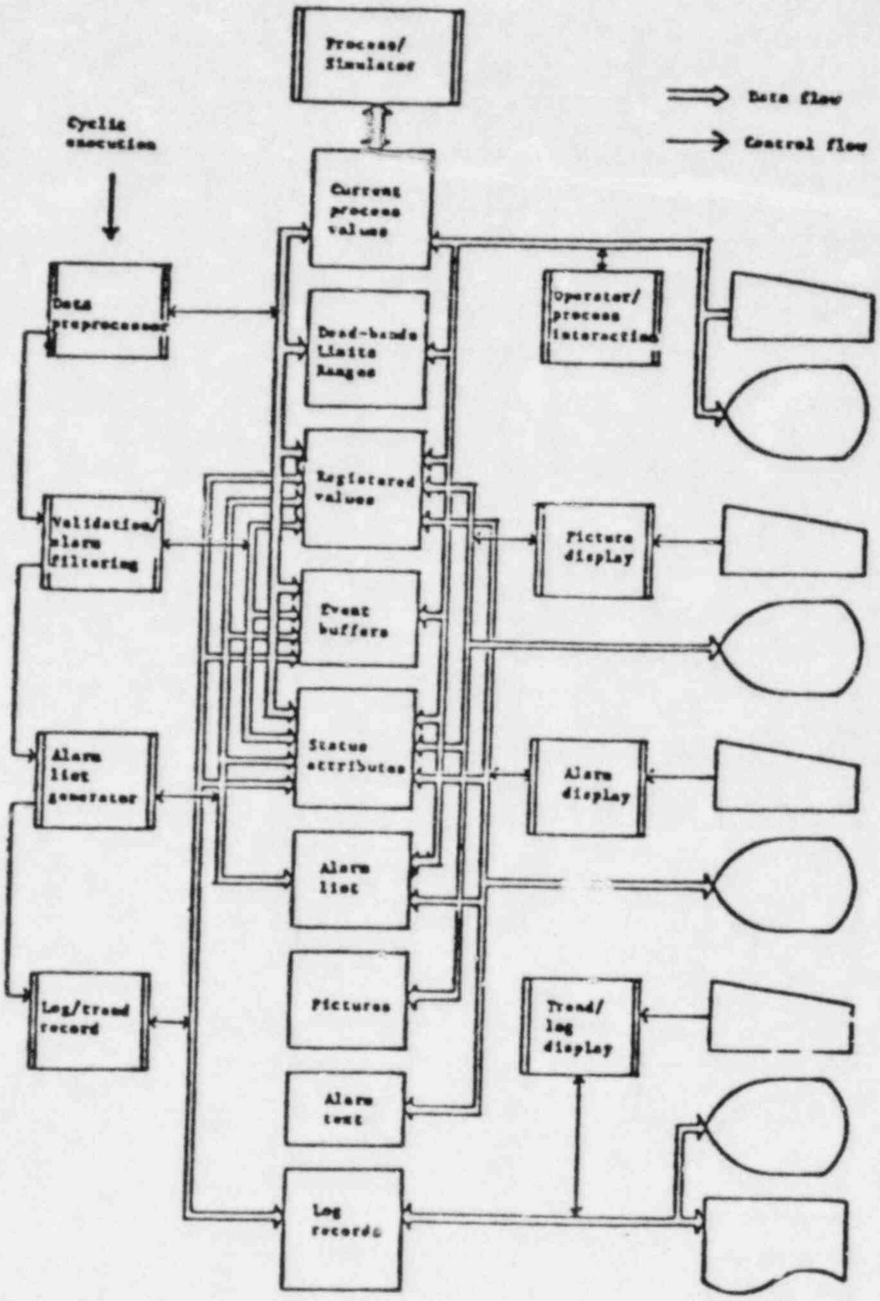


FIGURE 2 STRUCTURAL OVERVIEW OF PROGRAM SYSTEMS AND MAIN DATA AREA OF HAMMLAB.



A MULTIMETHODS APPROACH TO SAFETY PARAMETER DISPLAY EVALUATION

W. W. Banks, H. S. Blackman
D. I. Gertman, R. J. Petersen

EG&G Idaho, Inc.

The Human Factors Engineering Office of EG&G Idaho performed this NRC-funded study to assist the NRC in objectively assessing licensee-developed safety parameter display (SPD) formats and designs. The purpose of this study was to quantitatively measure the degree to which a tachistoscopic method of display evaluation would correlate with the results of a multidimensional rating approach to display evaluation. The ultimate goal was to identify the method which accounts for the greatest amount of operator performance, yet costs the least amount of money. Results of the following three experiments will be presented; (a) tachistoscopic, (b) multidimensional rating scale, and (c) the combined results of a and b.

The test material for all experiments consisted of three multivariate data display formats all under development as SPDs for reactor control rooms presenting safety parameter display data at the loss-of-fluid test (LOFT) facility. The three display formats studied were stars, deviation bar graphs, and meters. Three questions were posed: (a) What is the degree of concurrence between these two independent methods used in display evaluation? (b) Can one of the two methods be used successfully to predict results of the other? (c) What dimensions of SPD formats appear to be most crucial to operators for performance and preference?

Eighteen adult volunteers were used as subjects. Their ages ranged from 26 to 44 years and all reported vision correctable to 20/20. All were currently qualified reactor operators from the LOFT reactor plant, with a mean of 9.4 years reactor operating experience.

Tachistoscope Method. A dual-channel tachistoscope (t-scope) was used to study the three display formats. The classic model of signal detection was employed collecting data for perceptual sensitivity, response criterion, percent correct, and reaction time. Two studies were made: signal detection and parameter recognition. The signal detection study found differences for display type, exposure duration, and interactions. For the dependent variables of perceptual sensitivity, percent correct, and response criterion, stars were significantly greater than the combination of meters and bars, and stars were significantly greater than bars. The interactions of display and exposure duration also showed a superior performance for the star display, but only with the short exposure as the difference diminished with increasing exposure durations. Recognition study results revealed no significant effects or interactions from any of the analyses.

Multidimensional Rating Scale Method. The authors used a combination of factor-analytic and forced-choice techniques to develop six scales for evaluating display interfaces: content density, content integration, format, cognitive fidelity, cognitive processing, and general acceptance. The study sought to determine if this multidimensional rating scale (MDRS) methodology would apply to the evaluation of the three display formats.

Statistically significant results were obtained only for content integration (CI) and cognitive processing (CP). In both cases, the order of preference from most to least preferred was bars, star, and meters. Orthogonal planned comparisons showed that bars and star differed significantly from meters for CP only ($p < 0.05$); no other comparisons reached significance.

Combined Results. To answer the three major questions posed as objectives for this paper, forward stepwise multiple regression analyses were conducted. Two sets of analyses were run combining the data from the MDRS study with the performance data from the recognition and detection studies. Multiple regressions were run, with the performance data from the recognition and detection studies serving as dependent variables (d, beta, percent correct, and reaction time) and the scores from the six subscales plus a total score from the MDRS providing the predictor or independent variables. Multiple regressions were run for each dependent variable against subscale scores (predictors) for each display type and when collapsed across display type.

Discussion. The results of the multiple regressions demonstrated that statistically significant relationships do exist between the performance measures of the tachistoscopic method and the MDRS. The MDRS can reliably predict between 11 and 67% of the variability in the t-scope measures of performance. Thus, the two methods do converge.

When the MDRS subscales were considered in isolation, collapsing across the dependent measures, and display type, it was found that FO (format) and CD (content density) each appeared nine times in the multiple regressions, indicating that these subscales are most critical in predicting performance. CF (cognitive fidelity) and CI (content integration) were the second most frequent and therefore salient in predicting performance, each occurring five times. GA (general acceptance) and sum (the total instrument score) appeared least frequently, (four and two times respectively). It is important to note that all six subscale scores and the total score were critical in prediction for the various multiple regressions. The researchers would also expect the critical subscales to change, dependent on display type and performance measure.

Three other findings of interest resulted from the multiple regression analysis: the dependent measures of d, Beta, and percent correct from the detection study were negatively correlated with the MDRS; the dependent measures of d, Beta, percent correct, and response time from the recognition study were positively correlated with the MDRS subscale scores and total score. The detection study only produced significant R^2 results with the star display, and the recognition study produced significant multiple regression results with the bar and meter displays.

To understand these results, it is necessary to consider the methodologies used in the detection and recognition studies, and of course the displays themselves. The detection study methodology sought to discriminate between displays based solely on abnormal parameter perception; a low level phenomena when considering human memory and learning. On the other hand, the recognition study sought to discriminate between displays based on abnormal parameter recognition; a much higher-level cognitive process. Also, since the MDRS asked the subject to rate the displays on content density, integration of content, organization of format, clarity, ease of processing, and aid to decision making, the subjects evidently rated the displays not based on purely perceptual aspects, but on how well, in their opinion, the display presented information for ease and accuracy of use. These postulates must be coupled with the fact that stars achieved significance for the detection study whereas bars and meters achieved significance only on the recognition study. The star display was unfamiliar to the operators and they did not have time to become so familiar with the display that they could accurately predict their performance with a recognition task using the MDRS. Thus, the relationship between ratings and actual performance is attenuated. For bars and meters, however, operators can predict performance because they have experience with these formats. The detection study collected lower-level cognitive data not being directly assessed by the MDRS and somewhat different than what the operator would normally consider in answering the general question of how well the display presents information for ease and accuracy of use. Thus the bar and meter displays did not give significant results with the detection data and the MDRS; however, since the star display was unfamiliar, the operators responded to the MDRS in a manner different than that for either bars or meters, thereby causing the purely perceptual performance data of the detection study to be predicted by the subscale scores of the MDRS.

The major conclusion is that one can predict the type of performance data yielded by the t-scope studies using the MDRS. It is also true that the t-scope adds a unique portion of explained variance not covered by the MDRS. The MDRS is sensitive to differences in operator familiarity with the display and predicts different levels of cognitive functioning commensurate with the operator's prior knowledge. Research is currently being conducted to include checklist and simulation evaluation techniques in this multimethods approach to further identify and validate possible means of display evaluation.

A Multimethod Approach to Safety Parameter Display Evaluation

W.W. Banks
H.S. Blackman
D.I. Gertman
R.J. Petersen

EG&G Idaho, Inc.

Overview

- Experimental questions
- Methods
- Results
- Conclusions

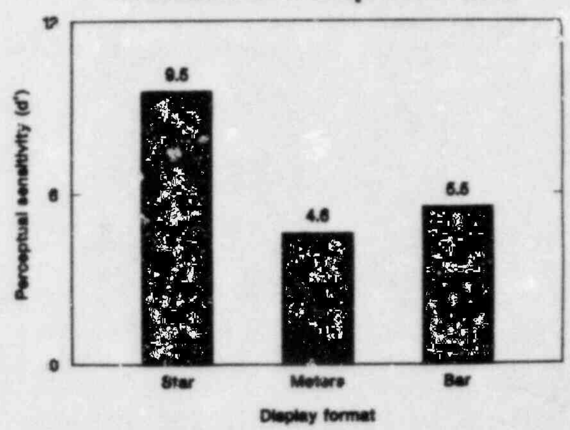
C2 0646

Experimental Questions Posed

- What is the degree of convergence between these two independent methods used in display evaluation?
- Can one of the two methods be used successfully to predict results of the other?
- What dimensions of SPD formats appear to be most crucial to operators for performance and preference?

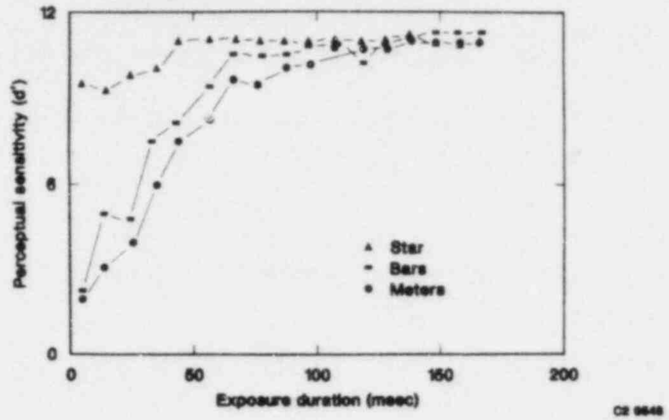
C2 0636

Detection Experiment

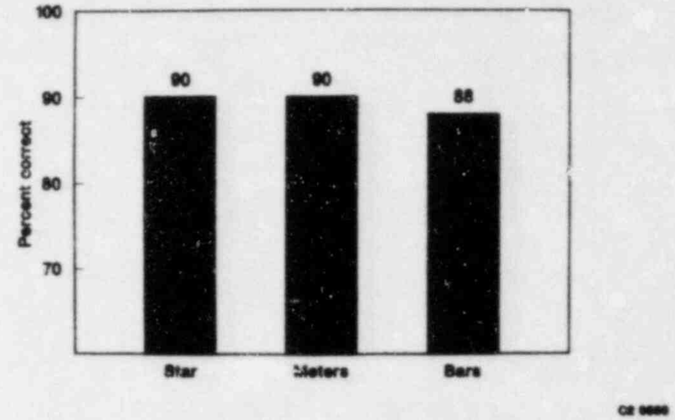


C2 0649

Detection Experiment



Parameter Recognition Experiment



20

Cognitive Dimensions

- Content Density (CD)
- Content Integration (CI)
- Format (FO)
- Cognitive Fidelity (CF)
- Cognitive Processing (CP)
- General Acceptance (GA)

C2 0373

1. Main effect for variable CI across display type.

df	MS	MS_{error}	F	Sig
2.51	1.675	.427	3.917	<.05

2. Main effect for variable CP across display type.

df	MS	MS_{error}	F	Sig
2.51	1.520	.407	3.728	<.05

C2 0376

Multiple Regression Analyses

- For each dependent variable against subscale scores in four cases.

    1. Collapsed across display type
    2. Bar displays only
    3. Meter displays only
    4. Star displays only

C2 0644

Statistically Significant R^2s

d′

	Star	Bar	Meter	Collapsed
Detection	0.44			0.11
Recognition		0.47	0.59	

B

	Star	Bar	Meter	Collapsed
Detection	-0.23			
Recognition			0.65	

C2 0646

Statistically Significant R^2s (cont'd)

Percent Correct

	Star	Bar	Meter	Collapsed
Detection	0.56			0.14
Recognition		0.67	0.44	

Reaction Time

	Star	Bar	Meter	Collapsed
Detection				
Recognition		0.50		0.16

C2 0647

Major Results

- Statistically significant relationships exist
- Measures are reliable
- Can predict up to 67% of the variability
- The tachistoscopic and MDRS methods converge
- Most salient dimensions in order of importance are
    - FO (format) and CD (content density)
    - CF (cognitive fidelity) and CI (content integration)
    - GA (general acceptance) and sum (total instrument score)

C2 0643

Findings of Interest

- The dependent measures of d', beta, and percent correct from the detection study were negatively correlated with the MDRS.
- The dependent measures of d', beta, percent correct, and response time from the recognition study were positively correlated with the MDRS subscale scores and total score.
- The detection study results only produced significant R^2 with the star display, and the recognition study results produced significant multiple regressions with the bar and meter displays.

C2 0642

Major Conclusions

- One can predict t-scope performance data using the MDRS
- MDRS is sensitive to differences in operator familiarity with displays
- Further work utilizing checklist and simulation techniques is warranted

C2 0641

Allocation of Functions to Man and Machine in the Automated Control Room

R. Pulliam *
H.E. Price *

This morning two speakers (J.P. Jenkins, W.W. Banks) have already emphasized the importance of human factors to the design of information displays. Later on, Mr. R. Kisner will talk about models of man-machine interaction. We shall not need to convince this audience of the importance of human factors in control room design. It is sufficient to say that we are all concerned about the role of man in nuclear power plant (NPP) control, especially as we approach higher levels of automation.

In fact, we expect automated control to produce a dramatic change in the role of the NPP operator. We hope and believe that this change will be for the better; in fact we believe that automation may provide the best hope for mastering the complexity of NPP control, and may permit the design of control systems which are at the same time safer, more efficient, and better suited to the characteristics of man.

The rush to automation has not always fulfilled these optimistic hopes. In fact, it is almost a rule that automated systems are in some ways not as satisfactory as the manual systems they replace. Nearly every new system creates some new workload as it alleviates others. Nearly every new system generates new causes of error. Automated systems are often considered "less friendly" by operators, and we constantly see instances in which the users resist automation, seek to override the new system, or continue to do by hand what the system was designed to do.

---

* BioTechnology, Inc., Falls Church, VA

Other speakers have suggested some of the causes, and their possible solutions. But we will emphasize that one of the principal causes of bad man/machine design is the fact that usually there has been no deliberate consideration of which tasks and decisions are best performed by man, and which by machine.

It is natural for systems designers to exploit automation to the limits of affordable technology. The question of <u>what</u> should be automated is seldom asked. Design decisions are usually made on engineering grounds alone, and are soon firmly cast into hardware and software, after which they permanently limit the flexibility of the human role. When functions are automated, the human operators may be unable to monitor events or to exercise useful control. On the other hand, when functions are delegated to man the users may be required to perform unnecessary chores or to do tasks for which humans are poorly adapted. To a large extent this failure to appropriately allocate tasks occurs because there is no established procedure for making these decisions during system design.

This problem is recognized by NRC and by many professionals in the NPP community. Our company, BioTechnology, Inc. (BTI), is conducting a project sponsored by NRC through the Oak Ridge National Laboratory, which hopes to provide NPP designers with a practical method for allocation of functions, either as a step in the design process or for evaluating the man/machine allocations in an existing control room design.

An Historical Study

The project we are reporting began about two years ago. At that time, BTI undertook a study for the Department of Defense in which we examined the R&D literature and the histories of recent systems procurements. In spite of DOD regulations which specifically require allocation of functions as a step in the design cycle, we did not find a single case in which the allocation of functions was determined on a system-wide basis, in an orderly way. This is true in spite of the fact that several methodologic models had previously been developed and were available to guide the allocation of functions. Many obstacles are responsible, but central among them is the absence of an accepted general method and a professional tradition for the allocation of functions. Accordingly, BTI recommended the development of a practical framework and a set of methodologic tools which a design team could use in allocating functions.

Developing a Method

In 1981, BTI began developing such a methodology for the nuclear power industry. In an effort sponsored by ORNL, we initially developed a conceptual method for allocating functions (or assessing existing allocations) in NPP control rooms. The method is applicable both to earlier technology using electromechanical process control and to later technology exploiting the computer.

BTI first examined the history of control technology, and then reviewed major models and methods which have been proposed for the allocation of functions. These begin with the "listing" approach. In 1951, Fitts proposed a table listing the differing capabilities of machines and man, to be used in support of decisions about automation. Since then, more elaborate lists have been put forward, for instance by Mertes and Jenny (1974), Edwards and Lees (1974), and Swain (1980). More elaborate simulations, procedural guides, and information support systems have also been developed, including HEFAM (Connelly & Willis, 1969), CAFES (Parks & Springer, 1967), SYSSIM (Ireland, n.d.), SAINT (Workman et al., 1975), HOS (Strieb & Wherry, 1979), and the Hypothetical-Deductive Model of Price and Tabachnik (1968). Several of these have features which might be applied in determining functions for nuclear power plant control, but most of them either were never developed in an operational form, or were predicated on the availability of large bodies of human reference data which do not yet exist. Thus, in spite of widespread concern over this problem, there appears to be no instance of a proven methodology for allocating functions to man or machine.

Findings of this preliminary research included a recommended rule-based, iterative procedure for allocating functions in the design of NPP control rooms, and some "lessons learned":

- There has been no successful system-wide use of an allocation method.
- Most proposed or demonstrated methods for allocating functions are helpful for psychomotor tasks, but not for the cognitive tasks which are central to NPP control operations.
- Allocation of functions is like engineering design: it is an iterative process that requires repeated cycles of preliminary design, test, and modification.
- Allocation decisions drive related requirements for training, procedure writing, and personnel selection.
- A major need in automated systems is for man-computer communications--that is, for a means by which (1) the operators can be aware of system states even when computers exercise control, and (2) the computer logic can be informed of human interventions and the purpose of those interventions.

- Engineering design depends on an institutional memory, within the profession, of past successes and failures. We need such a memory to facilitate allocation (and for other human factors) decisions. This may have been the key finding at this stage of the research: No allocation procedure will work without the creative, informed judgment of experienced professionals.
- Several methodologies, recommended in the past for allocating functions, were flawed by a simplistic assumption, as I will explain next.

The Two-Variable Decision Model

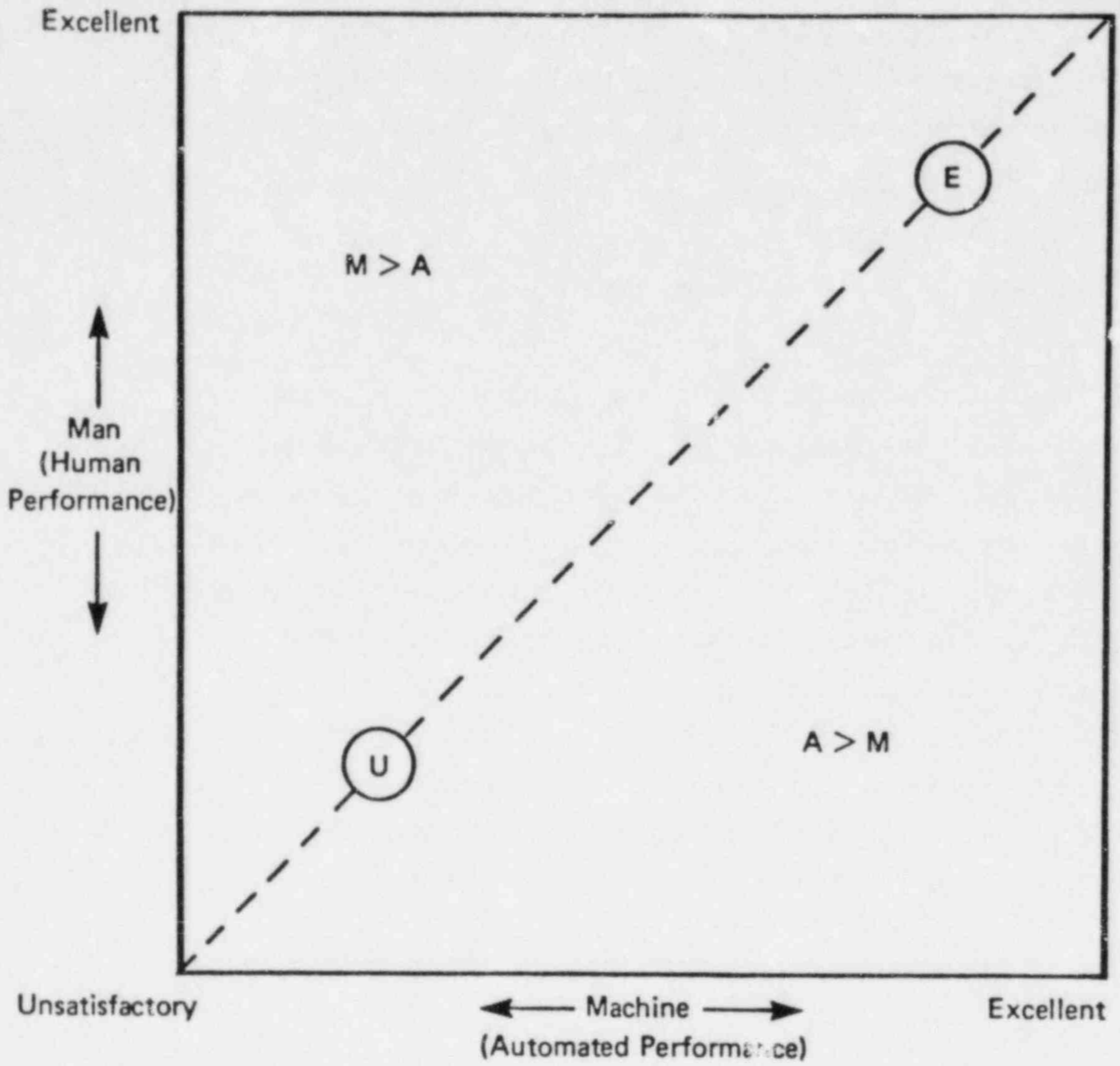
The question of whether a designated function will be better performed by man or by machine has sometimes been viewed as a single-dimension question. It was assumed that if man performs a task poorly, a machine will necessarily perform it well. This is obviously not the case; there are tasks, such as low-speed sorting of objects by color, which both men and machines perform very well, and other tasks, such as multi-variate value weighing, for which neither men nor machines are well suited. In fact each allocation decision requires two separate assessments, one concerning the effectiveness of man and one concerning that of machine.

The relationship between these two assessments can be illustrated by a two-dimensional decision space, in which any task or function is represented by a point. We will examine first the general characteristics of the decision space (Exhibit 1) and then a specific decision matrix (Exhibit 2) which can be drawn within that space.

Exhibit 1 represents the decision space concerned, which is defined by two dimensions. The vertical (X) dimension

Exhibit 1

Decision Space for Relative Control Performance of Man and Machine



represents the relative effectiveness of man, scaled from "unsatisfactory" at the bottom to "excellent" at the top, and the horizontal (Y) dimension scaled left to right represents the corresponding effectiveness of a machine. The X and Y values of a point in that space will represent the estimated probable effectiveness with which men or machines, respectively, can perform a specified task or function, and the position of that point will prove useful as a means of deciding how the function should be allocated.

At a gross level, this decision space can be divided into two areas by the diagonal line U-E, representing the values of $\frac{X}{Y} = 1$. Any point in the upper left area now represents a function which is best suited to man, and any point in the lower right area, a function best suited to a machine. But this distinction alone is not a basis for an allocation decision, since special conditions exist at several points. At the lower left, for instance, in the area marked U, are tasks which are not performed well by either man or machine. Such tasks may be actually infeasible, or impossible to achieve safely. During the early days of flying, a point in this area would have described the function of piloting an aircraft. By contrast, at the upper right corner near E is an area in which all functions are performed so well, by either man or machine, that the allocation decision is largely a matter of free choice. In fact, any function defined by a point close to the diagonal line U-E is one for which man and machine are equally well (or equally poorly) suited. An allocation for these functions can be based principally on criteria other than the relative suitability of men and machines, viewed as engineering components.

We can redraw the decision space of Exhibit 1 as a decision matrix (Exhibit 2) in which there are five differentiated regions. The decision strategy which is appropriate to apply in allocating functions is significantly different for each region.

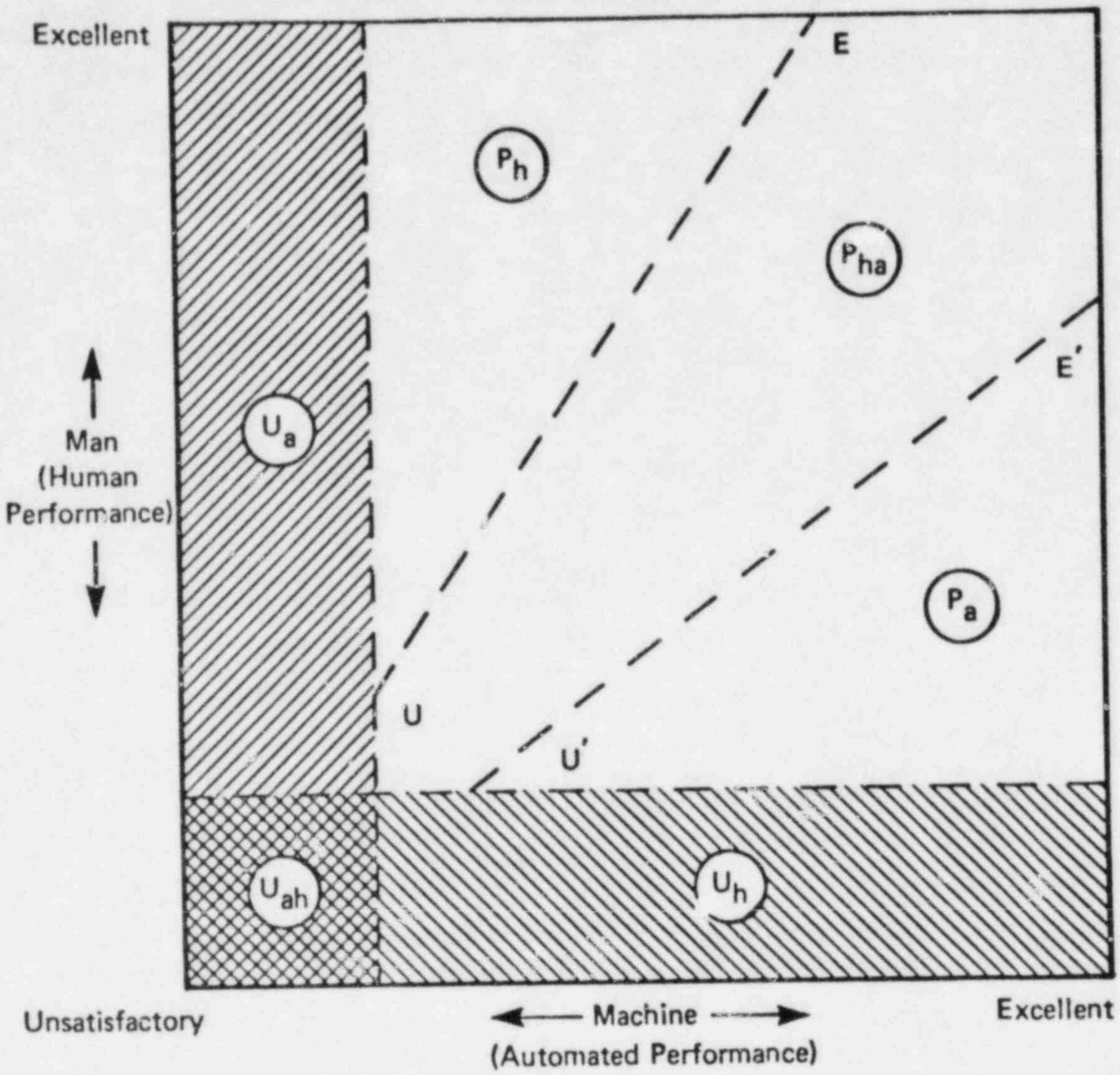
The matrix includes two regions shown as shaded, U_a (unacceptable: automation), and U_h (unacceptable: human). Functions falling in region U_a are too low on the "machine performance" scale to be considered for automation; they can presumably be allocated to man by default. Conversely, in region U_h , any allocation will presumably be to machine. However, at the intersection of U_a and U_h is the region U_{ah} , where both men and machines perform unacceptably. This corresponds to the area U in the earlier figure. Any function which falls in this region should be considered for redesign, or included in a system only as a final resort.

Other regions in the matrix, not shaded, represent functions which might be acceptably performed by either man or machine, with varying degrees of advantage. In the region P_h (preferred: human) man is expected to be substantially superior as a control component. Functions in this region will be allocated to man, in the absence of other overriding considerations. Conversely, in the region P_a (preferred: automation), allocation will ordinarily be to machine.

Finally, there is the region P_{ah} , which is bounded by regions U_a , U_h , and by the lines of constant proportional difference U-E and U'-E'. At all points in this region the difference between the expected performance of man and machine is not great. This is a region of less certain choice, so far as the relative control performance of man and machine is concerned. In this region the allocation decision can be based

Exhibit 2

Decision Matrix for Allocation of Functions



on considerations other than the engineering performance of man and machine as control components. These considerations include costs, worker preferences, and the availability of proven design experience.

The matrix of Exhibit 2 should be used during system design to evaluate the merits of man or machine as control components. Any function planned for a system should be evaluated for its estimated values of expected man/machine suitability, and recognized as belonging to the decision class which the matrix suggests. Note that no numerical values are suggested for the boundaries within the matrix. Both of the man and machine performance variables (X and Y dimensions) are themselves multivariate parameters which resist quantification. In the absence of an ability to scale X and Y, no reasonable values can be assigned to the internal boundaries of the matrix. Furthermore, it must be recognized that the matrix deals only with the question of which allocation is preferred from the engineering component point of view. The decision rules suggested by the matrix may be overruled by considerations other than the relative effectiveness of man or machine, viewed as control system components only. This may happen, for instance, for reasons of cost, legal restrictions, worker preferences, or a technologic inability to construct a system using the ideal allocation.

BTI's Recommended Procedure: Constructing an Hypothesis

After establishing the conceptual framework, BTI proceeded to (1) elaborate a practical, step-by-step, reproducible method by which allocations can be made, and to (2) identify criteria sets to be used in applying the method. This method is now being fully developed and will be applied to a selected real case in the NPP industry.

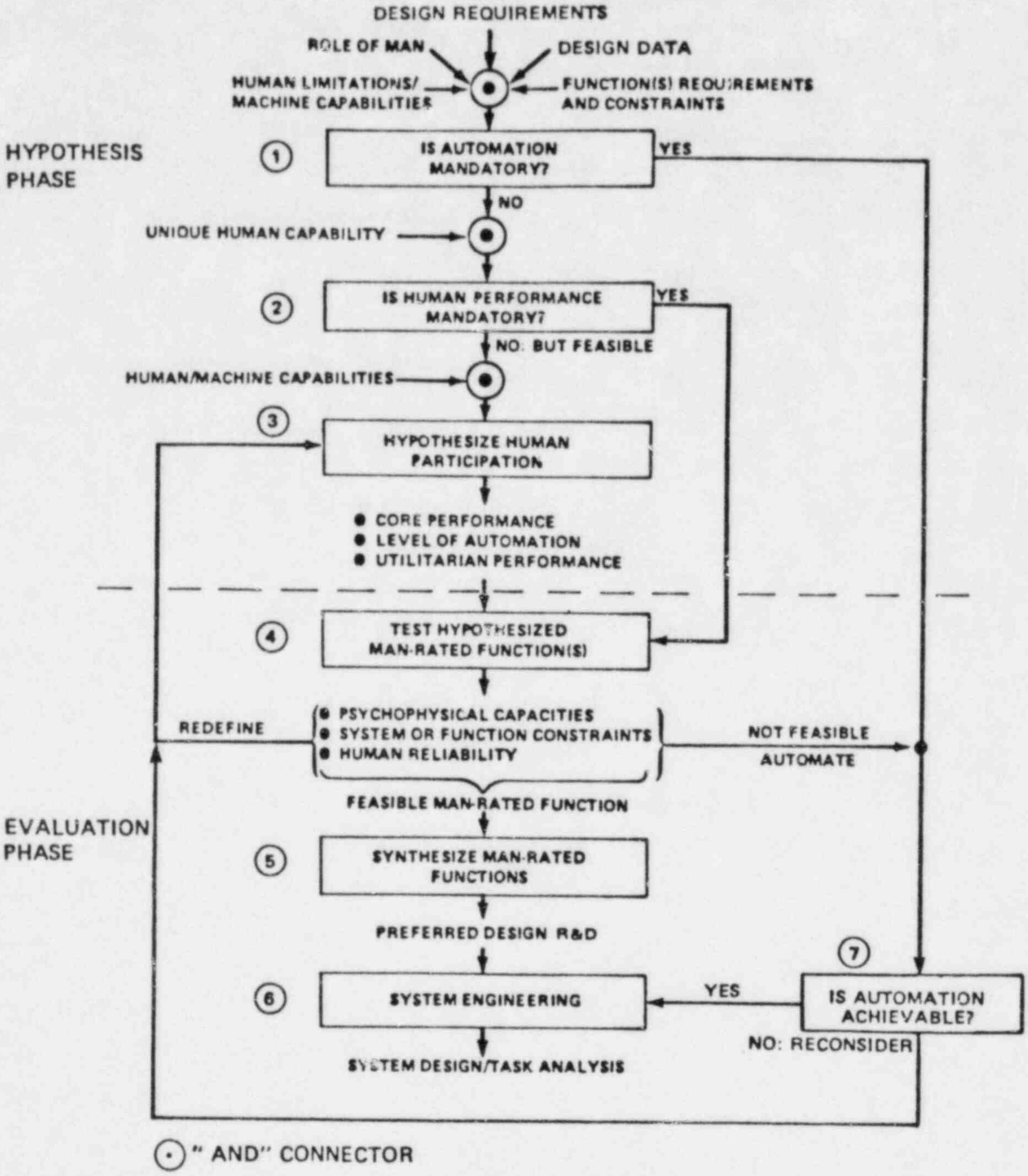
The procedure differs from earlier schemes in at least one more feature: earlier procedures provided hypothetical or procedural solutions only. However sound they were, they provided only an untested hypothesis as to the correct allocation solution. The BTI procedure has added deductive (or empirical) tests of the hypothesized solution. Furthermore, these specific tests are to be followed by corrective feedback, so that the method can search heuristically toward an optimized man-machine interaction. The method is designed to be applied continuously, throughout the system design process, and to produce a series of increasingly accurate approximations to the objectives expressed in a system requirements statement.

Exhibit 3 illustrates principal steps of the proposed method. Note the median dashed line, which separates an initial hypothetical analysis from the following evaluation phase. This second phase is called the "deductive" phase when deductive rather than empirical tests are employed, as must be the case during early (concept or preliminary) design phases.

In the procedure, initial steps identify those functions which must be allocated to either man or machine for obvious reasons. These allocations fall in regions U_a or U_h of the matrix (Exhibit 2). Such allocations must be made to automation (Step 1) for instance, when regulation or policy requires it, when hostile environments preclude the presence of man, or when the required system reaction times exceed human response limitations. Allocations to human control (Step 2) may be mandatory when, for instance, there is a requirement to develop strategies, to detect patterns or trends, or when meaning or values must be assigned to events. Additional tests are applied for economic and technical feasibility (Step 3), and in some cases a tentative design decision may have to be fed back for reconsideration at the system requirements level.

Exhibit 3

The Allocation-of-Functions Process



Steps (1) and (2) are repeated first at the whole-system level, then for subsystems, and finally for portions of subsystems until those parts of the system which clearly must be controlled by man or computer have been partitioned off and allocated properly. This will normally leave substantial portions of the system, and of the operating procedure, which can reasonably be allocated either to man, to machine, or to some combination of the two. At Step (3), these functions are classified according to a performance taxonomy and allocated on a best-choice basis, using regions P_a, P_h, and P_{ah} of the matrix.

BTI's Recommended Procedure: Evaluation

At this point in each cycle of the system design, an allocation of functions to man or machine has been hypothesized. In a design which has reached the mockup or prototype phase, an empirical test is appropriate. But a set of deductive tests is provided as well, which can be used during concept formulations and other early design phases.

First (Step 4 in Exhibit 1), those functions hypothesized as "man-rated" are reviewed in detail against the known psychophysical capabilities of man, against system constraints, and against reliability requirements. If found feasible in these tests, a next step (Step 5) asks whether the human job, as it is emerging, is acceptable to an operator. Modifications are made at this point to ensure that operators will feel supported and important, that the job is coherent, and that it will fit into a reasonable authority and social structure. Finally, depending on outcomes of tests (Steps 4 and 5), elements of a preferred man-machine design are provided to systems engineering (Step 6) or are fed back to other steps of the design process.

The seven-step process just described is reported in detail in NUREG/CR-2623 (Price, Maisano, & Van Cott, 1982). At each point in this process decision aids are provided, but the actual decisions remain judgmental. It is suggested that the procedure be applied by a team including at least one experienced human factors engineer and one control engineer. The method provides an orderly decision procedure and a set of decision aids which include some representative quantified human performance data. More importantly, the procedure provides for documentation of the decision process. This documentation, one of the "conditions" discussed earlier, makes it possible for allocation decisions to be communicated widely within the systems design organization. It provides a basis for iterative improvement and elaboration of detail in the man-machine relationship, and for interaction with engineering design decisions as the system design evolves. Finally, it serves to expand the institutional memory.

Allocation as a Process in System Design

The steps of the allocation process provide an orderly decision sequence for allocating functions. But more important than any particular sequence of steps is the commitment to a deliberate allocation process. In the past, when designers have achieved good man-machine designs, that success has been largely due to an intuitive consideration of human factors. Intelligent designers have tried to foresee how users would interact with the machine. Often they have had experience as hands-on users, and they have considered human factors problems and solutions without consciously practicing that science. Unfortunately, the more frequent case has been one in which engineering designers have pursued an engineering solution, without any consideration, deliberate or otherwise, of what should be automated. The recommended procedure provides a methodology for the deliberate consideration of man-computer roles, as part of the systems design process.

We assume that the allocation of functions must eventually become a formal step in control system design. That step will cost professional effort; it will "interfere" with engineering decisions, and will require additional paperwork within the design documentation system. Is it worth the cost? The answer is clear: Human error now causes about half of the accidents leading to a release of radiation, and from 20 to 50 percent of reported plant failures are due to human error (Sugarman, 1979). In those cases where it is the equipment that fails, it is human action which must minimize the consequences and bring the plant under control. The human element is the more complex part of the man-machine symbiosis, yet we now invest less than 10 percent of the design effort in consideration of the human role. Not only are greater efforts justified, but the allocation of functions is only one step in what should be a more general investment in human factors engineering during design.

When systems were relatively simple, intuitive judgments about the human role were more likely to be right, and the general adaptability of human beings could be depended upon to make up for some inadequacies of human engineering design. As systems become more complex, this is no longer true, and it will be increasingly necessary to invest in human factors analysis as an integral part of the design process.

References

Connolly, M.N., & Willis, J.E. The development of a human effectiveness function allocation methodology (HEFAM) (Research Memorandum SRM 70-11). San Diego, CA: Naval Personnel and Training Research Laboratory, October 1969.

Edwards, E., & Lees, F.P. The influence of the process characteristics on the role of the human operator in process control. *Applied Ergonomics*, 1974, *5*(1), 21-25.

EPRI/ESSEX Corporation. Integrating human factors engineering into nuclear power plant design (NP-1637). Alexandria, VA: ESSEX Corporation, Draft, June 1982.

Fitts, P.M. (Ed.). Human engineering for an effective air-navigation and traffic-control system. Columbus, OH: Ohio State University Research Foundation, March 1951.

Ireland, F.H. SYSSIM--A system simulation program for use on a time-shared computer. Moorestown, NJ: Systems Engineering Missile and Surface Radar Division.

Mertes, F., & Jenney, L. Automation applications in an advanced air traffic management system--Volume III: Methodology for man-machine task allocation (Report No. DOT-TSC-OST-74-14-III). McLean, VA: TRW Incorporated, August 1974.

Parks, D.L., & Springer, W.E. Human factors engineering analytic process definition and criterion development for CAFES (Report No. D180-18750-1). Seattle, WA: Boeing Aerospace Company, Research & Engineering Division, January 1976.

Price, H.E., & Tabachnick, B.J. A descriptive model for determining optimal human performance in systems--Volume III: An approach for determining the optimal role of man and allocation of functions in an aerospace system (including technical data appendices) (NASA CR-878). Chatsworth, CA: Serendipity Associates, March 1968.

Price, H.E., Maisano, R.E., & Van Cott, H.P. The allocation of functions in man-machine systems: A perspective and literature review (NRC NUREG CR-2623). Oak Ridge, KY: Oak Ridge National Laboratories, June 1981.

Strieb, M.I., & Wherry, R.J., Jr. An introduction to the Human Operator Simulator (Technical Report 1400.02-D). Willow Grove, PA: Analytics, December 1979.

Sugarman, R. Nuclear power and the public's risk: Human factors, safety problems, and the complex art of risk assessment. *IEEE Spectrum*, November 1979, *16*(11), 59.

Swain, A.D., & Guttmann, H.E. Handbook fo human reliability analysis with emphasis on nuclear power plant applications (NRC NUREG CR-1278). Albuquerque, NM: Sandia Laboratories, April 1980.

Wortman, D.B., Sigal, C.E., Pritsker, A.B., & Seifert, D.J. SAINT II documentation manual (Report No. AMRL-HESS-74-4). West Lafayette, IN: Pritsker & Associates, Inc., December 1975.

Review of Operational Aids for Nuclear Plant Operators*

(Summary of Paper to be presented at
Tenth Water Reactor Safety Research Information Meeting
October 12-15, 1982, Gaithersburg, MD)

R. A. Kisner

Oak Ridge National Laboratory

INTRODUCTION

Many approaches are being explored to improve the safety of nuclear plant operations. One approach is to supply high-quality, relevant information by means of computer-based diagnostic systems to assist plant operators in performing their operational and safety-related roles. Privately and federally funded research has resulted in the development of operational aid concepts to improve plant monitoring, diagnostic and corrective capabilities, and operator-process communication. Many of these concepts have passed from the idea stage to the point of testing.

The evaluation of operational aids to ensure safe plant operations is a necessary function of NRC. However, such evaluation is made difficult by the lack of reliable quantitative performance measures and function analysis data. This lack is a result of the nuclear power industry not having adopted a rigorous systems approach as characterized by the aerospace/aircraft industry. As a result, to obtain these data for design use requires post-engineering synthesis, that is, reconstruction of the original design process.

Furthermore, a situation the reverse of the systems approach has evolved: many operational aid systems are being developed without adequate analysis of the operator's role, system function, and operator tasks. This is analogous to having solutions in search of problems. Analyses, would help point to specific functions and tasks for which the operator may require assistance, especially those in the areas of information processing and problem solving.

This work has two purposes: to collect limited data on a diversity of operational aids, and to provide a method for evaluating the safety implications of the functions of proposed operational aids. After a discussion of the method evaluation now under study, this paper will outline this data collection to date.

---

*Research supported by the U. S. Nuclear Regulatory Commission under contract with Oak Ridge National Laboratory, operated by Union Carbide Corporation for the U. S. Department of Energy.

METHOD

Several alternative approaches can be employed in the evaluation of potential functions of operational aids:[1]

1. No review - No prior NRC approval is necessary. The supposition is that any responsible effort to implement an operational aid represents a net improvement in operational safety.

2. No adverse effects - NRC approval is necessary before testing and operation of the operational aid. The aid is considered a nonsafety-related system. It must not be required for safety, and failures must not significantly affect the ability of plant safety systems to function as required or cause plant conditions more severe than without the aid.

3. Improved safety - Prior NRC approval is necessary before testing and operation of the operational aid. The supposition is that an improvement in safety is required and that the licensee must demonstrate that the aid represents such an improvement. Satisfaction of specific critieria is required.

A general list of functions that can be identified either to improve or adversely affect safety could be useful to support these approaches, especially the latter two. Several sources exist for deducing or generating a list of tasks and functions that could benefit from operational aids:

1. Operator workload timelines. Tasks and functions that contribute to workload peaks and overload.

2. Operator error analysis. Error-prone functions and tasks.

3. Operator emergency response models.[2] General functions to achieve safety goals during an emergency.

4. Operator function classification.[2] The overall functions of an operating crew derived from a context-free taxonomy.

A possible methodology for evaluation of an individual aid is (1) compare its functions with those determined to be effectual, ineffectual, or hazardous (this is a theoretical verification of the efficacy of specific functions), and (2) test its ability to implement its specified functions at a simulator or other facility (this is an experimental verification)[1].

DATA

Information about specific operational aids under development by various groups is incomplete and has been difficult to obtain. To enlarge and improve the data base, a questionnaire was prepared and used to canvass a limited number of organizations. The questionnaire included the following categories:

1. Problem definition

2. Function

   2.1 Role/User
   2.2 Memory
   2.3 Control

3. Design

   3.1 Scheme
   3.2 Computer hardware
   3.3 Computer software
   3.4 Verification
   3.5 Standards

4. Plant Interface and Environment

   4.1 Isolation
   4.2 Installation

5. Performance

   5.1 Reliability/Availability
   5.2 Response time
   5.3 Input data verification

6. Operation

   6.1 Interface
   6.2 Interaction
   6.3 Responsibility of operation
   6.4 Crew verification of system response
   6.5 Workload
   6.6 Communication

7. Maintenance and Testing

   7.1 Requirements
   7.2 Responsible organizations and duties
   7.3 Methods used to verify accomplishment
   7.4 High maintenance components
   7.5 Self-testing and on-line diagnostics

8. User Training
   8.1 Addition training needed
   8.2 Extent of knowledge of system needed
   8.3 Use of system during training
   8.4 Future users

9. Documentation

   9.1 User control of documentation
   9.2 Currency
   9.3 Availability
   9.4 Perspective

10. Work Status

   10.1 Current
   10.2 Expected operation

Responses varying widely in detail have been received from 14 organizations. Their salient features will be presented in the full paper.

CONCLUSIONS

Without a clear description of the functions and tasks of operations personnel, it is difficult to determine how best to provide them with computer-based assistance. The evaluation of computer-based aids developed on the basis of partial knowledge is equally difficult. Nevertheless, such systems are being developed and their effectiveness and safety value must be assessed. This can be done to a limited extend by the methods described.

REFERENCES

1. R. DiSalvo, "Regulatory Approaches to Disturbance Analysis and Surveillance Systems," February 23, 1981, internal report, Office of RES, U. S. NRC.

2. R. A. Kisner and P. R. Frey, "Functions and Operations of Nuclear Power Plant Crews," NUREG/CR-2587, April 1982.

TASK ANALYSIS OF NUCLEAR POWER PLANT CONTROL ROOM CREWS[1]

Donald C. Burgy
General Physics Corporation
Columbia, Maryland

A task analysis of nuclear power plant control room crews is being performed by General Physics for the Office of Nuclear Regulatory Research. The objective of the project is to conduct a crew task analysis which will provide data for evaluating six areas:

- Human engineering design of control rooms and retrofitting of current control rooms
- The numbers and types of control room operators required with requisite skills and knowledge
- Operator qualification and training requirements
- Normal, off-normal, and emergency operating procedures
- Job performance aids
- Communications

The task analysis methodology employed in the project offers an effective approach to collecting information which describes power plant crew composition, their activities and their environment. The task data collected on the crews will provide a firm technical basis for the development of guidelines and regulations directed at improved operational safety.

There are five key factors of the overall program approach that should be considered when reviewing the methodology or comparing the results of this

---

[1]Presentation to the Light Water Reactor Safety Conference, Gaithersburg, Maryland on October 12, 1982. This work was supported under contract NRC-04-82-005 from the U.S. Nuclear Regulatory Commission to General Physics Corporation. In addition, subcontractors to General Physics for the project include BioTechnology, Incorporated of Falls Church and AMAF, of Columbia, Maryland.

project with other task analyses which may be performed on the same operator population. These factors include:

- Data collection will be done in a field environment - Task data will be collected by project teams at power plant sites while observing the activities of the control room crew. Subjects will be personnel experienced in plant operations performing tasks as specified by company procedures and operating practices.

- The control room crew will be observed responding to plant events - Activities will be observed within a context termed an operating sequence which has an identifiable beginning and end point. A condition will be hypothesized which will require that the crew perform a series of tasks.

- The focus of the project will be limited to the power plant control room - The nuclear power plant control room (or a power plant control room simulator) will be the setting for all observational exercises. The actions of plant equipment operators or others which occur outside the control room during operating sequences will not be observed. However, information on communications between control room crews and personnel in remote locations will be included.

- The activities of control room crew members will be sampled across a range of conditions - A small percentage of all possible power plants and operating sequences will be observed. Statistical validity is not a criteria for selection. Instead, variables will be sampled across a range of conditions in order to verify the completeness of the task analysis methodology and data structure.

- A dynamic data base will be developed - Data collected in the field will be processed and stored within a computer-searchable data base. The data base will be designed to incorporate other task analysis data required in the future. Data will represent the situation as observed in the field and the data base will allow updating should key influences on operator actions, such as operating procedures or control room design, change in the future.

In order to assure adequate consideration of all the factors in the data collection approach, a program plan and data collection plan were prepared at the front-end of the 1[illegible]-month project.

The Program Plan[2] for the crew task analysis described the historical perspective of task analysis, the new task analysis approach for this project

---

[2]Program Plan: Task Analysis of Nuclear Power Plant Control Room Crews. Prepared for the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C., March 9, 1982.

and the relation of this project to other job and task analyses being conducted in the nuclear power industry to date. In addition, the plan delineated the power plant sites to be visited, the operating sequences to be selected, and the crew members to be observed. The plan also presented the initial concept of the computerized data base.

Historically, task analysis has consisted of a set of sequenced tasks whose completion leads to the initiation of a subsequent task. The analysis was single-thread in nature since there was little if any interaction or branching in the successive tasks being performed. In the crew task analysis project, a large number of tasks which could be linked together in a number of ways was expected. Each operator has different tasks that could vary in number, manner of linking, sequencing and branching. The variation is due to the type of plant (e.g., BWR or PWR), the vintage and architect-engineer, and the operating modes of each plant. Thus, a multi-thread task analysis approach was called for as well as a means for sorting and retrieving the multivariate task data from each plant and operating sequence.

Of major concern in the project was the relation of the crew task analysis approach to other ongoing industry efforts in job and task analysis. During the planning phase, selected project staff participated in the Institute of Nuclear Power Operations (INPO) job/task analysis effort. The participation allowed a detailed understanding of INPO's project objectives and approach in order to assure compatibility of the resultant data from the two projects. Additional input to the project concerning methodological issues was gained through a pilot task analysis[3] performed by General Physics for Oak Ridge National Laboratory. The pilot study provided valuable insight into data collection methods available and allowed the crew task analysis project to refine and modify the techniques and approach already tested in a control room/simulator environment.

---

[3]Barks, D. B., Kozinsky, E. J., and Eckel, S. Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors (NUREG/CR-2598). Prepared for Oak Ridge National Laboratory, May 1982.

The Data Collection Plan[4] for the crew task analysis was prepared following a peer review and NRC review of the Program Plan. Detailed data collection procedures and demonstrations of suitability for analysis were contained in the Data Collection Plan.

The data collection approach was divided into three phases:

- Phase A: Table-top Task Analysis. Prior to each power plant visit, the data collection team will spend 4-6 weeks in preparation. Each operating sequence will be analyzed to develop a preliminary description of task content. Plant documentation and subject matter expert opinion are the data sources.
- Phase B: On-site Data Collection. Plant visits will be of one or two weeks duration at each of eight plants. The preliminary descriptions of task performance will be refined and verified through a sequence of plant operations expert reviews and walk-through/talk-throughs of the operating sequences in the control room or simulator.
- Phase C: Data Entry. Following completion of the site visit, descriptive task data will be refined further through the last in a series of quality control checks and submitted to data base specialists for entry into the computerized task data base.

Detailed procedures for acquisition of task data in each phase were written to assure consistent quality in the final data. Data collection forms that were initally developed in the Program Plan were revised and currently include: (1) an Operating Sequence Overview, (2) a Task Sequence Chart and (3) a Task Data Form.

The detailed descriptions of each operating sequence chosen are contained in the first form - Operating Sequence Overview (OSO). In a brief narrative format, crew actions and plant functions are described. Specifically, the initial plant conditions, sequence initiator, expected progression of action, final plant conditions and major systems involved are noted. The OSO establishes the operational context in which all task data is described.

---

[4] Data Collection Plan: Task Analysis of Nuclear Power Plant Control Room Crews. Prepared for Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C., July 12, 1982.

An initial description of the tasks in the operating sequence are contained in the second form - Task Sequence Chart (TSC). The form is designed to show the sequence of tasks and their corresponding cues within each operating sequence. Information presented for each task includes the task statement, task purpose, initiating cue for the task and the plant system involved in task performance.

The primary data collection record for crew task performance is contained in the third form - Task Data Form (TDF). The form is designed to capture task descriptive data, i.e., that data which characterizes a task and can be verified by observation of dynamic crew performance. The TDF is presented in Figure 1 and is structured in three sections called plant identification, task identification and description of task action. The information included in the first two sections comes from the previous data forms described above. In the third section, task action is broken into a "model sentence" format which details:

- Who performs the behavioral element
- Location where action is performed
- Discrete operator action (behavior)
- Component, parameter, state of system involved
- Means of action
- Communication required by task

For most of the data fields, there is a list of standard terms that can be used (see Figure 2). Some fields are free-format (i.e., no standard terms) to allow for variation among plant specific nomenclature. Through the use of a standard vocabulary, the crew task analysis should not suffer from the limitations sometimes realized in previous task analysis efforts. In addition, the standard vocabulary allows for easier quality control efforts during all phases of data collection.

The second section of the Data Collection plan described a demonstration analysis plan. It featured a discussion of the approach to demonstrating suitability of the data for analysis. A requirement of the project was to show how the task data is applicable through examples to the six objective areas stated above. In the analysis plan, an example for suitability for each objective area was identified based on hypothetical questions asked by

FIGURE 1

TASK DATA FORM (DESCRIPTIVE)

Page No. ______

PLANT IDENTIFICATION	TASK IDENTIFICATION	
Plant Name ______	Operating Sequence ______	Task Statement ______
Unit Number ______	erating Sequence ID ______ A-C ID ______	Task Purpose ______
NSSS Vendor ______	Operator Function ______	INPO Task Code ______
A-E ______	Operator Sub-function ______	Task Sequence No. ______
TG Vendor ______	Comments ______	Task Duration ______
CR Type ______	______	Procedures ______
OL Date ______	______	______
______	CUE ______	Data Collected at: ______

Who Takes Action JOBCAT	Location of Means of Action LOC	Behavior		Object of Action						Means of Action MEANS	Communication Link		
		TIME	VERB	COMPONENT	PARAMETER	STATE	OTHER OBJECT	PLANT SYSTEM	INPO EQUIV		RESPOND.	RLOC	CONTENT

FIGURE 2

STANDARD DATA BASE ENTRIES ("TASK DATA FORM DESCRIPTIVE")

<table>
<tr><th colspan="14">DESCRIPTION OF TASK ACTION</th></tr>
<tr><th rowspan="2">Who Takes Action
JOBCAT</th><th rowspan="2">Location of Means of Action
LOC</th><th colspan="2">Behavior</th><th colspan="6">Object Action</th><th rowspan="2">Means of Action
MEANS</th><th colspan="3">Communication Link</th></tr>
<tr><th>TIME</th><th>VERB</th><th>COMPONENT</th><th>PARAMETER</th><th>STATE</th><th>OTHER OBJECT</th><th>PLANT SYSTEM</th><th>INPO EQUIVALENT</th><th>RESPONDENT</th><th>RLOC</th><th>CONTENT</th></tr>
<tr>
<td>Choose one of the following:
RO-1
RO-2
RO-3
SRO-1
SRO-2
(Shift Supervisor)
SS
(Shift Superintendent)
STA
(Shift Technical Advisor)
SM
(Station (Plant Manager)</td>
<td>Place where object of action is located (Panel Number or Other Workstation Identifier and Panel Name)</td>
<td>The elapsed time in the sequence at which an element is initiated by a member of the crew.

Not pre-filled. Determined on-site. Where multiple elements are concurrent, assign them all the same start time.</td>
<td>Verb representing crew behavior. From modified Berliner classification.
Inspects
Observes
Reads
Monitors
Scans
Detects
Identifies
Locates

Interpolates
Verifies
Remembers
Calculates
Chooses
Compares
Plans
Decides
Diagnoses

Moves
Holds
Pushes/Pulls
Positions
Adjusts
Types

Answers
Informs
Requests
Records
Directs
Receives

Attaches
Removes</td>
<td>Choose one word from one of the following groups:
<u>Mechanical Group</u>
Accumulators
Blowers/Fans
Boilers
Compressors
Condenser
Containment
Damper
Penetrations
Control Rods
Demineralizers
Filters/Strainers
Heat Exchangers
Heaters
Main Steam Line
Pumps
Reactor
Stack
Sump
Tanks
Turbine
Valve Operators
Valves
<u>Electrical Group</u>
Batteries
Circuit Closure Devices
Circuit Breakers
Switchgear
Electrical Conductors
Electrical Distribution Buses
Generators
Turbine (GT)
Diesel
Interlock
Motors
Rectifiers
Transformers
<u>Instrumentation & Controls Equipment</u>
Computer
Sensors/Control Instruments:
Controller
Master Controller
Sensors</td>
<td>Choose one word from one of the following groups:
<u>Mechanical Group</u>
Differential Pressure
Differential Temperature
Flow
Humidity
Level
Position
Pressure
Speed
Temperature
Time
Vacuum
Vibration
<u>Electrical Group</u>
Current
Frequency
Power
Resistance
VARS
Voltage
<u>Radiation Group</u>
Dose Rate
Radioactivity
<u>Reactor Group</u>
Delta flux
Neutron flux
Startup Rate
<u>Chemistry Group</u>
Concentration of Gas
Concentration of Solids</td>
<td>On/off
In/out
Open/close
Fast/slow
Increasing
Decreasing
Steady
Automatic/Manual
Setpoint (Below/above)
Input signal
Output signal
Operability
Reset
Silence
Trend
Test
Start
Stop
Zero
On-scale
Off-scale
In-band
Out-band
Above/below limit
Value
Dilute
Borate
Lockout
Trip
System Activation
System Isolation</td>
<td>Open – will usually be a procedure or job performance aid (JPA)</td>
<td>Generally from TSC

Local Plant Nomenclature.

From Plant Procedures & Panel Nomenclature.</td>
<td>INPO Generic Equivalent.

From translation list written prior to plant visit and used during pre-fill of Task Data Form.</td>
<td>This is the operator-plant interface—a control, display, or others.
<u>Control Group</u>
Continuously variable controls
Discrete controls
<u>Display Group</u>
Annunciators
Indicator lights
Meters
Meters/recorders
Computer
Recorders
Counters
Digital Displays
CRTs
<u>Other Group</u>
Verbal
Printed material
Safety equipment
Communication equipment:
-Conventional powered telephone system (CPTS)
-Sound powered telephone system (SPTS)
-Walkie-talkie radio transcievers (WTRT)
-Page party announcing system (PPAS)</td>
<td>Job category of person communicated with:
RO-1
RO-2
RO-3
SRO-1
SRO-2 (Shift Supervisor)
SS (Shift Superintendent)
STA (Shift Technical Advisor)
SM (Station/Plant Manager)
Load Dispatcher
Health Physics Team
Health Physics Section
PEO (Plant Equipment Operator)
NRC-HQ Bethesda
NRC-Regional Office
County Civil Defense
Local PD
Sheriff
State Energy Management Agency
State Nuclear Energy Division
Plant Security (PS)
Maintenance Section
Chemistry Section
CR Crew
On-Site Personnel
Non-specific caller</td>
<td>Location of party communicated with:
On-site
Control-room
Off-site</td>
<td>Open-Ended</td>
</tr>
</table>

For Parameter it is often necessary to identify the associated component.

Model Sentence – Description of Task Action:

In order to *(the reason why the task is being performed (1))* subject *(the individual who performs the action(s) of the task/subtask (2))* at *(panel or other workstation ID) (3))* performs the following task behavior(s) *(verb), (4)*, addressing *(object or behavior (5))*, by means of *(source of information, mechanism of action.*

potential user groups (a list of questions is contained in the Program Plan). The demonstration examples will show how the user searches and retrieves task descriptive data from the data base, analyzes the data through an accepted methodology or process, and draws conclusions related to the hypothetical question asked of the data base. For some examples, supplementary data (e.g., control board layout/dimensions, equipment design specifications) will have to be obtained from the plant or from a further analysis of the task descriptive data. The supplementary data will be described in the final demonstration examples as well as the approach for obtaining that data.

The rationale for the separation of data sources is to highlight what unique data will be furnished by the NRC crew task analysis project versus what data is easily available from existing plant documentation or from other task analysis data bases (e.g., INPO's job/task analysis project). The additional data is not required to allow the user to search the data base effectively. The nature of the specific question confronting the user will determine (1) the level of detail of the data required to resolve the issue and (2) whether or not additional data is required.

The results of the crew task analysis project should demonstrate the robustness of the task data base structure to resolve questions relating to human engineering design, staffing and qualifications, training, procedures, job performance aids and communications in the nuclear power plant control room. Emphasis in the project is placed on collection of dynamic crew performance data within the context of well-defined operating sequences versus collection of detailed data such as control room instrumentation and control inventories.

The Human Error Analysis Program

William J. Luckas, Jr.
Engineering Analysis & Human Factors Group
Brookhaven National Laboratory

The objective of this program has been to develop and apply realistic human performance data and models to help evaluate the human's role in nuclear power plant safety. To meet this objective, the major FY 82 effort was placed in several areas of investigation and accomplishment, namely:

- The further development of Human Error Rates (HERs)

- The use of Performance Shaping Factors (PSFs) and quantified expert judgement in the evaluation of human reliability - the Success Likelihood Index Method (SLIM).

- The development of the Operator Action Tree/Time Reliability Correlation (OAT/TRC) approach for post event human decision errors.

- The publishing of the Conference Record for the 1981 Workshop on Human Factors and Nuclear Safety.

Please note that the reference list at the end of this paper contains the following mentioned documents along with other safety research in human performance related references recently published at Brookhaven National Laboratory.

Human Error Rates

The use of Licensee Event Reports (LERs) and appropriate experience has resulted in the detailed identification, analysis, and categorization of over four hundred implicit human errors. These have been obtained directly from the detail scrutiny of over six thousand LERs. The errors were obtained on events relating to safety related pumps and valves and to instrumentation and control (I&C) and electrical components. They have provided an actual human error data base many times greater than provided explicitly by the LERs "cause code" or the LER Data Summaries. These detailed analyses of LERs has been documented in NUREG/CR-2417 (for safety related pumps and valves) and NUREG/CR-2987 (for I&C and electrical components).

These data bases are intended to provide a realistic assessment of the real human error data base obtainable from the LERs. As a result, these data bases will provide a synthesis with the methodology found in NUREG/CR-1880 (for safety related pumps and valves) and NUREG/CR-2416 (for I&C and electrical components) for obtaining the opportunities for those errors identified and put into the bases. This will lead to the generaton of HERs where

$$\text{HER} = \frac{\text{human errors}}{\text{opportunities for those errors}}$$

from nuclear data and licensed operational experience.

Success Likelihood Index Method (SLIM)

The use of Performance Shaping Factors (PSFs) and quantified expert judgement using SLIM is important in the evaluation of human reliability. It should be noted that the amount of authentic quantitative human reliability data which exists is small (and is likely to remain small for the foreseeable future). It is, therefore, likely that subjective judgement and extrapolation will continue to play an important part. Nevertheless, present extrapolation techniques are covert, unsystematic, and rely on the knowledge of a limited number of judges. They do not systematically take into account the way in which PSFs combine together to affect the probability of success in particular situations. Moreover, certain tasks cannot effectively be quantified using reductionist approaches. For these tasks, involving diagnosis, decision making and other cognitive activities, a holistic technique will probably be necessary.

Quantified subjective judgement has emerged from the previous analysis as being of critical importance for human reliability evalution. SLIM is a quantified subjective judgement approach which uses PSFs as comprising any or all of the factors which cobmine to produce the observed likelihood of success. The basic premise of the approach is that when an expert judge (or judges) evaluate(s) the likelihood that a particular task will succeed, he or she is essentially considering the utility of the combination of PSFs in the situation of interest in either enhancing or degrading reliability. SLIM has the means of positioning a task on a subjective scale of likelihood of success, which is subsequently transformed to a probability scale. This positioning is derived by considering the judges' perceptions of the effects of the PSF in determining task reliability. NUREG/CR-2987 documents the initial appraisal of SLIM.

Operator Action Tree/Time Reliability Correlation (OAT/TRC)

The development of OAT/TRC is presented as an interim approach for estimating the probability of nuclear power plant (NPP) operators failing to take actions required to terminate or mitigate potential NPP accidents. This model is intended to fill the void in the area of human performance known as decision making or cognitive behavior by complementing the currently used event tree/fault tree methods. This approach is understandable to the systems engineer as well as the human factors expert and relies on available information as to the real time available to make a decision prior to unrecoverable core damage. In this way, it is clearly linked to the consequences of the action as it should be. NUREG/CR-3010 documents the interim framework of OAT/TRC.

References

1. NUREG/CR-2417 - Identification and Analysis of Human Errors Underlying Pump and Valve Related Events Reported by Nuclear Power Plant Licensees.

2. NUREG/CR-2987 - Identification and Analysis of Human Error Underlying Electrical/Electronic Component Related Events Reported by Nuclear Power Plant Licensees.

3. NUREG/CR-1880 - Initial Quantification of Human Errors Associated with Reactor Safety System Components in Licensed Nuclear Power Plants.

4. NUREG/CR-2416 - Initial Quantification of Human Error Associated with Specific Instrumentation and Control System Components in Licensed Nuclear Power Plants.

5. NUREG/CR-2986 - The Use of Performance Shaping Factors and Quantified Expert Judgement in the Evaluation of Human Reliability: An Initial Appraisal.

6. NUREG/CR-1879 - Sensitivity of Risk Parameters to Human Errors in Reactor Safety Study for a PWR.

7. NUREG/CR-2211 - Modeling of Multiple Sequential Failures During Testing, Maintenance, and Calibration.

8. NUREG/CR-3010 - Post Event Human Decision Errors; Operator Action Tree/ Time Reliability Correlation.

9. Conference Record for NRC/BNL/IEEE Standards Sponsored December 1979 Workshop on Human Factors and Nuclear Safety.

10. Conference Record for NRC/BNL/IEEE Standards Sponsored September 1981 Workshop on Human Factors and Nuclear Safety.

Human Factors in Loose Parts Monitoring

James H. Howard, Jr.

The Catholic University Human Performance Laboratory

An operating nuclear power reactor produces a wide range of audio-frequency signals. These signals fall into two major categories. First, there are relatively constant, broadband background sounds associated with the hydrodynamic flow of coolant and steam within the nuclear steam supply system (NSSS) and steady-state mechanical sounds such as those produced by coolant pump motors. Second, there are relatively brief duration transient signals which occur only infrequently. These can be produced by normally functioning equipment such as actuators and relays as well as by problematic events including equipment failures (e.g., bearing chatter), and metallic impacts of loose parts with components of the NSSS.

For several years it has been recognized that these audio frequency signals, especially the transient signals, can be a significant source of safety-related information. For example, it is likely that detached and freely-drifting or loose and abnormally vibrating parts within the NSSS ultimately will impact structural components of the reactor.

Recent experience has suggested that loose parts can occur with surprising frequency (Thie, 1981), and significant structural damage can occur if these parts are not detected and eliminated at the earliest possible time (Martin, 1982). Because of the safety significance of loose parts, most water cooled nuclear plants now in operation are equipped with a loose parts monitoring system (LPMS). These systems include accelerometers attached to areas where loose parts are likely to collect such as the reactor vessel and steam generator plenums. The electrical output of these sensors is amplified and led to instrumentation in the control room. Typically, the system can operate either in automatic mode to monitor the incoming data continuously, or in manual mode with the human operator listening to the audio output over headphones or a loudspeaker.

The present study was undertaken to assess the safety-related information potentially available to power reactor operators in the audio output. To accomplish this several tasks were completed. First, a literature review was carried out to identify acoustic signal parameters of primary interest and to relate these physical parameters to known human auditory detection and recognition capabilities. An objective of this task was to determine the level of performance to be expected during manual operation of current and future loose parts monitors. Second, the current use of auditory information in loose parts monitoring was determined. This task entailed examining manufacturer product descriptions and visiting a major instrumentation vendor and two nuclear plants for discussions with equipment designers and users. The plants included one which has had extensive experience with loose parts and another which has not experienced a loose parts incident in several years. Third, recommendations were made regarding the optimal use of the human auditory capability to ensure the safe operation of nuclear plants as well as for additional research which may be required to eliminate any deficiencies that now exist. In the remainder of this presentation I will summarize our findings.

The collision of a loose part with an NSSS structure will set up shock waves within the structure at the natural frequencies of the system components. These fluid- or structure-borne waves radiate from the point of impact to the sensor location. A variety of factors will influence the audio signal that is heard by the operator. These include the modal geometry of the reactor structures; sensor characteristics such as mounting, location, and sensitivity; the mass and velocity of the impacting part; background noises; and electrical transients from the containment environment.

A number of acoustic parameters characteristic of loose parts impacts are likely to be significant in determining the auditory detection and recognition performance of human operators. First, impact transients are of brief duration. Second, they occur relatively infrequently on a day-to-day basis. Third, impact signals are generally characterized by a rapid onset and an exponential decay although a number of factors will influence the envelope shape. Fourth, the signals are masked by a varying background noise and often occur at very low signal-to-noise ratios. Fifth, multiple or overlapping transients occur because of repeated impacts and multiple transmission paths to a single sensor. Sixth, the carrier signal for impacts has a complex, relatively broad spectrum within the audio-frequency range determined by several factors.

How are these acoustic data used? Our findings indicate that three major functions are currently served by a LPMS: surveillance, diagnosis, and feedback. The first of these functions, surveillance, is identified as primary in Regulatory Guide 1.133 (NRC, 1981). The surveillance function focuses on the early detection of metallic loose parts in the NSSS. Although surveillance depends primarily on continuous monitoring in the automatic mode, occasional monitoring does and should occur in the manual mode.

The second, diagnosis, function of the LPMS will generally follow the detection of a loose part. This function involves an assessment of the safety significance of the detected part in order that appropriate action may be taken to eliminate any potential safety threat at the earliest possible time. During diagnosis, support personnel estimate the location, number and approximate size of the parts. This information will be used to supplement data from a number of other sources to assess the safety significance of the loose parts.

A third function of manual listening involves the use of audio feedback to provide supplementary information regarding the response of remote equipment. For example, the sounds produced by control rod adjustments can reveal whether the equipment has responded properly to the operator's commands. This function of the LPMS was revealed in our interviews of control room personnel; it is not mentioned in either Regulatory Guide 1.133 or in previous reviews of loose parts monitoring (e.g., Kryter & Ricker, 1979).

What listening skills are involved in these tasks? We have identified three broad categories of auditory capabilities which underlie the three functions described here. Auditory detection is involved in surveillance, auditory recognition underlies diagnosis, and semantic interpretation of auditory signals will be involved in both the diagnostic and the feedback functions of the LPMS. In detection the operator decides whether an impact occurred or not--are loose parts actually present or am I hearing only noise? In recognition, the listener (an operator or engineer) characterizes or identifies the source of the impacts. Where do they originate; what is producing them? In semantic interpretation, the listener attaches meaning to the sounds; that is, the sounds are related to the listener's knowledge of the reactor and its function. This enables the safety significance of the parts to be determined.

Several conclusions may be drawn. Unfortunately, only a very brief overview can be presented here; the conclusions are spelled out in detail in the final report for the project (Howard, 1982).

First, the surveillance function can be well served by automatic monitoring with state-of-the-art, digitally-based equipment. Manufacturers of these systems employ various signal processing techniques to minimize false alarms and to optimize the reliability of automatic impact detection. These features are important since intolerably high false alarm rates seem to be a continuing problem with some older equipment. The automatic monitoring should be supplemented by periodic "manual" listening by human operators either to investigate an alarm condition or for occasional surveillance. The ability of the human listener to detect brief-duration acoustic transients in noise or to notice a change in normal background conditions is exceptional, particularly over relatively short listening periods. If detection performance is to be optimized, however, improved training procedures should be introduced. Many operators have little or no knowledge of what loose parts impacts actually sound like since little opportunity exists to hear them "on the job." It is well known in the psychacoustics literature that listener expectancy can play an important role in detection performance.

Second, the diagnostic function has been automated or partially automated in only the most sophisticated LPMSs. These systems employ a greater number of sensors to determine the location of loose parts precisely. Given this information, the impact signatures may be analyzed to estimate the mass of the impacting objects. Despite these recent advances, however, automatic diagnosis is limited by both theoretical and practical considerations. The possibility of using the human auditory capability more systematically to supplement the automatic analysis should be considered. In particular, the human listener is extremely good at comparing the spectra of complex sounds since even subtle spectral differences are heard as changes in timbre or sound quality. Perhaps some auditory procedure could be developed in which the listener makes a series of discriminations between live or recorded impacts from objects of unknown mass and a calibrated series of sounds. Psychophysical scaling methods could be used to convert these judgments into an estimate of mass. When these methods are combined with state-of-the-art automatic techniques, improved diagnosis should result.

Third, it is unfortunate that many operators are not aware of the potential for using the LPMS as a remote feedback instrument. If ambiguity exists regarding the state of any sonically active in-containment equipment such as actuators, motors or relays, then existing monitors may be useful as a supplemental source of information. Furthermore, with the placement of additional sensors, this function could be extended to include other remote equipment in high-radiation or difficult access areas. For example, acclerometers could be mounted on coolant pumps, solinoids, turbines, and piping. With proper training and adequate comparative aids, this equipment could be monitored using the auditory modality. Similar noise and vibration monitoring techniques are currently in use on nuclear submarines.

In conclusion, everyday experience suggests that people are very good at identifying the source of acoustic signals. The "skilled ears" of the sonar technician, the physician, and the auto mechanic enable them to identify important events simply by listening. Our findings suggest that the human listener's natural ability to recognize and interpret complex sounds may be used to greater advantage in monitoring the audio output of nuclear power reactors than is currently the case.

References

Howard, J.H., Jr., Catholic University Human Performance Laboratory, "Auditory Perception in Loose-Parts Monitoring," USNRC Report NUREG CR/3008, 1982. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

Kryter, R.C. & Ricker, C.W., Oak Ridge National Laboratory, "Characteristics and Performance Experience of Loose-Part Monitoring Systems in U.S. Commercial Power Reactors," USNRC Report NUREG/TM-254, 1979. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

Martin, T.T., "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant," USNRC Report NUREG-0909, 1982. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

Thie, J.A., Power Reactor Noise, La Grange Park, Illinois: American Nuclear Society, 1981.

U.S. Nuclear Regulatory Commission, Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." Copies are available from U.S. Government Printing Office, Washington, D.C. 20402. ATTN: Regulatory Guide Account.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION TASK ANALYSIS

R. P. Bateman, Ph.D.
S. B. Hottman
D. W. Biers, Ph.D.

Systems Research Laboratories, Inc.
2800 Indian Ripple Road
Dayton, Ohio 45440

This report presents the results of a task analysis and recommendations for the training and certification of operations technicians at independent spent fuel storage installations. Its purpose is to provide a technical basis for initial and continuation training for operations technicians at Independent Spent Fuel Storage Installations (ISFSIs). It also provides guidance for testing operations technicians to ensure that training objectives have been achieved. The recommended testing provides a basis for certification of ISFSI operators. The basis for this report was a task analysis conducted at the ISFSI at Morris, Illinois. Supervisors were interviewed and a preliminary job analysis was used to determine required operator skills. Training, safety and operating documents and checklists were reviewed and task inventory forms were developed with the help of ISFSI supervisors. Operations Technicians were then interviewed and the task inventory forms filled out with information on task frequency, difficulty, hazard, time to complete and error potential. These data were analyzed to determine required operator skills and proficiency levels necessary for safe ISFSI operations. The training and testing for certification necessary to verify the skills and proficiency levels were inferred from the data base and the Morris operations records.

The study was initiated to answer the following questions:

1. What skills are necessary to be an effective ISFSI operations technician?

2. What areas and levels of technical knowledge are necessary for the operations technicians to operate the facility safely?

3. What type of examination/certification procedure would best verify the presence or absence of the skills and knowledge necessary to operate an ISFSI safely?

The spent nuclear fuel receipt/storage cycle consists of (1) casks containing spent fuel are off-loaded from the transport vehicle; (2) cask is placed by crane in the unloading pit; (3) fuel is removed from cask under water and transferred to a storage basket; (4) storage basket is placed in one of the storage basins; (5) cask is removed from the unloading pit; (6) cask is decontaminated and surveyed; (7) cask is loaded onto transport vehicle and shipped off-site.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION TASK ANALYSIS

The time required for this entire procedure depends upon several variables including the type of cask, weather, time of day, shift size, and backlog to name a few. Because a majority of the fuel receipt/storage work occurs under water (for radiation protection purposes), the operations technicians may employ various devices to enhance their fuel basket tool manipulation coordination. These devices include underwater TV camera, underwater periscope, and/or binoculars.

Following receipt, unloading, and storage of the spent fuel, operations at an ISFSI center around maintaining the fuel bundles in a safe and secure environment. The basin (or pool) in which the fuel is stored to prevent contamination of the environment must be cooled and the water filtered to remove impurities which may transport radioactive sources. Filter material and other substances that have been exposed to contamination must receive special handling and be stored in a Low Activity Waste (LAW) vault. Installation systems must be checked for proper operation to prevent leaks of contaminated air or water. Fuel accountability checks must be made and radioactivity must be monitored to ensure that the entire system is under control. Once the fuel has been stored, the ISFSI operation becomes more of a process monitoring than a materials handling situation.

Because the receipt of fuel was not accomplished during this study, actual observation of work activities was not possible. Task inventory questionnaires were used to obtain subjective judgements from experienced operations technicians in a form suitable for numerical analysis. Data was gathered on 16 activities (217 tasks) which were judged to be typical of routine operations. From the results of the Task Inventory, there emerged a list of 63 tasks, approximately 30% of the tasks selected for analysis, which merit additional attention. For a task to be included on this list, it had to meet the criterion of being rated above average in at least one of the following areas: Difficulty, Hazard, and error-likelihood. Twenty-five of the tasks were judged to be both difficult and hazardous. One was judged to be both difficult and error-likely. Ten tasks were rated difficult, seven were rated hazardous, and twenty were identified as error-likely.

This information was analyzed along with data on frequency and time spent in order to make recommendations on training and certification procedures. The existing training program at the Morris ISFSI was analyzed in detail to obtain baseline data on the training of operations technicians. Training documents were studied, actual training materials were reviewed, and training and testing records were examined to determine what was being done. For comparison, material from the training program at the Barnwell Nuclear Fuel Plant was reviewed. In general, the existing training program at the Morris ISFSI was found to be satisfactory. The certified operations technicians are highly qualified and fully competent in the handling and storage of spent nuclear fuel. The training program is basically good without changes. Recommendations include the use of inert fuel bundles and simulators to provide hands-on training in psychomotor skills. There are recommendations to simplify procedures, streamline designs of controls and displays, and reassign some tasks.

It was concluded that a person needed only a high school education and normal psychomotor skills to qualify as an operations technician. Far more important than knowledge or dexterity are the character traits of reliability, carefulness, and dependability. In the process of earning certification, novice personnel acquire skills in crane operations, process control, rail car operations, radiation monitoring, and log keeping. The job requires familiarity with a number of utility systems and processes such as demineralization and electrodecontamination.

While the existing training program was judged to be effective in preparing operations technicians to perform their tasks safely and efficiently, the certification examinations presently in use were judged inappropriate. A valid certification program requires observation and evaluation of actual performance. The certification program at the Morris ISFSI is especially strong in this area. A minor recommendation is that the "walk-through" examination be expanded to require more actual operation of equipment. In some cases, the use of simulated equipment is recommended.

The written examinations used for certification were designed to measure general intelligence level and general academic background. It is recommended that examinations be developed to measure specific knowledge required to perform the tasks. It is further recommended that these tests be standardized and maintained at NRC. In this way, NRC will have the ability to measure the required knowledge level of all ISFSI operations technicians and the ability to evaluate a random sampling of technicians during on-site "walk-through" examinations. This combination permits the individual ISFSIs to control the certification of their employees, yet provides the NRC with a reliable procedure to verify the presence or absence of the skills and the knowledge necessary to operate an ISFSI safely.

EMERGENCY OPERATING PROCEDURE VALIDATION METHODOLOGY

JAMES L. vonHERRMANN

WOOD-LEAVER & ASSOCIATES

The Plant Status Monitoring (PSM) Program was initiated by the Nuclear Regulatory Commission to develop and validate methods to systematically address a number of important safety issues concerned with enhancing the operator's ability to respond to potential accident conditions. In the flurry of post-TMI activity related to investigating the role of the operator in overall plant safety, the need was perceived for a logical framework to address these various issues in a manner which would ensure that any resultant conclusions and recommendations would be firmly anchored to a thorough physical understanding of the plant response to important potential accident conditions.

The basic thesis of the PSM program is that, while there are numerous facets of the overall man/machine interface problem, any efficacious changes to plant design and/or operation must be based on a firm foundation consisting of:

- An explicit identification of potential accident sequences and the plant states comprising these sequences.
- A careful delineation of the actions required of the operator at each plant state.
- A clear understanding of the physical phenomenon associated with each plant state.

In previous projects performed under the Plant Status Monitoring Program (Refs 1,2), it has been demonstrated that Operator Action Event Trees (OAETs) can provide this systematic tabulation of the key operator actions and plant symptoms associated with the various stages of risk significant multiple failure accident sequences.

One of the significant tasks to which this foundation of information has recently been applied is the production of improved emergency operating procedures guidelines.

Subsequent to the accident at Three Mile Island (TMI), industry groups have endeavored to develop emergency procedures which do not require the operator to diagnose a specific event or series of events before guidance is provided. In the first phase of the work reported here, the function- or symptom- oriented approaches which have evolved since TMI are summarized and discussed. It was concluded that these alternate approaches to guideline development - as exemplified by the programs of groups associated with each of the four major U.S. vendors - have, in theory, the potential to produce effective guidelines. However, when attention is focused on a limited number of critical safety functions (or symptoms indicative of the performance of these functions), the concern arises that diverse accident conditions which exhibit common or similar symptoms can result in ambiguous operator diagnosis and ineffective response.

Thus, the potential pitfalls which must be avoided in the practical application of these alternate approaches to guideline development are closely linked with the primary motivation for their development. The pre-TMI procedures required the operator to know too much before he could be assured of taking the right action. The proposed remedy is to provide guidance based on much less information (a limited number of key symptoms associated with the performance of a few critical functions). However, whenever guidance is based on limited information, extreme care must be taken to assure that it is always correct and unambiguous.

Systematic methods were, therefore, developed which can help assure that functional or symptomatic guidance can, in reality, lead to unambiguous and effective diagnosis and response regardless of the specific failure events. These methods were based on the use of the Operator Action Event Trees noted above.

OAET-based methods were developed which can systematically "produce" unambiguous guidelines in three basic ways:

(1) Preliminary or incomplete guidelines can be "finalized" using input gained from a systematic OAET-based investigation of the incomplete guidelines.

(2) Complete guidelines can be systematically reviewed and any inadequacies corrected.

(3) Guidelines can be produced directly from the OAETs.

The guideline review methods (which, along with the guideline development method, is presented and discussed in Ref. 3) systematically compares the actions and symptoms documented in the OAETs with the actions and diagnostic symptoms cited in the Guidelines under review. This comparison is designed to answer four basic questions:

(1) Is the collection of symptom sets complete? That is, are there risk significant states requiring operator action which could occur but for which no guideline instruction applies?

(2) Are the instructions always right? That is, if the guidelines say "when you see Symptom Set A take Action Set P," is Action Set P always appropriate for every situation that can produce Symptom Set A?

(3) Are the action sets always complete? That is, are there important actions which should be carried out at a particular state which are not included in the action set indicated at that state?

(4) Are the instructions always unambiguous? Are there plant states which produce symptom sets which the operator might confuse with guideline symptom sets and thereby take inappropriate action?

Subsequently, the ability of these OAET-based methodologies to produce effective guidelines applicable to a Westinghouse PWR plant design was investigated. An additional product of this methodology application was the identification of those aspects of Westinghouse plant design, operation, or response to multiple failure accident sequences which could result in incomplete, ambiguous, or incorrect guidance to the operator if not carefully addressed in the guideline development process.

This application of the methodologies developed and presented in Reference 3 utilized OAETs developed for the Zion 1 Westinghouse PWR (Ref. 2). Best estimate analyses provided by the NRC's Severe Accident Sequence Analysis (SASA) Program were used as the primary source of information related to the physical plant response to multiple failure accident sequences.

The results of this investigation (documented in Reference 4) demonstrate that the OAET-based methodologies can provide a very effective tool to the regulatory process associated with the development, review, and ultimate implementation of functional emergency procedure guidelines applicable to Westinghouse PWRs. These methods could be especially valuable as an integral part of the regulatory process because:

- From the regulatory side, they provide an easily audited process which also provides very high assurance that the guidelines submitted by the Westinghouse Owner's Group and implementation plants submitted by the individual utilities operating Westinghouse PWRs will result in unambiguous operator guidance under all important accident conditions.

- From the industry side, they provide a well defined process by which regulatory concerns over the technical content of guidelines and procedures applicable to Westinghouse plants can be systematically satisfied.

In conclusion, it is recognized that the development of effective emergency procedures entails inputs from a wide variety of sources, ranging from plant transient analyses to human factors analyses. The OAETs offer a mechanism by which information concerning the realistic thermal-hydraulic response of plants to risk significant accident sequences can be systematically presented in a form which can be readily integrated into human factors engineering analyses. In order to produce effective guidelines, there must be a strong interaction between the human factors analysts and the plant thermal-hydraulics analysts. The OAET-based methodologies presented in this volume appear to provide the critical link which allows this interaction to occur.

REFERENCES

1. J.vonHerrmann, R.Brown, A.Tome, "Light Water Reactor Status Monitoring During Accident Conditions" NUREG/CR-1440, June 1980

2. R.Brown, J.vonHerrmann, J.Quilliam, "Operator Action Event Trees for the Zion 1 Pressurized Water Reactor," NUREG/CR-2888, Sept. 1981

3. J.vonHerrmann, "Development of Methodology for Evaluation of Emergency Procedure Guidelines, Volume I: Methodologies," to be published as NUREG

4. J.vonHerrmann, W.Brinsfield, R.Brown, "Development of Methodology of Evaluation of Emergency Procedure Guidelines, Volume II: Applications to Westinghouse Plants," to be published as NUREG

SURVEY OF FOREIGN REACTOR OPERATOR
QUALIFICATION, TRAINING, AND STAFFING REQUIREMENTS

by
Markley L. Au
and
Ellis W. Merschoff

U.S. Nuclear Regulatory Commission

1. Background

NUREG-0863, "Survey of Foreign Reactor Operator Qualification, Training, and Staffing Requirements" was published by the Office of Nuclear Regulatory Research in May 1982. It was compiled in an effort to provide the Commission with the information needed to place various proposed changes to the regulations in the areas of qualifications, training, and staffing in perspective with respect to the practices of other countries. These changes to the regulations include revisions to 10 CFR Part 55, which would upgrade education, training, certification, and requalification requirements; a revision to 10 CFR 50.54, which would upgrade staffing requirements for nuclear power plants; and a revision to 10 CFR 50.34 based on the issues discussed in NUREG-0737, "Clarification of TMI Action Plan Requirements.

The questionnaire was developed to request detailed information in five areas, shift staffing requirements, eligibility requirements for operators, operator training program requirements, initial operator's license or certification requirements, and operator retraining requirements. It was sent to the regulatory authorities in Argentina, Belgium, Brazil, Canada, Finland, France, Federal Republic of Germany, India, Italy, Japan, Korea, Mexico, the Netherlands, Spain, Sweden, Switzerland, Taiwan, and the United Kingdom. All responded.

Upon reviewing the responses to the questionnaire, it became apparent that certain factors should be considered if one wishes to draw conclusions from the data. These factors are:

- In order to ensure that the survey questions were applicable to the different organizational and educational systems encountered internationally, the questions were intentionally broad in scope. As a result, the specificity of the responses varied widely. It is frequently unclear whether the statement or condition is a regulatory requirement or simply a practice. This same type of uncertainty is common in the U.S. regulatory system (e.g., a regulatory guide which endorses a consensus industry standard is not necessarily a requirement, but is often cited as such and is usually adhered to in practice).

- Only a half-dozen or so of the respondents have accrued a significant number of reactor-years operating experience. The cumulative reactor-years of operating experience has been calculated for each respondent and is provided as Table I.

- It is not always clear whether the respondent was speaking as a regulator or as a member of the regulated industry. This relationship is known to vary internationally.

TABLE I

Cumulative Reactor Years
of Operating Experience

Country/Area	Reactor Years*
Argentina	7
Belgium	37#
Brazil	0
Canada	74
Finland	7
France	152
Federal Republic of Germany	93
India	32
Italy	53
Japan	126
Korea	3
Mexico	0
The Netherlands	20
Spain	31
Sweden	37
Switzerland	32
Taiwan	5
United Kingdom	527
United States	547

*As calculated using data from Nuclear News - February 1981.

#Reactor years include Westinghouse PWR Plant BR3 in operation at Mol, Belgium.

2. Results

The data obtained from this survey have been displayed in matrix form in the NUREG report to facilitate comparison among the respondents. All five of these matrices have been included in this paper as Appendix A. Since not all questions facilitated a concise answer, heavy use is made of footnotes to supplement the information represented on each chart. If more information in a given area is desired, such as a more complete description of the training program or operational responsibility, Appendix C of NUREG 0863 should be consulted.

2.1 Matrix 1, Shift Staffing

1. The tabulation of the total number of operating personnel in the control room during reactor operations was performed for a single operating unit rather than for each of the various multiple unit control room configurations in order to prevent the tabulation from becoming overly complex. The number of operators varied from a minimum of two to a maximum of

seven. Staffing by two operating personnel suggests that one senior reactor operator (SRO) or equivalent and one reactor operator (RO) are on duty during reactor operations. The extreme of seven operating personnel reported by Japan may be attributed to data which include assistant operators and balance of plant personnel. It should be noted that several countries indicated the use of assistant shift supervisors as well as shift supervisors.

2. In general, the minimum number of shift crews maintained by each respondent appears to be a practice of the respective utility and not an explicit regulatory requirement. Most foreign utilities maintain five or six shift crews per unit.

3. The majority of the respondents reported that each shift normally works 8 hours per day. The majority of respondents also indicated that there are restrictions on the amount of overtime hours worked by operating personnel. These restrictions may be attributed to national labor regulations, union, and national practices. Eight respondents have no specific requirements limiting the maximum number of continuous working hours that an operator is permitted to work in the control room.

4. A detailed description of the various shift structures can be obtained from the organization charts provided in NUREG 0863.

2.2 Matrix 2, Eligibility Requirements for Operators

1. Of the eighteen respondents, eight indicated a requirement for technical school training for reactor operators. Two more, India and the Netherlands, indicated some form of college/university level education but not necessarily a university degree requirement for reactor operators. Korea and Mexico are the only respondents that require a university degree.

2. Working experience requirements for reactor operators varied from a minimum of no specific experience to a maximum of seven years. The minimum experience was reported by the United Kingdom and the maximum was reported by Japan (only one of the required seven years experience in Japan must be gained working on a reactor similar to the one on which the operator will be assigned). Six respondents accept conventional power plant (fossil fuel or marine power plant work experience) as part of the experience requirement for reactor operators.

3. Of the eighteen respondents, five require a college or engineering degree for senior reactor operators and nine require a college or engineering degree for shift supervisors.

4. Senior Reactor Operators are expected to have from 1 to 5 years of nuclear power plant experience. Some conventional power plant operating experience is acceptable, but experience should be predominately in nuclear power plant operations. For shift supervisors, three respondents indicated that the candidate must serve as assistant shift supervisor for at least two years, and ten respondents indicated that experience as a reactor operator is required.

5. Sixteen of the eighteen respondents required some form of medical examination for the reactor operator, senior reactor operator, and shift supervisor candidates. Only six indicated requirements for aptitude or psychological testing of candidates.

2.3 Matrix 3, Training Program

1. Five of eighteen respondents indicated that the utilities were responsible for initial screening of reactor operator candidates. The length of the reactor operator training program varied from five months to seventy-two months depending on the prior experience and education of the trainee. Academic subjects in the training programs of most respondents included nuclear physics, thermodynamics, fluid mechanics, radiation safety and protection, and nuclear power plant systems. Fifteen of the respondents indicated some form of casualty exercise as part of the training program.

2. Nine respondents indicated that the trainees are tested during the training program, while five indicated that a final examination is given. Seventeen respondents required in-plant training, which suggests a consensus on the significance of such training for reactor operating personnel. Simulator training appears to be considered an important component of operator training by sixteen respondents. Training programs are generally reviewed by the national/central government regulatory authority to determine adequacy.

3. Fifteen of the respondents reported that the training instructors were experienced nuclear power plant senior reactor operators or engineers, supplemented as necessary by outside professional technical instructors. In general, utilities are responsible for maintaining an operator training program. However, they are not generally required to maintain a permanent training staff as a regulatory requirement.

2.4 Matrix 4, Initial Operator's License or Certification

1. Fourteen respondents require some form of examination for licensing or certifying reactor operators by the regulatory authority. The plant superintendent or manager is generally responsible for determining when a candidate is ready for the licensing examination. There are no requirements for licensing examinations by the regulatory authorities of four respondents. There is a consensus that the examinations should cover response to emergency situations, academic subjects, and knowledge of nuclear power plant operating procedures.

2. Of those respondents providing their passing criteria for regulatory examinations, about half use a quantitative evaluation (numerical grading in %) and the remainder uses a qualitative approach (e.g., "demonstrates adequate knowledge"). In most cases, the test results are passed back to the utility.

3. Generally, an operator's license is valid for two to three years.

2.5 Matrix 5, Operator Retraining

1. Generally, there are formal retraining programs for reactor operators and senior reactor operators although such programs are not universally required. Most of the respondents indicated that retraining included operations on reactors (e.g., start-up, shutdowns, and response to transients) as well as a one-to-two-week concentrated training period annually

3. Summary

The data collected as a result of this survey are presented in Matrices 1 through 5 of NUREG-0863. In general, there appear to be more similarities among programs than there are differences. Specifically, similarities can be noted in the areas of working hours, shift staffing, and experience requirements while significant differences exist in the areas of educational requirements and overtime restrictions.

Most respondents required either two or three operators in the control room during reactor operations, and a total of either five or six shift crews. A normal shift is eight hours per day and restrictions on the amount of overtime usually exist. However, these restrictions vary widely and may be expressed in terms of either hours per day, hours per week, hours per month, or hours per year.

A comparison of operator eligibility requirements revealed a spectrum of educational requirements varying from none to a university degree. Most respondents require some nuclear power plant experience for reactor operators. A medical examination for operators, senior reactor operators, and shift supervisors is required by a majority of the respondents.

NUREG-0863 provides an extraordinary amount of information on the details of the respondent's training program organization and content. Since a great deal of the information was not suitable for display in matrix format, the reader is encouraged to consult Appendix C of the NUREG for any information not found on Matrices 1-5.

MATRIX 1, SHIFT STAFFING

• RO - REACTOR OPERATOR
• SRO - SENIOR REACTOR OPERATOR
• SS - SHIFT SUPERVISOR

	(1) TOTAL NUMBER OF OPERATING PERSONNEL IN CONTROL ROOM DURING OPERATIONS	(1) BREAKDOWN OF OPERATING PERSONNEL IN CONTROL ROOM DURING OPERATIONS (ONE UNIT)				(1) NUMBER PERSONNEL DURING REACTOR SHUTDOWN IN CONTROL ROOM	(2) MINIMUM NUMBER SHIFT CREWS UTILITY MAINTAINS	(2) SHIFT WORKING HOURS	(3) RESTRICTIONS ON OVERTIME		(3) WHAT ARE THE RESTRICTIONS*	(4) MAXIMUM NUMBER CONTINUOUS HOURS OPERATOR AT CONTROL ROOM OPERATING BOARD				(5) TYPICAL SHIFT ORGANIZATION CHART CONTAINING MINIMUM NUMBERS AND TYPES OF PERSONNEL IN EACH SHIFT
	ONE UNIT$_L$	SHIFT SUPV	SRO	RO	OTHER	ONE UNIT$_L$	ONE UNIT	HOURS/DAY	YES	NO		YES	NO	HOURS	REG. AUTHORITY	
ARGENTINA	3	1	1 PO	1 SO		2	6	8	•		USUALLY NO WORK IN OVERTIME 8 HRS/DAY IN EXCEPTIONAL CIRCUMSTANCES	•		8 HRS EXCEPTIONAL CASE 16 HRS		SEE APPENDIX C ARGENTINA
BELGIUM	3	NI	NI	3		2	5	8		•	NR		•	2 ADDITIONAL HOURS		SEE APPENDIX C BELGIUM
BRAZIL	3	1	1	1		2	5	8	NR	NR	NR		•			SEE APPENDIX C BRAZIL ANSI N 18.1 1971
CANADA	2	1	1 CR	NA		2	5	8	•		100 HRS/YR*					SEE APPENDIX C CANADA
FINLAND F	2 L	1	NA	1		1	6	8	•		ONCE A WEEK 16 HRS SHIFT NO OVERTIME AFTER NIGHT SHIFT		•		IRP	SEE APPENDIX C FINLAND
FRANCE	3 L	1	1	1		3	CS	7, 8 OR 9	•		DAILY WORK HOURS CANNOT EXCEED 10 HOURS WEEKLY WORK HOURS CANNOT EXCEED 1. 50 HOURS PER WEEK 2. 48 HOURS PER WEEK (AVERAGE) EXTENDING OVER 12 CONTINUOUS WEEKS	•		10 HOURS	REGULATORY BODY	SEE APPENDIX C FRANCE 1 2
FEDERAL REPUBLIC OF GERMANY	3	1	NI	2		3	5	8	•		• 8 HRS/DAY INCLUDING • 12 HRS/DAY (SUNDAY ONLY)					SEE APPENDIX C FEDERAL REPUBLIC OF GERMANY
INDIA	3	1	1	1		3	4	8	•		LICENSED OPERATOR NOT PERMITTED TO WORK BEYOND 8 HOURS/DAY NO OVERTIME RESTRICTIONS		•			SEE APPENDIX C INDIA
ITALY	3	1	1	1		3	6	8	•		120 HRS/YR*		•			SEE APPENDIX C ITALY
JAPAN	9.12 $_J$	1	4_J		4.7 $_J$	9.12	NR	8	•		8 HRS/DAY*	•		16		SEE APPENDIX C JAPAN
KOREA	2	NI	1	1		2	5	8	•		8 HRS/DAY		•			SEE APPENDIX C KOREA
MEXICO	3	NI	2	1		2	NR	48 H/WK	•		3 HRS/DAY 9 HRS/WEEK*		•			SEE APPENDIX C MEXICO
THE NETHERLANDS	2	1	NA	1		2	5.6	8	•		8.5 HRS/WEEK, NO MORE THAN 12 HRS CONTINUOUSLY	•		4 ADDITIONAL HOURS		SEE APPENDIX C NETHERLANDS
SPAIN	3	1 L	1 A/S	1	1	2	6	8	•		• NO MORE THAN 12 HOURS STRAIGHT WITH 12 HOUR BREAK • NO MORE THAN 72 HOURS IN 7 DAY PERIOD • NO MORE THAN 14 DAYS WITHOUT 2 DAYS OFF*	NI				SEE APPENDIX C IN SPAIN SECTIONS 5 THROUGH 5.5
SWEDEN	2	1	NA	1 SW		2	7	8	•		48 HRS/4 WEEKS 50 HRS/MONTH* 150 HRS/YR	•		14 SEE ITEM 3		SEE APPENDIX C SWEDEN
SWITZERLAND	2	1 B	1 PE	1		2	7 + 1 PE	8	•		2 HRS/DAY 220 HRS/YR*	•		8 HRS 40 MIN	BIGA	SEE APPENDIX C SWITZERLAND
TAIWAN	2	1	1	1		1 AND 1	CS_1	8		•	8 HRS DAY* NO OVERTIME RESTRICTIONS	•		16	PLANT OPERATION DIVISION	SEE APPENDIX C TAIWAN
UNITED KINGDOM	2	NI	1	1		1	5	8		•	• SUPERVISORS DO NOT WORK OVERTIME • OPERATOR OCCASIONALLY WORK DOUBLE SHIFT ON REST DAYS		•	2 ADDITIONAL HOURS		SEE APPENDIX C UNITED KINGDOM
UNITED STATES	3	1	1	2		1 AND 1	NR	8	•		• NO MORE THAN 16 HOURS STRAIGHT (EXCLUDING SHIFT TURNOVER TIME) • NO MORE THAN 16 HOURS IN ANY 24 HOUR PERIOD NO MORE THAN 24 HOURS IN ANY 48 HOUR PERIOD NO MORE THAN 72 HOURS IN ANY SEVEN DAY PERIOD (ALL EXCLUDING SHIFT TURNOVER TIME) • BREAK OF AT LEAST EIGHT HOURS ALLOWED BETWEEN WORK PERIODS	•		4 HRS ON DUTY WITH 10 MINUTE BREAK	NRC	

FOOTNOTES:
- NR - NOT REGULATORY REQUIREMENT
- NI - NO INFORMATION PROVIDED
- [illegible] - ON SITE
- NA - NOT APPLICABLE
- B - IN BUILDING OF CONTROL ROOM
- * - MAXIMUM HOURS LIMITED BY GOVT. OR UNION
- A/S - ASST. SHIFT SUPV
- L - ONLY ONE UNIT TABULATED DUE TO LIMITED AMOUNT OF INFORMATION
- CR - CONTROL ROOM OPERATOR (CRO) EQUATE TO SRO
- [illegible] - TWO PERSONS IN CONTROL ROOM IF THIRD ON SITE
- PE - PICKET ENGINEER HAS SRO LICENSE ONSITE
- F - NO SENIOR REACTOR OPERATOR (SRO) BUT EQUIVALENT SHIFT SUPERVISOR
- J - MULT UNIT & UTILITY POLICY
- SW - RO & SRO ONE OF SAME IN SWEDEN
- PO - PRIMARY OPERATOR
- SO - SECONDARY OPERATOR
- CS - INFORMATION PROVIDED FOR TWO UNITS/MINIMUM CREW PER SHIFT. SEE APPENDIX C
- CS_1 - CREW PER SHIFT GIVEN. SEE APPENDIX C

The education, training and experience requirements for U.S. operators contained in Regulatory Guide 1.8 and ANS 3.1-1971 were not included as U.S. requirements due to the tenuous relationship between requirements and regulatory guides. It should be noted therefore, that U.S. requirements, not practices, are listed on the matrix as U.S. position.

Care should be used when comparing requirements such as education. The differences between the American and European secondary and university level educational systems are marked. Additionally, other sources of education and technical knowledge such as technical and military schools do not easily lend themselves to comparison. A significant amount of variation can be found in the course content and level of presentation for similar institutions in a given country. For example, the Massachusetts Institute of Technology and the United States Naval Academy both teach Aerospace Engineering, however, there is a substantial difference in curriculum content and teaching methods used at these two prestigious institutions.

A-1

APPENDIX A

MATRIX 2. ELIGIBILITY REQUIREMENTS FOR OPERATORS

• RO - REACTOR OPERATOR • SRO - SENIOR REACTOR OPERATOR • SS - SHIFT SUPERVISOR	(1) MINIMUM EDUCATION PREREQUISITES FOR REACTOR OPERATOR	YEARS FORMAL SCHOOLING	UNIVERSITY LEVEL		(2) MINIMUM EXPERIENCE FOR REACTOR OPERATOR (YRS)	(3) MINIMUM EDUCATION PREREQUISITES FOR SENIOR REACTOR OPERATOR/SHIFT SUPV	
			YES	NO		SRO	SS
ARGENTINA	7 YRS PRIMARY SCHOOL	TECHNICAL SCHOOL 6-8		•	NPP - 2	TECHNICAL SCHOOL PO	ENGINEERING DEGREE
BELGIUM	FORMAL SCHOOLING TO AGE 18	TECHNICAL SCHOOL 8 (+)		•	FPP - 2 NPP - 1	SAME AS (1) SAME AS (1)	SAME AS (1) SAME AS (1)
BRAZIL	HIGH SCHOOL EQUIVALENT	SCIENTIFIC OR TECHNICAL (NI)		•	GENERAL FPP - 3 NPP 1	SCIENTIFIC OR TECHNICAL	UNIVERSITY OR ENGINEERING DEGREE
CANADA$_{CR}$	NA	NA	NA	NA	NA	NR$_{CR}$	NR
FINLAND$_{F}$	BASIC SCHOOL & TECHNICAL SCHOOL	12		•	TOTAL - 2 NPP - 1	NA	ENGINEERING DEGREE
FRANCE	NR	12$_{NR}$		•	OPERATOR/ TECHNICIAN 2-5	NR	NR
FEDERAL REPUBLIC OF GERMANY	CRAFTSMAN TECHNICIAN	12 YRS GENERAL AND VOCATIONAL SCHOOL FOR TECHNICIAN IN ADDITION 2 YRS TECHNICAL SCHOOL (C)		•	NPP - 2.0** FPP - 0.5	MASTER CRAFTSMAN TECHNICIAN	ENGINEERING DEGREE
INDIA	HIGH SCHOOL EQUIVALENT	SCHOOLING - 10 DIPLOMA ENG - 3 OR SCHOOLING - 12 SCIENCE UNIV - 2		•	2	ENGINEERING DEGREE	ENGINEERING DEGREE
ITALY		8		•	2 YRS TRAINING INCLUDE 10-11 MONTHS ON REACTOR SIMILAR TO ONE TO BE ASSIGNED	SCHOOLING - 13 ENG/TECH CERT →	
JAPAN	NR	NR		•	NPP - 7* INCLUDE 1 YR SAME TYPE REACTOR	NR	NR
KOREA	HIGH SCHOOL	ND	•		NPP - 2	UNIVERSITY DEGREE	NI
MEXICO	ENGINEERING SUBJECTS	4	•		ANSI/ANS 3.1 1978	UNIVERSITY DEGREE →	
THE NETHERLANDS	COLLEGE FOR MARINE ENGINEERING GRADE B OR EQUIVALENT	6$_{(+)}$		•	FPP - 2 OR EQUIVALENT NPP - 1	NA	GRADE C
SPAIN	TECHNICIAN	TECHNICAL SCHOOL		•	FPP OR NPP - 1 NPP - 1 TOTAL - 2	HIGH SCHOOL PLUS	NR
SWEDEN$_{SW}$	GENERAL EDUCATION TECH SCHOOL	13		•	NPP - 3.5-4 PLUS 1 IN SERVICE	NR	NR
SWITZERLAND	TECH SCHOOL	13		•	GEN'L - 2 NPP - 1-2	SAME AS (1) →	
TAIWAN	HIGH SCHOOL	12		•	NPP - 3 OR FPP - 2 OR COLLEGE DEGREE & NPP - 1 OR FPP - 1	TAKE SRO EXAM DIRECTLY UNIVERSITY DEGREE BUT NOT LIMITED TO DEGREE HOLDER →	
UNITED KINGDOM	FORMAL SCHOOLING TO AGE 16	NI		—	NO SPECIFIC PRIOR EXPERIENCE SPECIAL TRAINING WHERE EMPLOYED	NATIONAL CERTIFICATE (EQUIVALENT) TO 2 YR. UNIV.	ENGINEERING DEGREE OR EQUIVALENT
UNITED STATES	NR	NR		*NR	NR	NR	NR

	UNIVERSITY LEVEL				(4) MINIMUM WORKING EXPERIENCE PREREQUISITE FOR (YRS)		IS RO WORKING EXPERIENCE PREREQUISITE FOR SRO OR SS?		(5) MEDICAL EXAMINATION						(6) APPITUDE TEST[1] OR PSYCHOLOGICAL TEST[2]					
	SRO		SS						RO		SRO		SS		RO		SRO		SS	
	YES	NO	YES	NO	SRO	SS	YES	NO	YES	NO	YES	NO	YES	NO	YES	NO	YES	NO	YES	NO
ARGENTINA		•	•		NPP - 2	ASSISTANT TO TRAINED SS	ND	ND	•		•		•		$*^1_2$		$*^1_2$		$*^1_2$	
BELGIUM		•		•	NI SS - 2	NPP - 4 SS - 2	ND	ND	•		•		•		$*_2$		$*_2$		$*_2$	
BRAZIL		•	•		GENERAL - 8 (NPP - 1 & FPP - 7) OR RO - 2	GENERAL - 2.5 (INCLUDE NPP - 1 & FPP - 1)		•	•		•		•		$*_P$		$*_P$		$*_P$	
CANADA$_{CR}$	NR	NR	NR	NR	NPP - 1.5 MPP$_{CR}$	12 YRS AFTER HIGH SCHOOL 4 YRS AFTER UNIVERSITY	NA	NA	NA	NA	•		•		NA	NA	NS	NS	NS	NS
FINLAND$_{F}$	NR	NR	•		NA	ENGR TOTAL 3 W/NPP - 1 TECHNICIAN - 7 W/NPP - 3		•	•		•		•			•	NI	NI	NI	NI
FRANCE	NR	NR	NR	NR	RO - 2 SHIFT - 4-7 OR OP TECH - 2.5	SRO - 2.5 ASST SS - 2 OR SHIFT WORK 9-13		•	•		•		•		$*_1$		$*_1$		$*_1$	
FEDERAL REPUBLIC OF GERMANY		•	•		NPP - 3	NPP - 3$_P$	•		•		•		•		NS →					
INDIA	•		•		NPP - 5	ASST SS - 2		•	•		•		•		$*_1$		NI →			
ITALY		•		•	NPP - 2	NPP - 2	•		•		ND →				— →					
JAPAN	NR →				NR	SAME AS (2)	•		— →						— →					
KOREA	•		NI	NI	NPP - 3-4	NI	•		•		NI →					•	NI →			
MEXICO	•		•		ANSI/ANS 3.1 1978 →		—	—	ANSI/ANS N546 1979 →											
THE NETHERLANDS	NA			•	NA	NPP - 2 (AS RO)	•		$*_P$		NA		•		$*_P$					•
SPAIN	NR	NR	NR	NR	FPP OR NPP - 3 → NPP - 1 TOTAL - 4		—	—	•		•		•		•		•		•	
SWEDEN$_{SW}$	NR	NR	RN	NR	NA	NPP - 5	•		•	•	NA	NA	•		NR →					
SWITZERLAND		•		•	GEN'L - 2 COP - 1.5 BOP - 2 TOTAL - 5.5	GEN'L - 2.0 COP - 1.5 3OP - 2.0 A - OP - 2.0 TOTAL - 7.5	•		•		o		•		$*_P$		NR		$*_P$	
TAIWAN	•		•		5	ND		•	•		•		•		•	•	$*_2$			•
UNITED KINGDOM	NR	NR	•		NR$_U$	NR$_U$	•		•		$*_{UK}$		$*_{UK}$			•		•		•
UNITED STATES	NR →				1 YR AS RO OR NPP & DEGREE		•		•		•		•			•		•		•

FOOTNOTES
NR - NOT REGULATORY REQUIREMENT
ND - NOT SPECIFICALLY DEFINED
NI - NO INFORMATION PROVIDED
— - NOT SPECIFIC REQUIREMENT
NA - NOT APPLICABLE

NPP - NUCLEAR POWER PLANT
FPP - FOSSIL POWER PLANT
MPP - MARINE POWER PLANT
P - PROPOSED REGULATION
CR - CONTROL ROOM OPERATOR (CRO) EQUATE TO SRO

F - NO SENIOR REACTOR OPERATOR (SRO) BUT EQUIVALENT TO SHIFT SUPERVISOR
U - UTILITY POLICY
SO - SHIFT OPERATIONS
NS - NO TEST ASSESSED BY SUPERVISORS
* - ONLY FOR SHIFT SUPERVISOR

SW - RO & SRO ONE OF SAME IN SWEDEN
UK - 1 SHIFT ENGINEER IN CHARGE (SE) = SS
2 CONTROL ROOM SUPERVISOR IS ASST. SE
(C) - SEE FIGURE IN APPENDIX C
** - FPP 0.5 MAY BE INCLUDED
PO - PRIMARY OPERATOR

APPENDIX A

MATRIX 3. TRAINING PROGRAM

• RO - REACTOR OPERATOR • SRO - SENIOR REACTOR OPERATOR • SS - SHIFT SUPERVISOR	(1.A) ENTRANCE EXAMINATION TO SCREEN CANDIDATE FOR TRAINING		(1.B) LENGTH OF TRAINING (MONTHS)		(1.C) SUBJECTS COVERED IN TRAINING		(1.D) TESTING DURING TRAINING		(1.E) PLANT TRAINING (MONTH)		(1.F) SIMULATOR TRAINING (MONTHS)		(1.F.1) SIMULATOR DUPLICATE OF PLANT		(1.G) CASUALTY EXERCISES		(1.H) TIME (%) DEVOTED TO EMERGENCY OFF NORMAL CONDITIONS VS ROUTINE OPERATION (MONTH)		(1.I) TRAINING PROGRAM APPROVAL BY GOVERNMENT REGULATORY ORGANIZATION		(2) QUALIFICATIONS REQUIREMENTS FOR TRAINING PROGRAM INSTRUCTOR	(3) ARE UTILITIES REQUIRED TO MAINTAIN PERMANENT TRAINING STAFF	
	RO		RO	SRO	RO	SRO	RO	SRO	RO	SRO	RO	SRO			RO	SRO	RO	SRO					
	YES	NO											YES	NO					YES	NO		YES	NO
ARGENTINA	•		36	36	TECH. & SYS.	TECH. & SYS.	YES	YES	12	12-24	NO_{NR}	NO_{NR}		•	YES	NO	20%	20-30%	•		PROFESSIONAL INSTRUCTORS NPP EXPERIENCED SHIFT SUPERVISORS	•	
BELGIUM	$•_U$		24	NI	TECH. & SYS.	TECH. & SYS.	YES & FINAL	NI	3	NI	YES	YES		•	NO	NO		NI	•		NPP EXPERIENCED OPERATOR		$•_{NR}$
BRAZIL	$•_U$		ND	ND	TECH. & SYS.	TECH. & SYS.	YES & FINAL	YES & FINAL	6-12	6-12	YES	YES	NR	NR	NR	NR	NR	NR	$•_C$		NR		$•_{NR}$
CANADA	$•_U$		NI	NI	TECH. & SYS.	TECH. & SYS.	NA	YES_{CR}	YES	YES	NA	2	•		YES	YES	NI	NI	•		UTILITY SELECT QUALIFIED INSTRUCTOR NPP EXPERIENCED OPERATOR	•	
FINLAND - R.F		•	12-24	NA	ACAD REVIEW & TECH./SYS.	NA	YES	NA	2	4	1_T	1_T	$•_Q$		YES	NA	20	NA	•		NR		•
FRANCE	•		6	12_E 36_W $12_{A/S}$ 2 SV	TECH./SYS. OPERATIONS & PROCEDURES TECH SPECS EMERGENCY PLAN		FINAL EVALUATION		YES	YES	YES	YES	•		YES	YES	$30\%_S$ $15\%_{NPP}$		•		NR-RECRUIT PROFESSIONAL INSTRUCTORS BASED ON TECHNICAL EXPERTISE	•	
FEDERAL REPUBLIC OF GERMANY	NR		48-72	48-72	TECH. & SYS. PRACTICAL	TECH. & SYS. PRACTICAL	YES FINAL	YES FINAL	$36\text{-}48_{(*)}$	$36\text{-}48_{(*)}$	1	1	$•_Q$		YES	YES	NO BALANCE PROGRAM NUCLEAR BASICS & PRACTICAL		$•_{(**)U}$		PROFESSIONAL OR EXPERTISE IN OPERATIONS NPP	•	
INDIA	•		24	NI	TECH. & SYS.	TECH. & SYS.	FINAL GOVT	NI	YES	YES	YES	NI	NI	NI	YES	NI	COVERED	NO	NI	NI	NPP, SRO OR RO	•	
ITALY	$•_U$		24	NI	TECH. & SYS.	NI	FINAL GOVT	NI	YES	YES	YES	NI		•	YES	NI	NI	NI	•		ASSIGNED TO SRO		•
JAPAN		•	5	0.5	TECH. & SYS.	TECH. & SYS.	YES	NO	YES	YES	YES	YES	•		NO	NO	60%	NI		•	DELEGATED BY UTILITY OR MANUFACTURER		$•_{NR}$
KOREA		•	7	NI	TECH	NI	YES	NI	NO	NO	YES	NI		•	YES	NI	50%	NI	•		NPP EXPERIENCED IN OPERATION WITH BS DEGREE		$•_{NR}$
MEXICO		•	ANSI/ANS 3.1		NI	NI	NI	NI	YES	YES	YES	NI	NI	NI	YES	NI	NI	SI	•		ANS/ANS 3.1 1978	•	
THE NETHERLANDS - D		•	12	NI	NUREG 0094 CHAPTER 4	NUREG 0094 CHAPTER 5_D	NR	NR_D	YES	YES	1	0.75		•	YES	NI	20%	40%	•		NPP EXPERIENCED IN OPERATIONS		•
SPAIN	•		12	12	TECH. & SYS.	TECH. & SYS.	YES	YES	36	36	3_{NR}	3_{NR}	$•_Q$	$•_Q$	YES	YES	24%	24%	•		PROF. INSTRUCTORS NPP EXPERIENCED	$•_{NR}$	
SWEDEN		•	20	NA	REVIEW ACAD & TECH.	NI	YES	NA	9-24	NA	2	NA	$•_Q$		YES	NA	50%		•		TECHNICALLY TRAINED NPP OPERATING EXPERIENCE	•	
SWITZERLAND	$•_U$		8	2	TECH. & SYS.	NI	YES	NI	YES	YES	YES	NI	•		YES	NI	40-50%	NI		•	NPP OPERATING EXPERIENCE	NI	NI
TAIWAN	•		36	36	TECH. & SYS.	TECH. & SYS.	YES	YES	YES	YES	YES	YES	$•_Q$		YES	YES	10%	10%	•		LICENSED SRO	•	
UNITED KINGDOM - UK		•	ND	12_{SRO} 30_{SS}	TECH. & SYS.	TECH. & SYS.	NI	NI	YES	YES	1.2	0.5	$•_Q$		YES	YES	ND	ND	$•_U$		SPECIALITY QUALIFIED INSTRUCTORS NPP EXPERIENCED OPERATORS DEGREE	•	
UNITED STATES		•	ND	ND	TECH. & SYS.	TECH. & SYS.	YES	YES	3	3	YES_{NR}	YES_{NR}	$•_C$		ND	ND	ND	ND	•		SRO		•

FOOTNOTES
U - UTILITY RESPONSIBILITY
ND - NOT DEFINED
NR - NOT REGULATORY REQUIREMENT
NI - NO INFORMATION PROVIDED
NPP - NUCLEAR POWER PLANT
NA - NOT APPLICABLE
E - ASSISTANT RO EXPERIENCE
W - WITHOUT PWR EXPERIENCE
A/S - ASSISTANT SHIFT SUPERVISOR
1. NO EXPERIENCE
2. SRO EXPERIENCE
Q - QUALIFIED "YES" (NOT ALL SIMULATORS ARE REPLICA)
C - REGULATORY AUTHORITY APPROVES TRAINING PROGRAM IN S.A.R.
R - FOR REPLACEMENT PERSONNEL
T - TEAM OF OPERATOR FROM SAME STATION
CR - CONTROL ROOM OPERATOR (CRO) EQUATE TO SRO
F - NO SENIOR REACTOR OPERATOR (SRO) BUT EQUIVALENT TO SHIFT SUPV. IN FINLAND
D - SRO SS
SV - SHIFT SUPV. PROMOTED FROM ASSISTANT SHIFT SUPV.
S - SIMULATOR
(*) - INCLUDES PRACTICAL EXPERIENCE
(**) - BASICS (PRACTICAL)
UK - 1. SHIFT ENGINEER IN CHARGE (SCE) SS
2. CONTROL ROOM SUPERVISOR IS ASST. SCE

APPENDIX A

MATRIX 4. INITIAL OPERATOR'S LICENSE OR CERTIFICATION

• RO - REACTOR OPERATOR • SRO - SENIOR REACTOR OPERATOR • SS - SHIFT SUPERVISOR	(I) RO SRO SS LICENSED BY REGULATORY AUTHORITY EXAMINATION		(IA) LEVEL OF UTILITY MANAGEMENT CERTIFYING PERSON READY FOR EXAMINATION	(IB) IS UTILITY REQUIRED TO TEST REACTOR OPERATOR PRIOR TO REGULATORY LICENSE EXAMINATION?		(IC) RE EXAMINATION AFTER FAILURE (MONTHS)			(ID) TYPES OF EXAMINATIONS			(IE) DOES EXAMINATION COVER RESPONSE TO EMERGENCY SITUATIONS?		(IF) REGULATORY EXAMINERS QUALIFICATIONS			(IG) PASSING CRITERIA FOR REGULATORY EXAMINATIONS	(IH) ACADEMIC SUBJECTS ON REGULATORY EXAMINATION	(II) KNOWLEDGE OF PLANT OPERATING PROCEDURES IN EXAMINATION		(IJ) ARE RESULTS OF REGULATORY EXAMINATION PASSED TO UTILITY FOR RETRAINING?		(IK) PERIOD OF LICENSE	
						FAILURES																		
	YES	NO		YES	NO	1ST	2ND	3RD	WRITTEN	ORAL	PRACTICAL	YES	NO	PROFESSIONAL AND ENGRS	LICENSED RO SENIOR MGR	BOARD OR COMMITTEE			YES	NO	YES	NO	YRS	EXPIRATION
ARGENTINA	•		• DIRECTOR, CHIEF OF PRODUCTION • CHIEF RADIO PROTECTION & SAFETY • CHIEF OF OPERATORS	•		3	12	24	•	•	NI	•		•		•	DEMONSTRATE ADEQUATE THEORETICAL & PRACTICAL KNOWLEDGE	BASIC NUCLEAR PHYSICS REACTOR PHYSICS RADIOLOGICAL PROTECTIONS NUCLEAR SAFETY SPECIFIC TYPES REACTORS & SUB SYSTEMS	•		NI	NI	2.5	
BELGIUM		*NR	CERTIFICATION BY PLANT SUPERINTENDENT OR UNIT MANAGER BY COMMITTEE		*NR	NR	NR	NR	•T	•T	NI	•T		NR	NR	NR	NR	OVERALL KNOWLEDGE NPP OPERATIONS	•		NR	NR	NI	NI
BRAZIL	•		THERMAL POWER GENERATION SUPERINTENDENT	•		2	6	24	•	•	•	•				*NRC	70%	SIMILAR TO USNRC REQUIREMENTS	•		•G		2	
CANADA	•		STATION MANAGER	•		4 8	NI	NI	•			•		•	•		80%	NPP FPP RP	•		•G			NONE
FINLAND F	•		STATION MANAGER OR OPERATIONS MANAGER	•		2	A	A	•	•	•	•		•		•	BY EXAMINERS	NPP SYS & OPERATIONS REGULATIONS	•		•G		2	
FRANCE *		•	STATION MANAGER$_{NR}$	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	2$_{NR}$	
FEDERAL REPUBLIC OF GERMANY	*R		PLANT MANAGER		•	2	ND	ND	•	•	•	•		•		•	75% PARTLY 85%	ACCORDING TO REGULATORY GUIDELINES	•		•		ND	1.2½$_E$
INDIA	•		TRAINING SUPERINTENDENT		•	3	NI	NI	•	•	*A	•			•	•	70%	PHYSICS, PLANT SYSTEMS, RADIATION PROTECTION	•		•			NONE
ITALY	•		PLANT MANAGER		•	NR	NR	NR	•	•	•	•		•		•	BY TECH COMMISSION	DESIGN & OPERATIONS, PROCEDURES & RADIATION CONTROL	•			•	3	
JAPAN	•		STATION SUPERINTENDENT		•	A	A	A	•	•	•	•		•		•	BY EXAMINERS	OPERATIONS S DECISION MAKING	•		•		3	
KOREA	•		PLANT SUPERINTENDENT		•	3$_B$	12$_C$	NI	•	•	•	•		•		•	80%	REACTOR THEORY, DESIGN CONSTRUCTION, CONTROL OPERATIONS, RADIATION PROTECTION AND MATERIAL HANDLING	•			•	2	
MEXICO	•		MANAGEMENT OF UTILITY OR DESIGNATED PERSON	•		NR	NR	NR	•		•	•		NR	NR	NR	NR	PER 10 CFR 55.21 & 55.22 PLUS THERMODYNAMICS, FLUID MECH & HEAT TRANSFER	•		NR	NR	2	
THE NETHERLANDS	*P		NPP SUPERINTENDENT		•	NR	NR	NR	•	•		•		NR	NR	NR	80%	PER NUREG 0095, PG 4.11	•		•		2	
SPAIN	*P		UTILITY MANAGEMENT NPP DIRECTOR OR PRODUCTION MANAGER	•		3	NR	NR	•	•	•	•		•		•	70% OVERALL 60% EACH CATEGORY	REACTOR THEORY OPERATING REGULATIONS DESIGN & OPERATING PROCEDURES RADIOLOGICAL PROTECTION AND MATERIAL HANDLING	•		•		2	
SWEDEN		•	UTILITY PLANT SUPERINTENDENT RESPONSIBLE FOR COMPETENT OPERATORS$_{NR}$	NR	NR	NR	NR	NR	•T	•T	•T	•T		NR	NR	NR	UTILITY/GOVT TRAINING PROGRAMS	TECHNICAL SUBJECTS	•		NR	NR	2$_S$	
SWITZERLAND	•		PLANT AND UTILITY OPERATIONS MANAGERS		•	5	NI	NI	•	•	•	•		•		•	DEMONSTRATE ADEQUATE KNOWLEDGE	NPP S[illegible] & OPERATIONS RP & REGULATIONS	•		•		1 & 2	
TAIWAN	•		PASS UTILITY'S CERTIFICATE EXAMINATION	•		2	6	24	•	•	•	•		*BWR PWR S			70%	REACTOR THEORY AND REACTOR OPERATIONS	•			•	2	
UNITED KINGDOM		*NR	STATION MANAGER$_{NR}$	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
UNITED STATES	•		VICE PRESIDENT FOR OPERATIONS		•	2	6	24	•	•	•	•		ND	ND	ND	80% OVERALL 70% EACH CATEGORY	• HEAT TRANSFER & FLUID MECHANICS • THEORY OF FLUIDS & THERMODYNAMICS • REACTOR THEORY • RADIOLOGICAL PROTECTION	•		•		2	

FOOTNOTES:
A - NO TIME LIMITS INDICATED
B - CANNOT APPLY IN NEXT THREE MONTHS AFTER FAILURE
C - TWO FAILURES CONSECUTIVELY
T - DURING TRAINING
NRC - TRAINED BY US NRC
NC - NOT SPECIFICALLY DEFINED

NI - NO INFORMATION PROVIDED
NR - NOT REGULATORY REQUIREMENT
NPP - NUCLEAR POWER PLANT
* - SEE APPENDIX C FRANCE PART IV
S - SIMULATOR
G - FEEDBACK GENERAL COMMENTS

Y - ABSENCE > 24 MONTH MUST BE GIVEN SAME TRAINING AS BEGINNER WITHOUT NUCLEAR POWER PLANT OPERATING EXPERIENCE
FPP - FOSSIL POWER PLANT
RP - RADIATION PROTECTION
P - PROPOSED REGULATION

F - NO SENIOR REACTOR OPERATOR (SRO) BUT EQUIVALENT TO SHIFT SUPERVISOR IN FINLAND
SW - RO AND SRO ONE OF THE SAME IN SWEDEN
R - USE SIMILAR PROCEDURE
E - IN CASE OF JOB INTERRUPTION

APPENDIX A

A-4

MATRIX 5. OPERATOR RETRAINING

• RO - REACTOR OPERATOR • SRO - SENIOR REACTOR OPERATOR • SS - SHIFT SUPERVISOR	(1) FORMAL RETRAINING PROGRAMS		(1A) LENGTH OF RETRAINING MONTHS		(1B) ACADEMIC SUBJECTS IN RETRAINING PROGRAM	(1C) FREQUENCY OF LECTURES (INTERVALS) YRS.		(1D) PERFORM OPERATIONS ON REACTORS, E.G., START UPS, SHUTDOWNS, & RESPONSE TO TRANSIENTS		(1E) CASUALTY DRILLS (FREQUENCY)				(1F) EXAMINATION				(1G) WEEKS PER YEARS DEVOTED TO RETRAINING			(1H) RETRAINING PROGRAMS REQUIREMENT OF REGULATORY AUTHORITY	
	RO	SRO	RO	SRO		RO	SRO	YES	NO	YES	NO	PLANT	SIMULATOR	YES	NO	FREQUENCY YRS	WHO ADMINISTERS EXAM?	RO	SRO	SS	YES	NO
ARGENTINA	YES	YES	DURATION VARIABLE	NI	NPP SYSTEMS RADIATION SAFETY	YES	YES	•			•			•		2	NI	SEE (1)			•	
BELGIUM	YES	YES	NI	NI	NUCLEAR PHYSICS NPP SYSTEMS CHEMISTRY, ELECTRICITY	1	NI	$•_S$		•		NI	NI	•		2	OPERATIONS MANAGER	6	6	6	$•_P$	
BRAZIL	YES_R	YES_R	24	24	NPP SYSTEMS RADIATION PROTECTION PROCEDURES	2	2	•			•	NR	2	•		AFTER EACH STEP	U	NR	NR	NR	•	
CANADA	NA	$YES_{NR,CR}$	NA	0.5_{CR}	NPP SYSTEMS RADIATION PROTECTION PROCEDURES	NA	2	NI	NI	NI	NI	NI	•		•	NI	NI	NA	$2\text{-}3_{CR}$	2-3		•
$FINLAND_F$	YES	NA	0.5	NA	NORMAL, ABNORMAL OPERATIONS	1	NA	$•_S$		•		1	1	•		2	U	2	NA	2	•	
FRANCE	YES	YES	5 DAYS/YR_S 1-4 DAYS EVERY 5 YRS_{NPP}		• REVIEW ON SIMULATOR (E.G., STARTUP & SHUTDOWN, INCIDENTAL & ACCIDENT) • ON SITE -SIGNIFICANT INCIDENTS & ACCIDENTS OF ALL NPP PER REQUEST OF SUPV. -OPERATING PROCEDURES -STANDARD SYSTEMS NPP	AS REQUESTED BY SUPV		•		NR	NR	•	•		•	CONTINUOUS EVALUATION	MANAGEMENT STAFF	5	5	5	NR	NR
FEDERAL REPUBLIC OF GERMANY	YES	YES	0.6	0.6	NUCLEAR PHYSICS NPP SYSTEMS PROCEDURES	$•_{*1}$	$•_{*1}$	•		•		•	•		•	NR	NR	2-3	2-3	2-3	•	
INDIA	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
ITALY	NR	NR	0.25_S	NI	ABNORMAL OPERATING CONDITIONS	NR	NI		•	•			•		•	NI	NI	1	1	1		•
JAPAN	YES	YES	0.5	0.5	REACTOR THEORY PLANT CHARACTERISTICS	1-2	NI	$•_S$		•		•	•		•	NI	EVAL BY ADMIN	1-2	1-2	1-2		•
KOREA	YES	YES	0.5	NI	NUCLEAR REACTOR PHYSICS HEALTH PHYSICS THERMODYNAMICS	1	NI	•		•			•		•	NI	NI	?	NI	NI	•	
MEXICO	YES	YES	24	NI	THERMODYNAMICS HEAT TRANSFER FLUID MECHANICS	NR	NI	•		NR	NR	NR	NR	•		NI	UTILITY ADMIN EXAM	NR	NR	NR	•	
THE NETHERLANDS $_D$	YES_P	YES_P	0.5_P	0.5	NO ACADEMIC SUBJECTS	YES_{ND}	YES_{ND}		NR		•				•		U	2	2	2	?	
SPAIN	YES	YES	0.7	0.7	RADIATION PROTECTION NPP SYSTEMS PROCEDURES	YES_{ND}	YES_{ND}	NR	NR	•		NR	•	•		1	TRAINER	2-3	2-3	2-3	•	
SWEDEN	YES	NA	0.5	NA	NORMAL/ABNORMAL OPERATIONS	1	NA	$•_S$		•			•		•	NI	NI	2-3	NA	2-3	•	
SWITZERLAND	YES	YES	0.5	0.5	NORMAL/ABNORMAL OPERATIONS NPP SYSTEMS	SEE (1A)	SEE (1A)	•		•		•	•	•		2.5	EVAL BY	2	?		•	
TAIWAN	YES	YES	0.5	0.5	NSSS VIDEO TAPES	MTHLY	MTHLY	$•_S$		•		•	•	•		NI	NI	2	2	2	•	
UNITED KINGDOM	YES	YES	0.25	0.25	NUCLEAR PHYSICS NPP SYSTEMS	NR	NR	NR	NR	•		•	•	•		NR	NR	NR	NR	NR	NR	NR
UNITED STATES	YES	YES	24_{MAX}	24_{MAX}	HEAT TRANSFER FLUID FLOW THERMODYNAMICS OPERATIONS	YES_{ND}	YES_{ND}	•			•			•		1	U	NO			•	

FOOTNOTES
P - PROPOSED REGULATION
R - OPERATOR REQUALIFICATION
S - SIMULATOR
U - UTILITY RESPONSIBILITY
D - SRO SS
NI - NO INFORMATION PROVIDED
ND - NOT SPECIFICALLY DEFINED
NR - NOT REGULATORY REQUIREMENT
NPP - NUCLEAR POWER PLANT
NA - NOT APPLICABLE
NSSS - NUCLEAR STEAM SYSTEM SUPPLIER
CR - CONTROL ROOM OPERATOR (CRO) CANADA
F - NO SENIOR REACTOR OPERATOR (SRO) BUT EQUIVALENT TO SHIFT SUPERVISOR
*1 - SEVERAL TIMES/YEAR

APPENDIX A

THE SAFETY-RELATED OPERATOR ACTIONS PROGRAM

Paul M. Haas
Engineering Physics Division
Oak Ridge National Laboratory

Introduction

The Safety-Related Operator Actions (SROA) Program is intended to provide information and data for use by NRC in assessing the performance of nuclear power plant (NPP) control room operators in responding to abnormal/emergency events. The primary effort has involved collection and assessment of data from simulator "experiments" (actually recorded observations of training exercises) and from historical records of abnormal/emergency events that have occurred in operating plants (field data). These data are to be used to develop criteria for acceptability of the use of manual operator action for safety-related functions. The program also has included studies of training simulator capabilities, of procedures and data for specifying and verifying simulator performance, and of methods and applications of task analysis. The program is scheduled to be completed in FY 1983. This paper summarizes the major results of the program to date as well as the plans for completion of the program and the general plans for two related programs which have been initiated.

Simulator and Field Data

The initial impetus for the SROA program was the need for data to assess proposed design criteria[1] for the choice of manual versus automatic action for completion of safety-related functions during design basis accidents. After a preliminary assessment of available data,[2] a program of data collection during "quasi-controlled" exercises was initiated in March, 1980. A parallel program was initiated to collect field data which could be used to "calibrate" simulator results. The approach taken in the proposed design criteria was that if the designer chose to rely on manual operator action, he had to allow certain time margins, depending on the severity of the event, complexity of actions, etc. If those time margins were not available, the actions should be automated. Consequently, the emphasis in the SROA program has been on collecting data on the time required for operators to take correct action, despite the recognition that a more comprehensive approach to allocation of functions is desired and that other measures of performance may be equally or more important in many cases. This simple approach was felt to be reasonable for interim use in a design standard until some basic changes are made in the approach to NPP control room design and a more comprehensive research and data base exists.

Data on operator response times, i.e., the time from the activation of an alarm or observable cue until the time of initial correct operator action, have been reported for a series of preliminary simulator exercises on a pressurized water reactor (PWR) simulator[3] and boiling water reactor (BWR) simulator.[4] A report on a more extensive series of exercises recently completed is in preparation. Response times are quite variable but tend to be correlated more to "operational" characteristics of the event, e.g., how rapidly it develops and how specifically it is annunciated, than to the severity of the event. However, there is obviously a question of the possible effects of stress during an actual event which is not reproduced in the simulator. Initial comparison of field data to simulator data[5] suggests that for highly experienced operators, response times in the simulator will be "on the average" considerably less (as little as one-sixth to one-seventh) of typical response times in the field. Very limited data presented at a previous Water Reactor Safety Meeting[6] indicated that response times on the simulator for inexperienced operators (trainees ready for operator licensing examinations) may be greater than typical of field data.

Some preliminary data is available on operator error rates estimated from simulator exercises and on the apparent impact on response time of some of the important performance shaping factors. Of the performance shaping factors considered, the only one identified as having a statistically significant affect on performance is overall plant experience.

Methods and Applications of Task Analysis

During FY 1981 a pilot study on task analysis methods was conducted as part of the SROA program to provide input to NRC in planning and conducting its current program of Task Analysis of NPP Control Room Crews. The study,[7] which included task analysis of operator response to four specific PWR accident sequences, developed a structure and methodology for a global task analysis that was subsequently modified for use in the more extensive NRC effort. It also demonstrated the use of simulator data to supplement and validate task analytic results obtained from traditional sources - operating procedures, interviews, systems documentation, etc.

In FY 1982, the same methodology has been used to conduct a task analysis of ten BWR events that had been examined previously in simulator exercises. The formal task analysis will provide a more objective description of operator behavior in response to abnormal/emergency events than was previously available and, hopefully, more objective data for formulation of a "model" of operator response. The model can provide the desired structure for future data collection and, hopefully, for definition of more comprehensive criteria for safety-related operator actions.

Simulator and Simulator Training Requirements

The program has included two separate but related studies concerned with simulator performance and the use of simulators in training. The first, completed in FY 1980,[8] summarized the then current state-of-the-art of NPP simulation and the use of simulation in NPP operator training. The fundamental conclusion was the need for a more systematic, objective basis for defining and measuring simulator and simulator training requirements.

The second study,[9,10] completed early in FY 1982, focused more specifically on techniques used in nuclear and non-nuclear industries for specification and verification of simulator performance. However, the fundamental conclusions and recommendations of the earlier study relating to the need for a systems approach to definition of training system requirements, including simulator performance requirements, were re-emphasized. A separate program has subsequently been initiated which will adopt well-established procedures for a systems approach to training system development to provide NRC with a model and supportive research base necessary to evaluate industry training programs.

Conclusion

The Safety-Related Operator Actions Program has included a number of separate but related studies concerned with NPP operator performance, task analysis techniques, and the use of simulators in operator training. The program is one of the earlier NRC research programs in the human factors area, having begun prior to TMI-2, and has in some ways "evolved" with the NRC research effort. The central task - development of criteria for safety-related operator actions based on simulator and field data - will be completed in FY 1983 and this will terminate the program as scheduled. A more comprehensive, more structured program of simulator and field data collection will be initiated, and related research elements which originated in this program will be carried out in separate programs.

References

1. American Nuclear Society, "Criteria for Safety-Related Operator Actions," ANSI/ANS-N660, 1981 Draft.

2. Haas, P. M. and T. F. Bott, "Criteria for Safety-Related Nuclear Plant Operator Actions: A Preliminary Assessment of Available Data," NUREG/CR-0901 (ORNL/TM-330), 1979.

3. Bott, T. F., P. M. Haas, E. J. Kozinsky, and D. Crowe, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Pressurized Water Reactor (PWR) Simulator Exercises," NUREG/CR-1908 (ORNL/TM-434), 1981.

4. Beare, A. N., D. S. Crowe, E. J. Kozinsky, D. B. Barks, and P. M. Haas, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Boiling Water Reactor (BWR) Simulator Exercises," NUREG/CR-2534 (ORNL/TM-8195), Draft in printing.

5. Beare, A. N., et al., "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Simulator to Field Data Calibration," NUREG/CR report (expected to be published November 1982).

6. Haas, P. M., "The Safety-Related Operator Actions Program at ORNL," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 29, 1980.

7. Barks, D. B., E. J. Kozinsky, and S. Eckel, "Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors," NUREG/CR-2598 (ORNL/Sub/79-40432/1), May 1982.

8. Jones, D. W., D. K. Baer, and C. C. Francis, "Nuclear Power Plant Training Simulators: Their Use in Operator Training and Requalification," NUREG/CR-1482 (ORNL/TM-395), July 1980.

9. Belsterling, C. A., S. Pandey, J. C. Praskievicz, E. J. Kozinsky, L. Felkins, and P. M. Haas, "Specification and Verification of Nuclear Power Plant Training Simulator Response Characteristics, Part I: Summary of Current Practices for Nuclear and Non-Nuclear Simulators," NUREG/CR-2353, Vol. 1 (ORNL/TM-7985/P1), December 1981.

10. Haas, P. M., D. L. Selby, T. W. Kerlin, and L. Felkins, "Specification and Verification of Nuclear Power Plant Training Simulator Response Characteristics, Part II: Conclusions and Recommendations," NUREG/CR-2353, Vol. 2 (ORNL/TM-7985/P2), May 1982.

THE SAFETY RELATED OPERATOR ACTIONS PROGRAM

HAS INCLUDED TASKS IN THREE INTERRELATED AREAS:

(1) COLLECTION AND ASSESSMENT OF OPERATOR PERFORMANCE DATA (FY 80-83)

- TRAINING SIMULATOR "EXPERIMENTS"
- FIELD DATA COLLECTION
- CALIBRATION OF SIMULATOR DATA
- RECOMMENDATIONS FOR CRITERIA

(2) TRAINING SIMULATOR EVALUATION

- ASSESSMENT OF SIMULATOR CAPABILITY AND USE (FY 79-80)
- SIMULATOR PERFORMANCE SPECIFICATION AND VERIFICATION (FY 81-82)

(3) TASK ANALYSIS

- DEVELOPMENT - PWR PILOT STUDY (FY 81)
- APPLICATIONS - BWR STUDY FOR MODEL AND CRITERIA DEVELOPMENT (FY 82)

SAFETY RELATED OPERATOR ACTIONS

COLLECTION AND ASSESSMENT OF OPERATOR PERFORMANCE DATA

HISTORY

1979 - PRELIMINARY ASSESSMENT FIELD DATA (NUREG/CR-0901)

- COMPREHENSIVE (SYSTEMS) APPROACH REQUIRED
- IF WANT INTERIM DATA USE FIELD-CALIBRATED SIMULATION

1980 - INITIAL PWR EXPERIMENTS (NUREG/CR-1908)

- SEVEN EVENTS, TEN OPERATOR TEAMS
- COLD LICENSE TRAINEES, NON-SITE-SPECIFIC

1980-81 - PWR FIELD DATA COLLECTION (ORNL/SUB-7688/1)

- RESPONSE-TIME DATA, THREE EVENTS
- SUBJECTIVE DATA ON PSF's

1981 - INITIAL BWR EXPERIMENTS (NUREG/CR-2534)

- TEN EVENTS, TWENTY-THREE OPERATOR TEAMS
- REQUAL. CANDIDATES, SOME SITE-SPECIFIC

1981-82 - BWR FIELD DATA COLLECTION (ORNL/SUB IN DRAFT)

- RESPONSE-TIME DATA ON SEVEN EVENTS APPROXIMATELY 120 OCCURRENCES AT FIVE SITES

SAFETY RELATED OPERATOR ACTIONS
COLLECTION AND ASSESSMENT OF OPERATOR PERFORMANCE DATA

FY 1982

SIMULATOR DATA (NUREG/CR DRAFT IN PREPARATION)

- PWR REQUALIFICATION PROGRAM

  SITE-SPECIFIC OPERATORS

  EVENT-BASED PROCEDURES

  14 EVENTS, 188 RUNS ANALYZED

- BWR REQUALIFICATION PROGRAM

  SITE-SPECIFIC OPERATORS

  SYMPTOMS-BASED PROCEDURES

  8 EVENTS, 40 RUNS

- PWR CERTIFICATION TRAINING

  SITE-SPECIFIC OPERATORS

  EVENT-BASED PROCEDURES

  3 EVENTS, 16 RUNS

SIMULATOR-FIELD CALIBRATION (NUREG/CR FINAL DRAFT IN REVIEW)

COMPARISON OF FY 80-81 SIMULATOR DATA WITH LIMITED FIELD DATA

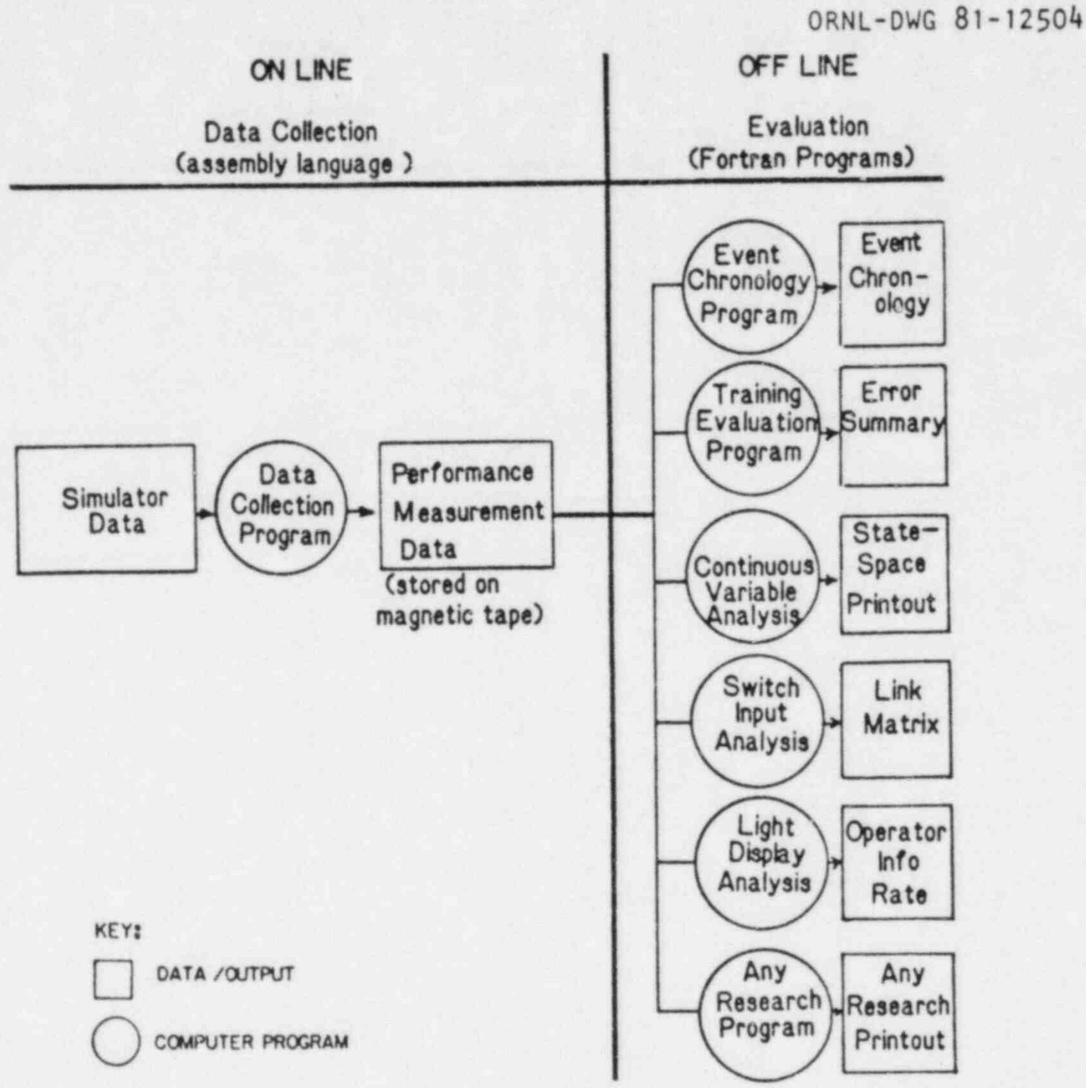


Fig. 1. Structure of the Nuclear Power Plant Operator Performance Measurement System (PMS).

ORNL-DWG 80-18438

TYPICAL PMS OUTPUT

```
                                 START OF LOCA AT    2816.
                     RESETS SAFETY INJECTION AT    3607.
                                TRIPS RCP #2 AT    3845.
                                TRIPS RCP #3 AT    3845.
                                TRIPS RCP #1 AT    3846.
                                TRIPS RCP #4 AT    3846.
               EMPTIES COLD LEG ACCUMULATOR AT    3865.
                  STOPPED REC. CHARGING PUMP AT    3894.
      OPENED RHR HX "A" COOLING VALVE(70-156) AT    4194.
      OPENED RHR HX "B" COOLING VALVE(70-153) AT    4194.
                     RWST LOW LEVEL ALARM AT    4666.
     INITIATES BACKUP TO AUTOMATIC SWITCHOVER AT    4815.
     CLOSED RHR HX "A" CROSSTIE VALVE(74-33) AT    4815.
     CLOSED RHR HX "B" CROSSTIE VALVE(74-35) AT    4815.
              CLOSED PUMP A-A MINIFLOW(63-4) AT    4887.
           CLOSED PUMP B-B MINIFLOW(63-175) AT    4910.
 CLOSED PUMPS "A" & "B" MINIFLOW TO RWST(63-3) AT    4930.
    OPENED RHR HX-B OUTLET TO SI PUMP B(63-11) AT    4954.
OPENED RHR HX-A OUTLET TO CEN. CHG. PUMPS(63-8) AT    4956.
  CLOSED RHR PUMP SUCTION HEADER FROM RWST(63-1) AT    4969.
                            RWST LEVEL @ 8%    4970.
                             OPENED (63-6) AT    5040.
                             OPENED (63-7) AT    5041.
              INITIATES MANUAL SWITCHOVER AT    5347.
          PULLED TO LOCK IN STOP HS-72-10A AT    5347.
          PULLED TO LOCK IN STOP HS-72-27A AT    5349.
  CLOSED RWST TO CEN. CHG. PUMP SUCTION(62-136) AT    5372.
  CLOSED RWST TO CEN. CHG. PUMP SUCTION(62-135) AT    5373.
      CLOSED CSP A-A SUCTION FROM RWST(72-22) AT    5456.
      CLOSED CSP B-B SUCTION FROM RWST(72-21) AT    5456.
         OPENED CS HX "A" ERCW INLET(67-125) AT    5473.
        OPENED CP HX "A" ERCW OUTLET(67-126) AT    5475.
           OPENED CP "B" ERCW INLET(67-123) AT    5478.
         OPENED CP HX "B" OUTLET(67-124) AT    5480.
                            OPENED (72-23) AT    5502.
                            OPENED (72-20) AT    5519.
    STARTED CONTAINMENT SPRAY PUMP A-A(72-27A) AT    5581.
        STARTED CONTAINMENT SPRAY PUMP(72-10A) AT    5583.
                           END OF EXERCISE AT    6738.
```

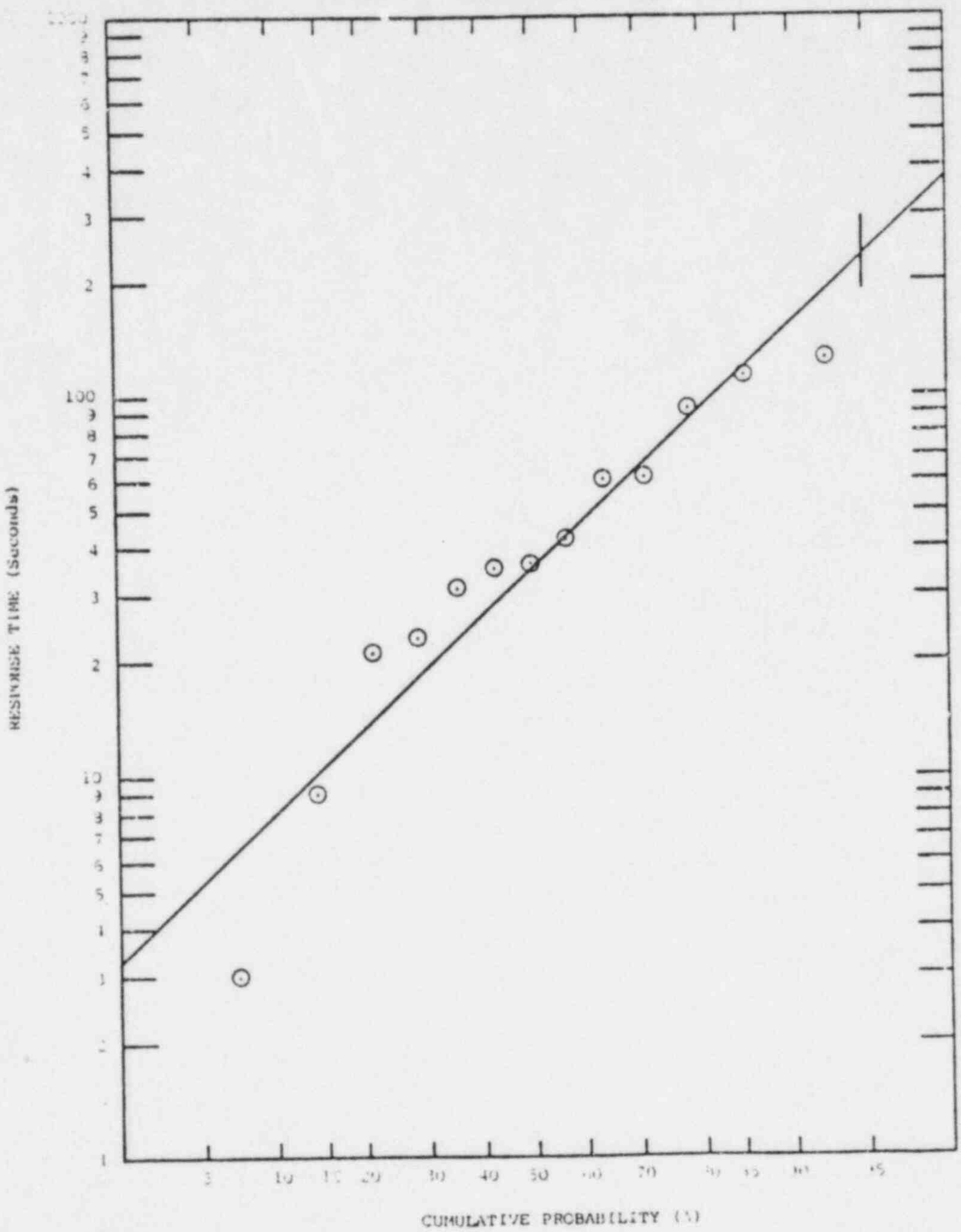
EMPTIES COLD LEG ACCUMULATOR AT 3865

CLOSED PUMP A-A MINIFLOW AT 4887

INITIATES MANUAL SWITCHOVER AT 5347

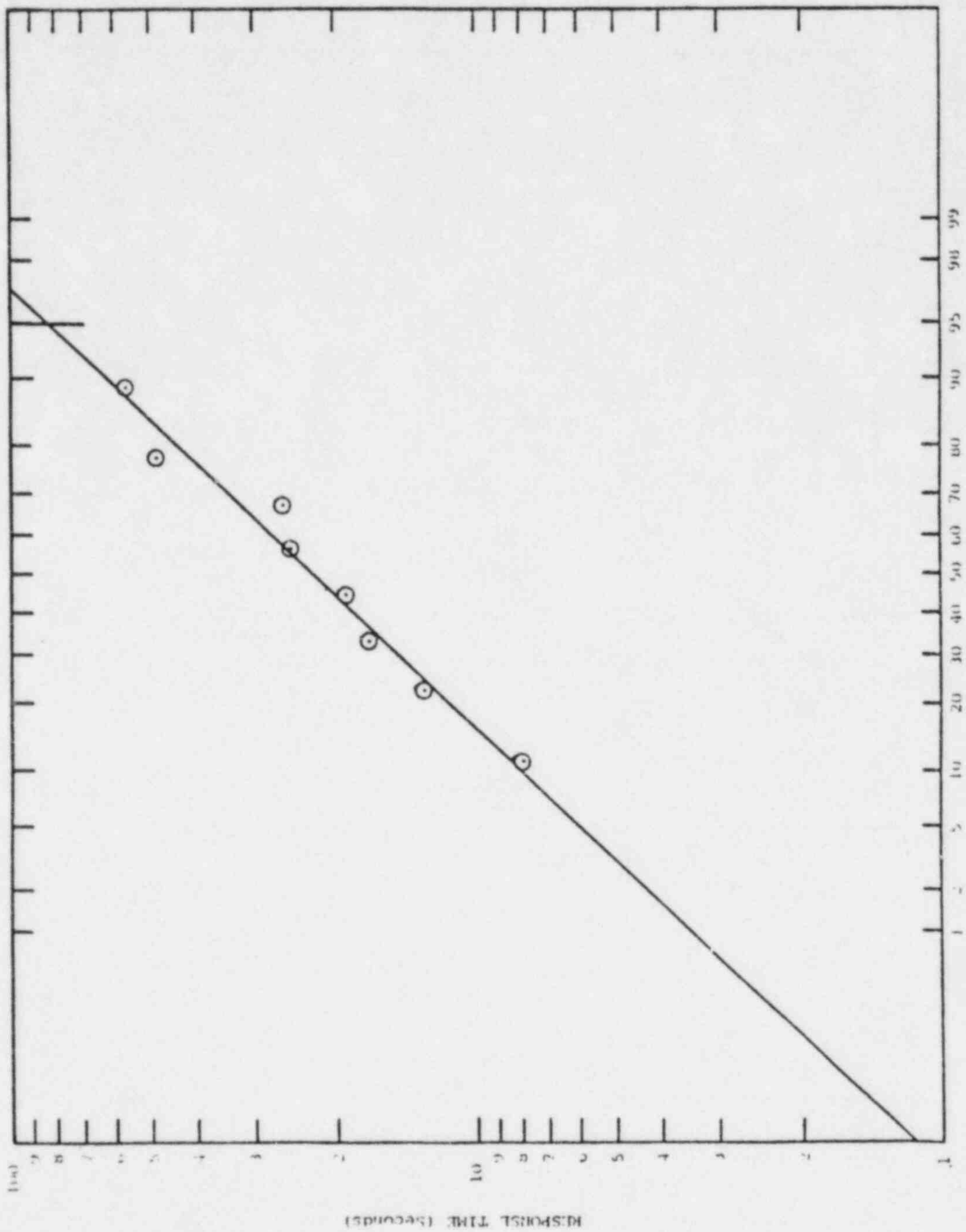
FY1982 PWR

SGTL



FY 1982 PWR

LARGE LOCA



FY 1981 BWR
CONDENSER TUBE LEAK

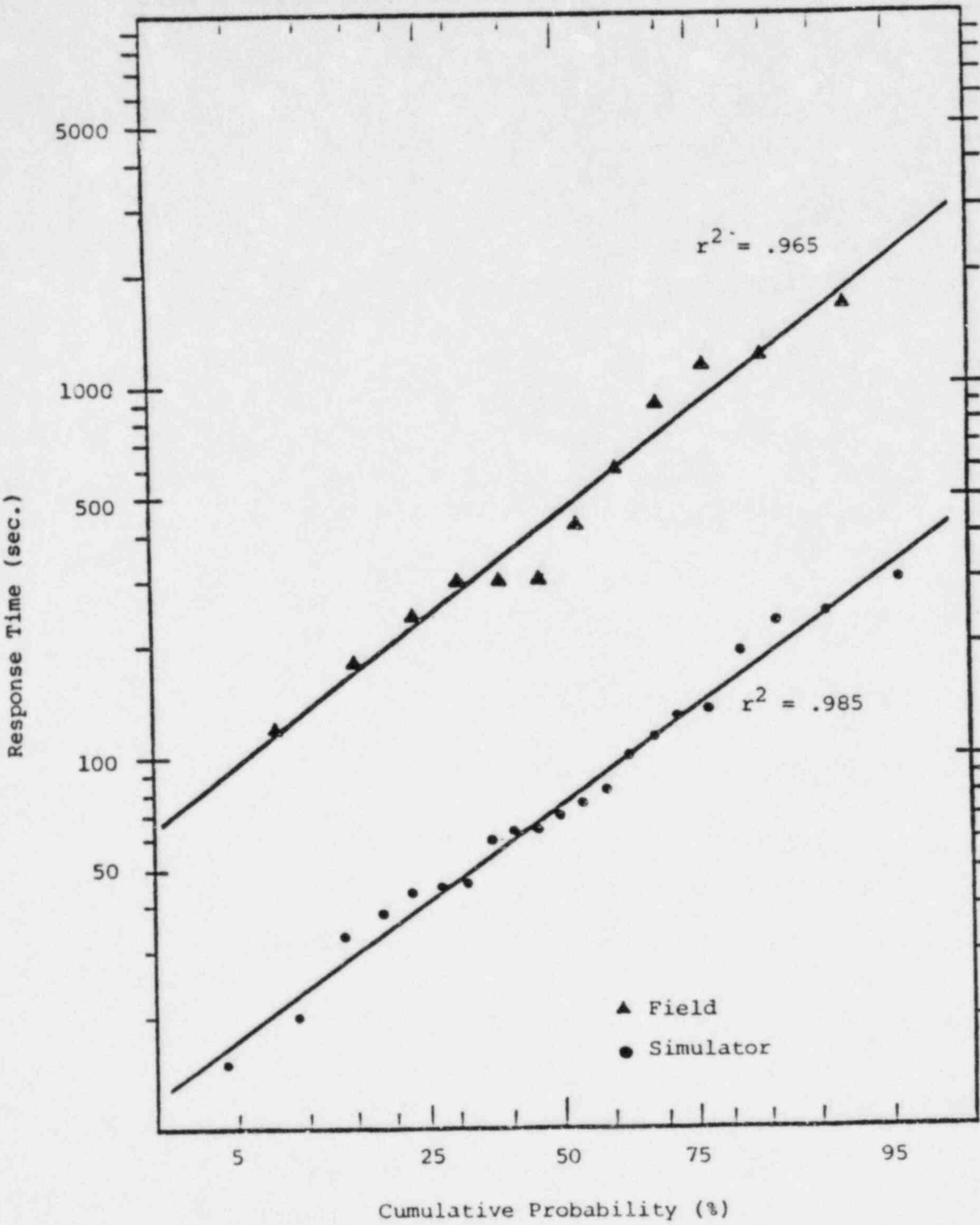


Figure C-6 Field and simulator response times for BWR condenser tube leak

SAFETY RELATED OPERATOR ACTIONS

COLLECTION AND ASSESSMENT OF OPERATOR PERFORMANCE DATA

HIGHLIGHTS OF RESULTS TO DATE

RESPONSE TIMES

- RT's FOR FIRST ACTION NOT SYSTEMATICALLY RELATED TO EVENT SEVERITY BUT TO TASK/EQUIPMENT/SITUATIONAL CHARACTERISTICS
- "ON THE AVERAGE" TRAINEES IN SIMULATOR TEND TO BE COMPARABLE TO OR SLOWER THAN OPERATORS IN THE FIELD; EXPERIENCED OPERATORS IN SIMULATOR TEND TO BE FASTER
- RESPONSES AND RT's HIGHLY VARIABLE

ERROR RATES (ERRORS OF OMISSION ONLY)

- FASTER TEAMS TEND TO MAKE FEWER ERRORS
- EVENTS REQUIRING (ALLOWING) FASTER RESPONSE TEND TO HAVE HIGHER ERROR RATES
- DISTINCT DIFFERENCE FOR "OPERATIONAL" VS. "INFORMATIONAL"
- OVERALL RATE - APPROXIMATELY 7%

PSF's

- INTERNAL - OF AGE, EDUCATION, EXPERIENCE, ONLY EXPERIENCE STATISTICALLY SIGNIFICANT (SMALL SAMPLE PLUS PROBLEM OF INDIVIDUAL VS. TEAM PERFORMANCE)
- EXTERNAL - PROCEDURES A MAJOR FACTOR IN ERRORS OF OMISSION

SAFETY RELATED OPERATOR ACTIONS

PLANS FOR PROGRAM COMPLETION IN FY 1983

- DEVELOP RECOMMENDED STRUCTURE OF CRITERIA FOR NRC TO ASSESS ACCEPTABILITY OF MANUAL VS. AUTOMATIC ACTION

- ASSESS ALL AVAILABLE DATA AND INFORMATION
  - PAST EXPERIMENTS (FY 80, 81, 82)
  - FIELD DATA
  - TASK ANALYSIS
  - SIMULATOR AND FIELD DATA (B7492)
  - SANDIA DATA
  - ORNL ALLOCATION OF FUNCTIONS WORK (B0438)

- DOCUMENT CRITERIA AND DATA BASE (NUREG/CR)

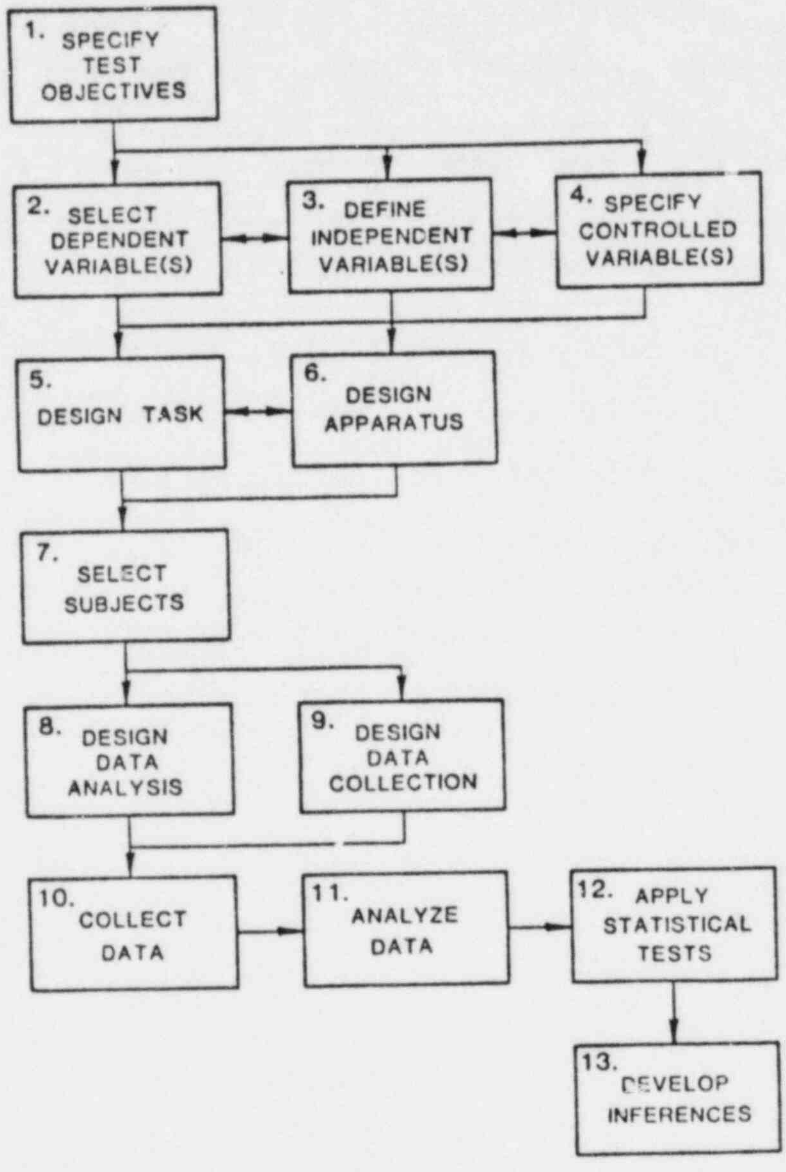
"TRAINING SIMULATOR EXPERIMENTS" (B7492)

(TO BE INITIATED IN FY 1983)

QUASI-CONTROLLED EXPERIMENTS SUPPORTED BY FIELD DATA TO ADDRESS SPECIFIC REGULATORY ISSUES

- SYMPTOMS-BASED VS. EVENT-BASED PROCEDURES
- RELATIONSHIP EDUCATION TO PERFORMANCE
- RELATIONSHIP EXPERIENCE TO PERFORMANCE
- EFFECTIVENESS OF OPERATOR AIDS DURING OFF-NORMAL EVENTS
- CONTROL ROOM STAFFING REQUIREMENTS
- RELATIVE EFFECTIVENESS OF VARIOUS TRAINING DEVICES

HUMAN FACTORS EXPERIMENT DESIGN



A "model" of operator behavior/performance is central to the SROA criteria and to further data collection.

"Measurement . . . is not something which can be employed out of context . . . The measurement, in fact, cannot be better than the description and classification on which it is based."

- Hollnagel & Rasmussen, 1981

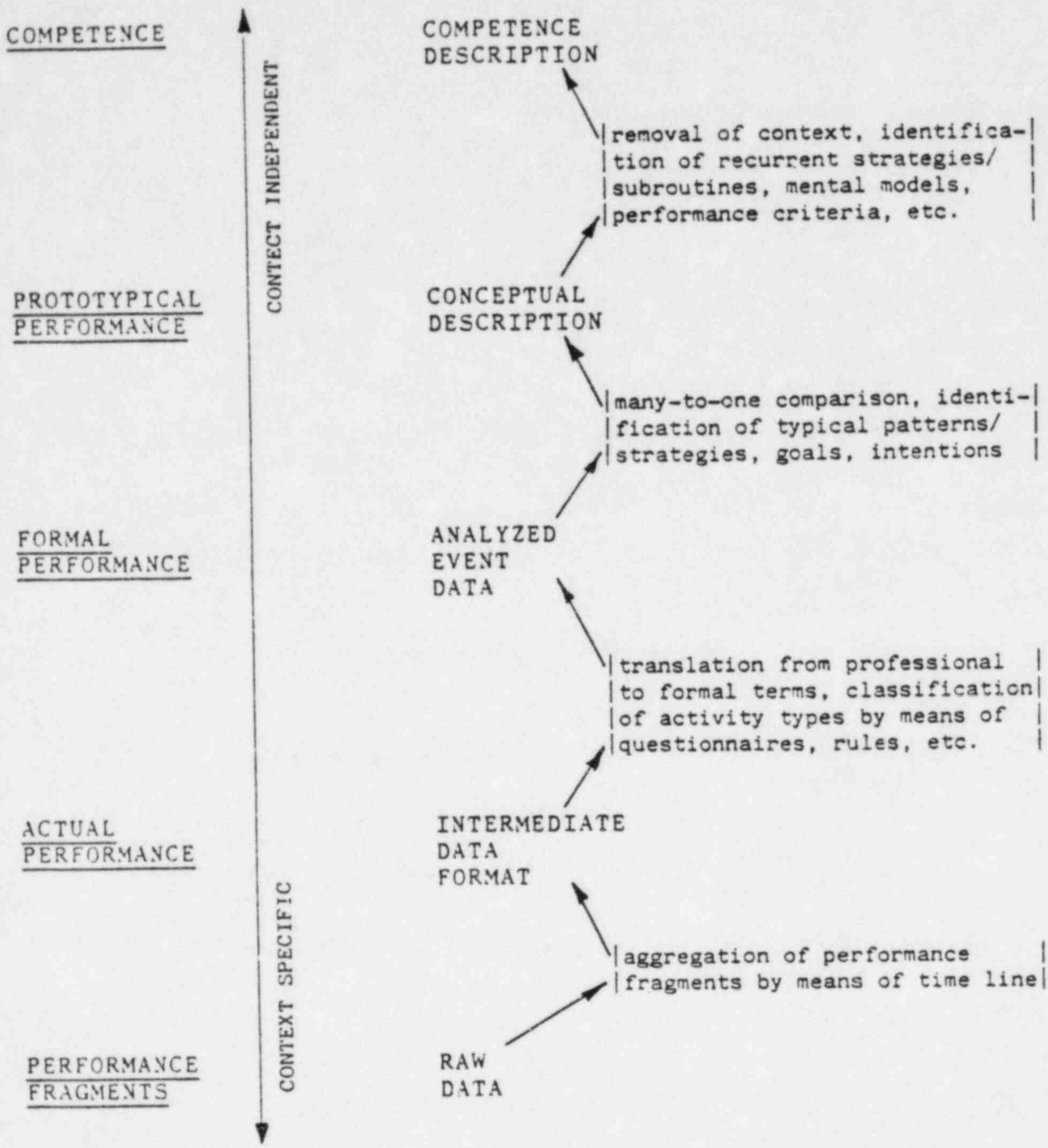


ILLUSTRATION OF THE STEPS
IN THE COMMON ANALYSIS OF DATA

Figure 2.

from Hollnagel & Rasmussen, June, 1981

PROPOSED APPROACH

MODEL/TAXONOMY

- IDENTIFY/CLASSIFY BEHAVIOR OF INTEREST
- IDENTIFY AND RELATE DEPENDENT & INDEPENDENT VARIABLES (HYPOTHESES)

DATA COLLECTION/ASSESSMENT

- SELECT OPERATIONAL SEQUENCES EMBODYING BEHAVIOR OF INTEREST (USE NRC-GPC-BIOTECH SEQUENCES & INPO SYSTEMS TAXONOMY AS POSSIBLE)
- ANALYZE SEQUENCES TO DEFINE "CORRECT" (PROTOTYPIC) RESPONSE
  - REQUIRED PERFORMANCE
  - SPECIFIC MEASURES OF PERFORMANCE (DEF. OF ERRORS)
  - INFORMATION ON PSF's
- COLLECT FIELD DATA DOCUMENTING DEVIATIONS FROM PROTOTYPIC RESPONSE
- DESIGN & CONDUCT SIMULATOR EXPERIMENTS WITHIN PRACTICAL CONSTRAINTS, MAKING OPTIMUM USE OF TRAINING PROGRAMS
- MAP CONTEXT - SPECIFIC DATA ONTO TAXONOMY, TEST MODEL

REVISE & EXPAND MODEL/TAXONOMY AS NECESSARY

ITERATE UNTIL HAVE DEVELOPED & VALIDATED MODEL TO LEVEL REQUIRED

MODEL AND STRUCTURE MAY BE USEABLE FOR SROA CRITERIA IN EARLY STAGES

Safety Related Operator Actions

Task Analysis

Background

- PWR "pilot study" (NUREG/CR-2598) preceding NRC/RES task analysis

    1) Demonstrate use of task analysis techniques and investigate data sources for analysis of emergency/abnormal events in NPP's.
    2) Assessed use of simulator and PMS data to supplement and validate traditional task analysis.
    3) Illustrated use of task analytic data base to help address safety related issues of concern to NRC.
    4) Developed "standardized" terminology and computerized data categorization/retrieval procedures for a task analytic data base specific to NPP's.

- Original plan for BWR (FY 1982) work was essentially identical study

- Plans revised to focus more specifically on needs of SROA Program

Safety Related Operator Actions

Task Analysis

FY 1982 Task Analysis Effort (NUREG/CR Draft in Review)

- 10 BWR events (FY 1981 simulator exercises)

- General information on task analysis techniques/applications (extension of PWR study)

- Provide specific task analytic data for more rigorous evaluation of FY 1981 BWR scenarios - simulator and task analysis data mutually supportive

- Provide more objective data to assess and quantify (possibly suggest) candidate behavioral model and taxonomy (several models examined within SAINT structure)

- Support SROA criteria development

SAFETY RELATED OPERATOR ACTIONS

TRAINING SIMULATOR EVALUATION

(1) SURVEY OF NPP SIMULATOR CAPABILITIES AND USE FOR OPERATOR TRAINING AND REQUALIFICATION (NUREG/CR-1482)

MAJOR CONCLUSIONS

- NO OBJECTIVE BASIS FOR SIMULATOR REQUIREMENTS
- NO SYSTEMATIC MEANS FOR DERIVING TRAINING REQUIREMENTS
- SIMULATOR CAPABILITIES NEED SOME UPGRADING, BUT MAJOR IMPROVEMENTS BY MORE EFFECTIVE USE

MAJOR RECOMMENDATIONS

- REQUIRE SIMULATOR TRAINING
- USE SYSTEMS APPROACH TO DEFINE TRAINING AND SIMULATOR REQUIREMENTS
- DEVELOP REGULATORY STRUCTURE

SAFETY RELATED OPERATOR ACTIONS
TRAINING SIMULATOR EVALUATION

(2) SIMULATOR PERFORMANCE SPECIFICATION AND VERIFICATION

- COMPARISON OF NUCLEAR VS. NON-NUCLEAR PRACTICES (NUREG/CR-2353, VOL. I)

- CONCLUSIONS AND RECOMMENDATIONS FOR NRC AND NUCLEAR INDUSTRY (NUREG/CR-2353, VOL. II)

MAJOR CONCLUSIONS

- NRC AND INDUSTRY SHOULD ADAPT SAT METHODS

- NUREG-0696 REQUIREMENTS MAJOR IMPACT ON DATA SOURCES FOR VERIFICATION

- NRC SHOULD BE MORE INVOLVED IN EVALUATION OF PERSONNEL TRAINING SYSTEM

- STANDARDS PROCESS SHOULD SUPPORT SYSTEMS APPROACH

SAFETY RELATED OPERATOR ACTIONS
TRAINING SIMULATOR EVALUATION

(2) SIMULATOR PERFORMANCE SPECIFICATION AND VERIFICATION (CONTINUED)

MAJOR RECOMMENDATIONS

- SAT/ISD
  - COOPERATIVE NRC-INDUSTRY PROGRAM PLAN
  - NRC RESEARCH PLAN
  - USERS GUIDE, JOINT NRC/INDUSTRY WORKSHOPS
  - FULL-SCALE PILOT STUDY
  - INTEGRATE LICENSING AND TRAINING ISSUES
  - SUPPORTING REGULATORY ACTIONS
- SIMULATOR PERFORMANCE(FIDELITY) VERIFICATION
  - DEVELOP/ADAPT COMPREHENSIVE METHODOLOGY
  - COORDINATE PLANS WITH NUREG-0696 IMPLEMENTATION
  - DEVELOP REG STRUCTURE FOR DISSEMINATION AND USE OF DATA
  - CONSIDER NATIONAL SIMULATOR CERTIFICATION TEAM
  - REQUIRE DATA COLLECTION ON SIMULATORS
- NRC INVOLVEMENT
  - "PARTICIPATIVE" ROLE
- SPECIFIC IMPROVEMENTS TO ANSI/ANS 3.5

NUCLEAR POWER PLANT PERSONNEL SELECTION & TRAINING (B0466)

PROGRAM OBJECTIVE

PROVIDE A SYSTEMATIC METHODOLOGY FOR DEFINITION AND EVALUATION OF SELECTION AND TRAINING REQUIREMENTS AND THE RESEARCH BASE NECESSARY TO SUPPORT THE IMPLEMENTATION OF THE METHODOLOGY.

NUCLEAR POWER PLANT PERSONNEL SELECTION & TRAINING (B0466)

REGULATORY ISSUES WHICH WILL BE ADDRESSED

SELECTION REQUIREMENTS - ENTRY LEVEL

- KNOWLEDGE
- SKILLS
- ABILITIES

QUALIFICATION REQUIREMENTS

- EDUCATION
- EXPERIENCE

TRAINING REQUIREMENTS

- MEDIA SELECTION
  - OJT
  - SIMULATOR
  - CLASSROOM
- SITE SPECIFIC VS. GENERIC SIMULATOR
- SIMULATOR FIDELITY REQUIREMENTS
- SIMULATOR EXAMS
- RETRAINING REQUIREMENTS
- COURSE CURRICULUM

NUCLEAR POWER PLANT PERSONNEL SELECTION & TRAINING (B0466)

FY-1982 OBJECTIVE

(MARCH, 1982 - MARCH, 1983)

- ADAPT SAT/ISD TECHNIQUES TO EVALUATION PROCESS
- DEMONSTRATE
- PROVIDE PROGRAM PLAN
- DEVELOP INTERIM CRITERIA FOR SELECTING SIMULATOR MALFUNCTIONS

Safety Related Operator Actions

Conclusion

EARLY NRC HUMAN FACTORS PROGRAM - EVOLVED WITH NRC PROGRAM

PROGRAM HAS INCLUDED BROAD RANGE OF TASKS - HAS LED TO DEFINITION OF OTHER BASIC PROGRAMS

NEED FOR COMPREHENSIVE APPROACH TO EMPIRICAL DATA COLLECTION RECOGNIZED IN FUTURE PROGRAM PLANS

NRC HUMAN FACTORS RESEARCH
ON
NUCLEAR INDUSTRY ORGANIZATION AND MANAGEMENT:
ASSUMPTIONS, OBJECTIVES AND MILESTONES

by

Thomas G. Ryan, Ph.D
Human Factors Branch
Division of Facility Operations
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

BACKGROUND

Throughout the 1970's several safety-related events occurred within the U.S. commercial nuclear power industry. Included among these were the H. B. Robinson, South Carolina emergency instrumentation failure (June 1973), Browns Ferry, Alabama fire (March 1975) and the Three Mile Island, Pennsylvania core melt-down (March 1979). Studies conducted into these and other safety-related incidents, most prominently the Reactor Safety Study (WASH-1400, 1975)[1] and the Report of the President's Commission (1979)[2], concluded that the majority of safety-related incidents could be traced to the human component of the system (management, technical, administrative). It was therefore recommended that human factors research and engineering be integrated into the management, design and operation of U.S. nuclear power plants.

The Nuclear Regulatory Commission (NRC) human factors research program described in the following pages focuses primarily on the functions and roles performed by nuclear industry management, within an organizational context, during power plant design, construction, start-up and operation. More specifically, this organization and management research program is a response to post Three Mile Island action and planning guidance including the NRC Action Plan (NUREG-0660)[3], NRC Long Range Research Plan (NUREG-0740)[4] and Human Factors Society Long Range Plan (NUREG/CR-2833)[5]. These, as well as other Government and industry action and planning documents, call for long-term improvements in on-site and off-site organizations and management to insure safe construction and operation of nuclear power plants during normal and abnormal conditions, and accident situations. While reactor safety is the explicit theme of these planning and guidance documents, the research program described herein also addresses general plant safety and plant security.

PROGRAM ASSUMPTIONS

Until recently, and appropriately, NRC human factors research has been directed at nuclear power plant control room, auxillary operator and sundry other technician and maintenance roles. The organization and management program discussed here is a systematic attempt to build on earlier human factors research, and to extend that work to an organizational framework with primary focus on above shift supervisor level management. The current program derives from the following assumptions:

- An organizational perspective is required to fully understand the structure, operating dynamics and human-related problems attendant in a nuclear power plant, if we are to institute timely and adequate remedial actions.

- Significant nuclear power plant accidents occurring during the past decade (e.g., Browns Ferry fire, Three Mile Island core melt-down) involved organizational as well as individual personnel failures.

- Organizational effectiveness, in fact the very survival of most organizations depends, in large measure, on the actions and behavior of its management.

- Management provides organizational continuity throughout power plant design, construction, start-up and operation.

PROGRAM OBJECTIVES

The objectives of this long-range research program are to analyze and model the nuclear facility from an organizational (systems) perspective with primary focus on management (above Shift Supervisor Level), and to determine what impact both organizational and management factors do have, and can have, on plant and public safety. The program addresses organization and management factors during all phases of power plant development and operation (i.e., design, construction, start-up, operation). Products of this research will include organization and management safety assessment standards (practice-safety outcome sets) and safety enhancement guidelines (revised practices to optimize plant and public safety). Research results will support current NRC licensing, analysis, inspection and enforcement activities, therefore, are confirmatory. Research results will provide a technical base for supporting future NRC rule-making actions in the areas of organization and management, therefore, are also exploratory.

Research is directed toward achieving new and objective safety assessment standards and enhancement guidelines. Safety assessment standards are defined as:

> Diagnostic statements, empirically derived and validated, describing the relationship(s) between cluster(s) of organization and/or management practices and cluster(s) of safety related indicators.

Standards are diagnostic since they describe current organizational and management practices. Each is supported by field validation data demonstrating practice - safety related indicator correlations. The term cluster represents our current supposition that patterns rather than individual practice - safety related indicator sets represent a more realistic perspective on organization and management effectiveness. The term practice refers to any formal or informal activity undertaken by the organization or its management which helps define the organization (e.g., reactor type, reactor location), establishes a policy or procedure (e.g., on-site resources, training and requalification), or carries out an established policy or procedure. Practices (activities) which define the organization and/or establish policies and procedures are considered management practices. Those which carry-out established policies and procedures are considered organization practices provided they involve at least one manager in their execution. Finally, the term safety-related indicator refers to an outcome of one or more organization or management practice symptomatic of general plant safety (e.g., workspace layout, lighting, ambient noise), reactor safety (e.g., reactor integrity, containment integrity, steam system integrity) or plant security (e.g., penetration and/or sabotage countermeasures).

Safety enhancement guidelines are defined as:

> Prescriptive statements, empirically derived and quantitative, describing the potential relationship(s) between cluster(s) of organiztion and management alternative practices and cluster(s) of safety related indicators.

Guidelines are prescriptive since they designate alternative practices, or practice implementation procedures, to achieve the desired safety-related outcome more completely, efficiently, etc. Therefore, guidelines represent advanced organizations and management configurations. Guidelines emerge from enhancement modeling of organization and management functions and roles, and from consensus (expert judgement) rather than field validation.

Safety assessment standards and enhancement guidelines achieved by this research program will support current and future NRC licensing, analysis, inspection and enforcement activities.

- Short-term support (FY 1983-84) is directed toward achieving a technical base for validating, revising and increasing the objectivity of guidelines and standards employed currently by the NRC to license and monitor near-operating and operating nuclear power plants. Included are NUREGs 0731[6] and 0800[7] establishing guidelines for utility management and organizations, and procedures for assessing the degree to which these guidelines have been achieved. Also included are requirements and standards contained in NRC Inspection and Enforcement Manuals, and references, for Plant Operations[8] and Safeguards[9].

- Long-term support (FY 1985-86) is directed toward achieving safety assessment standards to support NRC licensing and inspection activities during power plant design, construction and start-up. Additionally, it is directed toward achieving safety enhancement guidelines for optimizing organization and management during all phases of nuclear power plant development and operation through: allocation of functions and roles, distribution of prerogatives and responsibilities, and development of decision aids and inter/intra organization communication networks.

RESEARCH PROGRAM MILESTONES AND SCHEDULES

Current Research

Organization and management research projects were initiated in June 1982 and October 1982 directed at achieving the short-term (FY 1983-84) goals of the program.

In June 1982, a 24 month contract was awarded to the Battelle Pacific Northwest Laboratory to develop and field validate new, innovative approaches and standards for assessing nuclear facility organization and management effectiveness in matters crucial to plant and public safety for near-operating and operating power plants. This research involves the following tasks and milestones.

- o Literature review to establish nuclear facility organizational, management and safety typologies. That is, to answer the following questions: What are the structures, characteristics and operating dynamics of its organization, management and safety elements? Is the organization open or closed (i.e., how is it influenced by outside forces such as stockholder groups, unions or corporate headquarters)? What are its boundaries? That is, can our research on organization and management be limited to the nuclear power plant level, or must we also address ourselves to either or both the utility and corporate levels? Are there generic organization, management and/or safety typologies, or are several required to account for the power plants operating in the U. S. today? Regarding safety, what are the physical events commonly associated with safety? How are they interrelated? What person-person/ person-machine interactions are involved in these events?

- Reviews of related organizations and their performance assessment programs (e.g., military, commercial aviation, service industries such as police and non-electrical utilities), and current nuclear utility programs based on NUREG 0731 and 0800, and Inspection and Enforcement Procedural Manuals for Operations and Safeguards. The purpose of these reviews is to capitalize in successful approaches and methods which come under the rubric of the organization, management and safety typlogies developed from the earlier literature review.

- Analysis of safety event reports and data (e.g., personnel turn-over, license qualification) developed currently at the power plants, to establish safety-related indicators (performance criteria). Reports and other data are being reviewed for their relevance to safety, reliability, freedom from bias, practicality, measurability and generalizability.

- Classification and grouping of organization and management safety-related practices and indicator sets emerging from earlier work and incorporating same into general approaches for assessing organization and management effectiveness in near-operating and operating power plants. In this context an assessment approach refers to a perception of the organization and its management stucture for the purpose of evaluation. Safety assessment standards are the practice-indicator sets used by the approach.

- o Selection of safety assessment approaches and standards for further development and field validation. The purpose of this task is to select assessment approaches based on their practicality (e.g., implementation cost, generalizability), acceptability to Government and industry and potential for success; develop implementing procedures or protocols; and collect field data to establish the veracity of both the approaches and standards involved.

At the end of the first 12 months of the project (May 1983), the NRC will have developed diagnostic information on: nuclear facility organization, management and safety dynamics, safety-related practices and indicators, practices which have no discernible safety related outcomes, and safety-related indicators not being attended to through current utility practices. This information will be used to support validation, reviews, revisions to NRC licensing guidelines and standards (i.e., NUREGs 0731 and 0800), and Inspection and Enforcement standards involving operating and near operating plants. At the conclusion of the project (May 1984), the NRC will have developed and field validated safety assessment approaches and standards along with user materials for the NRC and the utilities. Figures 1 and 2 describe and display respectively project milestones scheduled for FY 1982-84.

In October 1982, a 24 month contract was awarded to the Idaho National Engineering Laboratory to conduct analyses and establish enhancement modeling requirements for organization and management functions and roles crucial to general plant safety, reactor safety and plant security, during power plant design, construction, start-up and operation. This research involves the following tasks and milestones.

Figure 1

ORGANIZATION AND MANAGEMENT RESEARCH PROGRAM

FOCUS OF ORGANIZATION AND MANAGEMENT PROGRAM:

o ISSUES IDENTIFICATION
  -- ORGANIZATION TYPOLOGIES
  -- MANAGEMENT TYPOLOGIES
  -- SAFETY TYPOLOGIES

o TECHNICAL DATA BASE DEVELOPMENT
  -- FUNCTION ANALYSES
  -- ROLE ANALYSES
  -- ENHANCEMENT MODELING REQUIREMENTS

o TECHNOLOGY DEVELOPMENT
  -- SAFETY ASSESSMENT STANDARDS
  -- SAFETY ENHANCEMENT GUIDELINES

o TECHNOLOGY EVALUATION
  -- CONSENSUS VALIDATION
  -- FIELD VALIDATION

o TECHNOLOGY TRANSFER
  -- PRACTICAL APPLICATION
  -- USER MATERIALS

ORGANIZATION AND MANAGEMENT RESEARCH PROGRAM

SAFETY ASSESSMENT STANDARDS PRODUCTS:

NUCLEAR POWER PLANT LIFE CYCLE

RESEARCH MILESTONES	FY 1982	FY 1983	FY 1984	FY 1985	FY 1986
ISSUES IDENTIFICATION	OPERATIONS	OPERATIONS START-UP CONSTRUCTION	DESIGN		
TECHNICAL DATA BASE DEVELOPMENT		OPERATIONS START-UP	CONSTRUCTION DESIGN		
TECHNOLOGY DEVELOPMENT		OPERATIONS	START-UP	CONSTRUCTION DESIGN	
TECHNOLOGY EVALUATION			OPERATIONS	START-UP	CONSTRUCTION DESIGN
TECHNOLOGY TRANSFER			OPERATIONS	START-UP	CONSTRUCTION DESIGN

Figure 2

- Development of a function inventory (i.e., listing of all major organization and management activities which define the organization, establish policies and procedures, an carry-out established policies and procedures). Each function so identified will be appended the following descriptors: (1) type function (i.e., individual, group), (2) performance requirements (i.e., information gathering, decision making, monitoring), (3) plant life cycle (i.e., design, construction, start-up, operation), (4) setting (e.g., normal/abnormal operation, emergency operation), (5) criticality to safety, (6) parent cluster (i.e., other functions immediately preceeding, following or occurring simultaneous with the function of interest), and (7) personnel involved (e.g., management, technical staff). The purpose of this task is to establish an organizational structure within which to study organization and management functions and roles deemed crucial to safety.

- Detailed analyses of functions and roles selected from the above inventory as being crucial to plant and/or public safety. Subsequent analyses will involve organization and management roles both individual and group engaged in functions crucial to safety. The purpose of this task is to develop a technical base for developing safety assessment approaches and standards for evaluating organization and management effectiveness during power plant design, construction and start-up; and to support enhancement modeling of selected functions and roles to optimize plant and public safety.

o Identification of enhancement modeling requirements to support optimization of organization and management during all phases of nuclear power plant development and operation. The purpose of this task is to identify organization and management functions and roles for modeling, modeling objectives (e.g., time-saving, efficiency, cost-benefit), modeling media (e.g., paper-pencil, computer-based), and modeling limitations (i.e., magnitude of function and role engineering which is practical).

At the end of the first 12 months of this project (September 1983) the NRC will have developed a technical base to support organization and management function engineering. That is, to assess the adequacy of current organization and management allocation of functions, prerogatives and responsibilities, and intra/inter organization communication networks. At the conclusion of the project (September 1984) the NRC will have added a technical base to support role engineering (i.e., assess the adequacy of current organization and management role allocations). Additionally, the NRC will have established modeling requirements for conducting exploratory research on advanced organization and management concepts to optimize plant and public safety. Figures 1 and 3 describe and display respectively project milestones scheduled for FY 1983-84.

FUTURE RESEARCH

FY 1985-86 research will be directed toward achieving valid, reliable safety assessment approaches and standards for evaluating organization and management effectiveness in matters crucial to safety during power plant design, construction and start-up. The research will proceed on the basis of findings and lessons learned during earlier safety

ORGANIZATION AND MANAGEMENT RESEARCH PROGRAM

Safety Enhancement Guidelines Products:

Nuclear Power Plant Life Cycle

Research Milestones	FY 1982	FY 1983	FY 1984	FY 1985	FY 1986
Issues Identification		Operations Start-Up	Construction Design		
Technical Data Base Development			Operations Start-Up	Construction Design	
Technology Development				Operations Start-Up	Construction Design
Technology Evaluation					Operations Start-Up
Technology Transfer					Operations Start-Up

Figure 3

assessment approaches and standards work, and on the basis of technical data derived from the earlier function and roles analyses project. Figures 1 and 2 describe display safety assessment standards research milestones scheduled for FY 1985-86. Related FY 1985-86 research will be directed toward achieving organization and management performance enhancement guidelines. Enhancement modeling and consensus validation (expert judgement) research will be undertaken on selected functions and roles determined crucial to safety and in need of restructuring, for power plant start-up and operation. Figures 1 and 3 describe and display respectively project safety enhancement guidelines milestones scheduled for FY 1985-86.

REFERENCES

1. Reactors Safety Study: An Assessment of Accident Risks in U.S. Commerical Nuclear Power Plants. NRC Re,ort WASH-1400 (NUREG-75/014), 1975.

2. Report of the President's Commission on the Accident at Three Mile Island: The Need for Change, GPO, 1979.

3. NRC Action Plan Developed as a Result of the TMI-2 Accident, NUREG-0660, 1980.

4. NRC Long Range Research Plan, NUREG-0740, 1982.

5. Critical Human Factors Issues in Nuclear Power Regulation and a Recommended Comprehensive Human Factors Long-Range Plan, NUREG/CR-2833, 1982.

6. Guidelines for Utility Management Structure and Technical Resources (Draft), NUREG-0731, 1980.

7. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, 1981.

8. Inspection and Enforcement Manual, Chapter 2515, "LWR Inspection Program - Operations", 1981.

9. Inspection and Enforcement Manual, Chapter 2515, "LWR Inspection Program - Safeguards". 1981.

MANAGEMENT AND ORGANIZATIONAL DESIGN: AN INITIAL LOOK AT A NEW PROJECT

Richard N. Osborn
Jon Olson
Battelle Human Affairs Research Centers

A new project, started this summer, is designed to establish one basis for empirically linking aspects of management and organizational design to the safe operation of nuclear power plants. Current work is focusing on (a) reviewing existing literature relevant to this linkage, and (b) isolating and incorporating numerous safety relevant indicators. Later stages of this project will involve hypothesis development, data collection, and analysis.

Why a new project stressing organizational issues? The most recent impetus stems from DHFS assessments of utilities applying for operating licenses. Draft guidelines incorporating many salient "lessons learned" from Three Mile Island (e.g., NUREG-0731) still yielded approaches to management issues based on primarily subjective criteria and assessment processes (see Osborn et al, 1982). Because of a lack of objective criteria, even skilled reviewers steeped in the history, culture, terminology, technology, and plans of an applicant continue to have difficulty in analyzing proposed management and organization in a systematic and justifiable way (see Nadel, 1982). This is reflected in the fact that management and organization criteria vary substantially across licensees. For instance, some utilities attach maintenance personnel (fossil and nuclear) to a central headquarters unit, some do not. To compound the assessment problem, existing analyses of safety related measures appear quite incomplete (e.g., Howard, 1976). Thus, measurement of safety needs to be investigated prior to launching the empirical work.

Yet, as ill-defined as management and organization criteria may be, many industry executives, regulators, and intervenors appear to agree that management and organization factors are important precursors to safe operations (see Widrig et al, 1980). What is clearly needed is systematic research that can provide empirical support for regulators and industry in their respective attempts to integrate safety concerns into management and organizational design. What aspects are important? Of the important aspects, which can be subject to regulatory influence?

Two important gaps in our current understanding confront any such attempted research. The first gap is the paucity of comprehensive conceptual or empirical studies of management and organizational factors within the industry. Existing studies appear limited by one or more of the following constraints: (1) concern with only one or a few sites, (2) lack of attempts to validate clinical estimates of cause-effect relationships, (3) overly critical judgments where existing conditions have been compared to some ideal, hypothetical, but untested conditions (here all licensees appear deficient), (4) lack of sound conceptual underpinnings, or (5) little, if any, concern for potentially important factors impinging on the plant from the corporate and industry level. These constraints seriously limit the ability to make empirically based regulatory decisions about the relationships among management, organization design, and safety. While no one project can rectify all such deficiencies, a carefully constructed research project can help establish a better conceptual and empirical basis for regulatory judgment.

The unique position of the regulator must be recognized and continually incorporated if research findings in this area are to be translated into effective regulatory tools. This position coincides with a second identified gap in the literature. The plurality of work specific to the nuclear setting focuses on the bottom of the organization where specific operational problems are most directly manifested. Researchers are forced to try to "work up" the organization or across to related individuals, groups, or departments (e.g., building organization charts from staffing information) in their attempts to find causal and mediating factors in the structure of the organization and management practices. Few studies take a holistic or systemic view which starts with the utility or plant. Yet, there are potentially rich sources of published information regarding plants and utilities. Still fewer attempt to combine both published information (e.g., NRC inspection reports) <u>and</u> primary data (e.g., surveys and interviews). A "top down" perspective is needed.

Not only is a "top down" perspective consistent with the role of NRC as a force outside the utility, but side benefits of such an approach can be expected. These include a sensitivity to a wider range of policy issues, constraints on management, and the history of the utility (e.g., Tech Spec changes, safety record during construction), and overall plant, rather than issue specific, performance (cf Osborn et al, 1980).

The "top down" view also raises an important analytical question for future analysis: What should be the unit of analysis in safety studies? The utility as a whole, the portion of the utility dedicated to nuclear operations, the portion specific to a particular plant/facility, and the various operating and staff units within the plant may be salient. The exact boundaries between these "units of analysis" may be difficult to draw (e.g., what part of a centralized maintenance function serving both nuclear and fossil operations should be considered?). Thus, central concerns early in the project include problems of defining (and the implications of alternative definitions) the unit of analysis to be used in explaining and predicting safety. There is a very real possibility that the appropriate unit of analysis may depend on the exact safety issue in question, and that the issue may not conform to convenient distinctions such as operations versus engineering or onsite versus offsite pressures (e.g., NUREG-0731).

So far, the report has highlighted the difficulty of assessing management and organizational factors, the potential importance of these factors for safety, and two important gaps in the literature (few published studies of utility management and organization and who/what is the appropriate unit of analysis for investigation). Prior to the empirical work, however, meaningful indicators of safety need to be developed.

We have started to identify (1) some conceptual foundations for defining safety, and (2) readily available indicators of safety. Current conceptual development has focused on plant/facility concerns and selected safety issues. (This will be the subject of a later report.) Table I shows the data sources already identified. For some indicators (e.g., LER's), previous work suggests a number of limitations and some potential if each is used in conjunction with other information (e.g., Howard et al, 1976; Chakoff, 1978; Conver, 1978; NUREG-0572; NUREG-0834). The current strategy presumes that the development of accurate, reliable indicators of safety should recognize both basic and applied concerns. Obviously, the development of measures of the relative "safeness" of operating plants/facilities is the most sensitive area to be raised in this project. Work to date suggests that even with the inclusion of new indicators and methods (e.g., morale, public stress, Probablistic Risk Assessment) we may fall short of capturing the varied meanings of safety. Consequently, any one set of readily available indicators may be inappropriate for measuring safety.

Current work has identified some generic problems with combining potential indicators. For instance, indicators may or may not move together to provide a consistent picture of safety (e.g., INPO evaluations, SALP reports, and the number of LER's). The who, what, how, and timing questions of measurement may account for some of the differences (e.g., 1980 utility reported deviations from Tech Specs versus INPO evaluations based on clinical observations versus SALP ratings). Some indicators may be expected to move in opposite directions (e.g., stability as indicated by proportion of operations at greater than 50% power versus mean time to failure of specific equipment). An understanding of these relationships is crucial to a more accurate measurement of safety.

Concurrent with the focus on specific indicators of safety, we are also concerned with its conceptual definition. Work in this area includes attempts to conceptually link safety to similar notions (e.g., quality). More applied issues of safety are also being considered (e.g., is safety only the absence of accidents or should attempts to prevent accidents also be considered?).

In sum, the rich, though primarily conceptual, literature linking management and organizational design with organizational effectiveness is being systematically explored. We have just started, and recognize that the creation of reliable indicators of safety and reasonable verified hypotheses linking specific management/organizational characteristics to safety will be a difficult task.

REFERENCES

Chakoff, H. E. et al. Licensee Performance Evaluation, NUREG/CR-0110. Teknekron, Inc. October 1978.

Conver, S. Individual Site Ratings from the I&E Employee Survey on Evaluation of Licensees. April 1978 (unpublished).

Howard, E. M., Conver, S. T., Easterling, R. G. & Schwink, W. A Statistical Evaluation of the Nuclear Safety-Related Performance of NRC Operating Reactor Licensees During 1976. August 1977 (unpublished).

Nadel, M. V. (with others). Analysis of Processes Used in Evaluating Utility Management and Organization for an NRC Operating License (BHARC-400/82/018). Seattle, WA: Battelle Human Affairs Research Centers, 1982.

Osborn, R. N., Hunt, J. G., & Jauch, L. R. Organization Theory: An Integrated Approach. New York: Wiley, 1980.

Osborn, R. N., Sommers, P. E. & Nadel, M. V. An Analysis of Existing Management and Organization Guidelines for Nuclear Operations (BHARC-400/82/015). Seattle, WA: Battelle Human Affairs Research Centers, 1982.

U.S. Nuclear Regulatory Commission. NRC Licensee Assessments, NUREG-0834. August 1981.U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. Guidelines for Utility Management Structure and Technical Resources. (NUREG-0731). Washington, D.C., September 1980.

Widrig, R. D., Fabian, J., Wood, M. T., Shikiar, R., Goodman, P.S., & Osborn, R. N. "Development of Management Assessment Procedures for Licensing Reviews: Phase I Planning Report." Richland, WA: Battelle Pacific Northwest Laboratory, January 1981.

NRC Human Reliability Research Program at Sandia National Laboratories

R. R. Prairie and A. D. Swain

Overview

Beginning with the application of Sandia's human reliability analysis (HRA) methodology to WASH-1400, the NRC has sponsored application and research studies for us to expand this methodology to probabilistic risk assessment (PRA) of nuclear power plants (NPPs). Our present research program is funded by the Office of Nuclear Regulatory Research and is managed by the Division of Facility Operations. This human reliability research program consists of the further development of our HRA methodology known as THERP (Technique for Human Error Rate Prediction), the derivation of estimates of human error probabilities (HEPs) and models of human performance which apply to NPP operations, the definition of methods for collecting data and information to support the HRAs which are required for PRAs of existing and future NPPs, and the collection and analysis of this data to compare with the estimated HEPs and models we have developed.

In addition to this research program, the Office of Nuclear Regulatory Research has given us the opportunity to teach courses to NRC personnel on our HRA techniques, models, and data, to include our technology as part of the National Reliability Evaluation Program (NREP), and to apply our methods and data to PRAs which are part of the Interim Reliability Evaluation Program (IREP). This work is managed by the Division of Risk Analysis.

Through the Division of Safety Technology, Office of Nuclear Reactor Regulation, we have evaluated the HRAs in the Zion and Indian Point PRAs. Since these HRAs made considerable use of our HRA methodology, estimated HEPs, and models, our evaluation has provided us with information to use in improving our human reliability research products for the Division of Facility Operations.

Finally, under sponsorship by the Division of Reactor Programs, Office of Inspection and Enforcement, we have managed contractor research to develop methods for evaluating the quality of written materials used in NPP operations.

This paper briefly describes the human reliability research program supported by the Division of Facility Operations and very briefly mentions our other work supported by the NRC. All of our work for NRC is related directly or indirectly to the safety assessment of light water reactors.

Research on HRA Methods, Models, and HEPs

The Handbook

In October 1980 NUREG/CR-1278 Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications (Ref. 1) was issued as a draft report for interim use and comment. Hundreds of comments have been received from the U.S. and abroad. We collated and classified these comments (ref. 2) which have been very helpful in preparing a revised version of the Handbook. Our own experience in using the Handbook for NRC-sponsored PRAs in the IREP plus our reviews of other PRAs which made use of the Handbook, also provided valuable inputs for the revised Handbook which is in the final stages of preparation (ref. 3). In addition, contractor support from Science Applications, Inc., and Technology for Energy Corporation was obtained to develop a systematic approach to the categorization of procedure-based HEPs related to NPP tasks (ref. 4).

The Workbook

As a companion document to the Handbook we prepared NUREG/CR-2254 A Procedure for Conducting a Human Reliability Analysis for Nuclear Power Plants which was published in December 1981 as a draft report for interim use and comment (ref. 5). We have now revised this document (ref. 6).

Whereas the Handbook is both a guide and textbook for HRA, the Workbook provides a step-by-step procedure to illustrate the use of the data, information, and models in the Handbook. The Workbook is oriented specifically toward PRA.

Handbook Exercises

Under contract to SNL, Human Performance Technologies, Inc., arranged for 29 experts in human factors and/or PRA to solve some practical problems using the Handbook as the primary source book. It was hoped that these experts might be able to provide some data or information on estimating HEP's to include in the Handbook. This did not happen; in general they noted the subjectivity of our estimated HEPs in the Handbook, but they had no substitutes to offer. However, the participants in this study provided many useful suggestions for the revision of the Handbook, especially Chapter 20 which summarizes the estimated HEPs and models. Reference 7 reports the results of the Handbook Exercises project.

Special Reports

Two reports were prepared for the 1981 IEEE Standards Workshop on Human Factors and Nuclear Safety (known as Myrtle Beach II). Reference 8 is a short article on human reliability and reference 9 is an article and discussion on the human performance data bank concept.

Two other reports were prepared for the 1981 joint meeting of the American Nuclear Society and European Nuclear Society on PRA. Reference 10 summarizes the general approach to HRA described in NUREG/CR-2254, and reference 11 describes an approach to the incorporation of HRA into a PRA. A later report prepared for the 1982 Workshop on Low-Probability/High-Consequence Risk Analysis presents a case study of an HRA of NPP operations (ref. 12).

Methods to Collect Data to Support HRA

Three approaches are being investigated to provide the means of collecting data to support HRAs in PRA studies as well as to provide information on human performance of use to design trade-off studies in NPPs. These approaches involve the development of a program plan for a human performance data bank, the use of training simulators to provide HEPs and response times for control room tasks, and the formulation of procedures for using expert judgment to derive estimates of HEPs and response times.

Human Performance Data Bank

In late FY80 the NRC provided SNL with sufficient funding for a 2-year project to develop a program plan for a human performance data bank oriented towards the collection of data in NPP operations that could be useful in PRAs, and to try out the plan to see if such a data bank is feasible and practical.

General Physics Corporation was selected as the SNL contractor to develop the program plan, as the first part of the project. To date two major reports have been prepared and are in press (refs. 13 and 14).

The next part of the project is to carry out a test of the data bank concept. It is anticipated that this work will get underway during the first quarter of FY83.

Simulator Program for HRA

Because of the subjectivity of much of the estimated HEPs in NUREG/CR-1278 the NRC decided to see if training simulators could be used to provide realistic estimates of HEPs for certain control room tasks in NPPs. A two-year contract was let by SNL to General Physics Corporation to determine if there are HEP data available from simulators in other fields (e.g., commercial aircraft simulators), if simulator studies could indeed yield data useful for HRA, and, if so, to generate and carry out a research program in conjunction with ongoing operator training programs in NPP simulators to obtain indices of HEPs.

An unpublished survey of available data from simulators was completed, yielding negative results.

Data from earlier simulator studies conducted by the Oak Ridge National Laboratory were reexamined, and it was ascertained that a Performance Measurement System (PMS) developed by GPC and in use at the TVA simulator facility was capable of yielding worthwhile HEP data, if the data were gathered under suitable controls. Therefore, a study plan was developed for the purpose of gathering HEP data under well-controlled conditions. The study is currently under way, with data being gathered at two simulators.

Use of Expert Judgment

Because of the shortage of actuarial data on HEPs for HRA, persons performing PRAs must often resort to the use of expert judgment. In some PRAs the methods used for expert judgment have not been documented; in others, the methods used do not conform to the best available psychological scaling techniques. It was decided, therefore, there was a need for devising methods that would make the best possible use of expert judgment.

Accordingly a contract was let to Decision Sciences Consortium to review the literature on psychological scaling and to develop a set of best methods for using expert judgment to derive estimates of human error probabilities. This work has been completed and is presented in references 15 and 16.

Special Projects

Human Reliability Support

For several years we have been providing support to NRC in the general area of human reliability. As part of this effort we report on human reliability information from sources outside the U.S. This past year, at no cost to NRC, we visited NPPs and simulators in France, Japan, Sweden, and Norway. Information obtained in such visits was used in revising NUREG/CR-1278.

Fuel Handling Task Analysis

Under contract to SNL, System Research Laboratories, Inc., analyzed the fuel handling tasks at the Morris, Illinois, Independent Spent Fuel Storage Installation (ISFSI). The purpose of the study was to provide a technical basis for initial and continuation training for operations technicians at ISFSIs. Reference 17 describes the results of a task analysis and recommdendations for the training and certification of these personnel at ISFSIs in general.

Support for Division of Risk Analysis

Course on Human Reliability Analysis

On March 1-5, 1982, we conducted a one-week course for NRC personnel in Bethesda entitled "Assessment of the Effects of Human Performance on Nuclear Power Plant Operations." The next course will be taught Feb. 14-18, 1983, with a second one planned for August or September 1983.

National Reliability Evaluation Program (NREP)

As part of the Technical Writing Group which prepared NUREG/CR-2300 - PRA Procedures Guide, we wrote Chapter 4 - Human Reliability Analysis (ref. 18).

Interim Reliability Evaluation Program (IREP)

We provided some support to the Brown's Ferry BWR and the Calvert Cliffs PWR PRAs. We performed the HRA portions of the Arkansas Nuclear One Unit No. 1 PRA (ref. 19). The HRA approach used in the latter study is detailed in NUREG/CR-2254.

Support for Division of Safety Technology

We evaluated the HRA portions of the Zion PWR PRA and the Indian Point PWR PRA. Our evaluations are documented in references 20 and 21.

Support for the Office of Inspection and Enforcement

For the Division of Reactor Programs, Office of I&E, we contracted with Human Performance Technologies, Inc., to carry out an evaluation of NUREG/CR-1369, Checklist for Evaluating Maintenance, Test, and Calibration Procedures Used in Nuclear Power Plants. This document was tried out by NRC inspectors in the NRC regions. The outcome is reference 22, which is Revision 1 of NUREG/CR-1369.

Concluding Comments

Our research and applied studies for NRC in FY82 in the general area of human reliability and human performance in NPP operations indicate that there are problems but that these problems are solvable. HRA is viable, and it provides a needed part of PRA. Without it, PRA would not be treating the single largest source of potential safety problems in NPPs--human errors.

References

1. Swain, A. D. and Guttmann, H. E., Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, Draft Report for Interim Use and Comment, NUREG/CR-1278, U.S. Nuclear Regulatory Commission, Wash., DC, October 1980.

2. Miller, D. P., Peer Review Comments on NUREG/CR-1278, SAND82-1196, Sandia National Laboratories, Albuquerque, NM, October 1982.

3. Swain, A. D. and Guttmann, H. E., Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

4. Fragola, J. R. and Dougherty, E. M., Human Error Data Classification System, NUREG/CR-2943, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

5. Bell, B. J. and Swain, A. D., A Procedure for Conducting a Human Reliability Analysis for Nuclear Power Plants, Draft Report for Interim Use and Comment, NUREG/CR-2254, U.S. Nuclear Regulatory Commission, Wash., DC, December 1981.

6. Bell, B. J. and Swain, A. D., A Procedure for Conducting a Human Reliability Analysis for Nuclear Power Plants, NUREG/CR-2254, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

7. Brune, R. L., Weinstein, M., and Fitzwater, M. E., Peer Review Study of the Draft Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, SAND82-7056, Sandia National Laboratories, Albuquerque, NM, Sept. 1982.

8. Swain, A. D., "The Human Reliability Concept," R. E. Hall, J. R. Fragola, and W. L. Luckas, Jr. (eds), Conference Record for 1981 IEEE Standards Workshop on Human Factors and Nuclear Safety, Aug. 30 - Sept. 4, 1981, Myrtle Beach, SC, Institute of Electrical and Electronics Engineers, Inc., New York, NY, 1982, pp. 29-31.

9. Miller, D. P., "Human Performance Data Bank," R. E. Hall, J. R. Fragola, and W. L. Luckas, Jr. (eds), Conference Record for 1981 IEEE Standards Workshop on Human Factors and Nuclear Safety, Aug. 30-Sept. 4, 1981, Myrtle Beach, SC, Institute of Electrical and Electronics Engineers, Inc., New York, NY, 1982, pp. 43-47.

10. Bell, B. J. and Swain, A. D., "Overview of a Procedure for Human Reliability Analysis," Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, NY, Sept. 20-24, 1981, American Nuclear Society, LaGrange Park, IL, 1981, III, 1527-1536.

11. Bell, B. J. and Carlson, D. D., "IREP Human Reliability Analysis," Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, NY, Sept. 20-24, 1981, American Nuclear Society, LaGrange Park, IL, 1981, II, 587-596.

12. Bell, B. J., "Human Reliability Analysis: A Case Study," Workshop on Low-Probability/High-Consequence Risk Analysis, June 15-17, 1982, Arlington, VA, New York: Plenum Press, in press.

13. Topmiller, D. A., Eckel, J. S., and Kozinsky, E. J., Human Reliability Data Bank for Nuclear Power Plant Operations, Volume 1: Review of Existing Human Reliability Data Banks, NUREG/CR-2744, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

14. Comer, M. K., Kozinsky, E. J., and Eckel, J. S., Human Reliability Data Bank for Nuclear Power Plant Operations, Volume 2: A Data Bank Concept and System Description, NUREG/CR-2744, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

15. Stillwell, W. G., Seaver, D. A., and Schwartz, J. P., Expert Estimation of Human Error Probabilities in Nuclear Power Plant Operations: A Review of Probability Assessment and Scaling, NUREG/CR-2255, U.S. Nuclear Regulatory Commission, Wash., DC, May 1982.

16. Seaver, D. A. and Stillwell, W. G., Procedures for Using Expert Judgment to Estimate Human Error Probabilities in Nuclear Power Plant Operations, NUREG/CR-2743, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

17. Hottman, S. B., Bateman, R. P., and Biers, D. W., Analysis of Training and Certification of Operations Technicians at Independent Spent Fuel Storage Installations, NUREG/CR-2712, U.S. Nuclear Regulatory Commission, Wash., DC, July 1982.

18. Chapter 4, "Human Reliability Analysis," (written by B. J. Bell and A. D. Swain), in Vol. 1 of NUREG/CR-2300, PRA Procedures Guide, U.S. Nuclear Regulatory Commission, Wash., DC, in press.

19. Kolb, G. J. (Principle Investigator), Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit 1 Nuclear Power Plant, Vols. 1 and 2, NUREG/CR-2787, U.S. Nuclear Regulatory Commission, Wash., DC, June 1982.

20. Cramond, W. R., Review and Evaluation of the Zion Probabilistic Safety Study, Letter Report, Sandia National Laboratories, Albuquerque, NM, March 5, 1982.

21. Kolb, G. H., Review and Evaluation of the Indian Point Probabilistic Safety Study, Letter Report, Sandia National Laboratories, Albuquerque, NM, in press.

22. Brune, R. L. and Weinstein, M., Checklist for Evaluating Maintenance, Test, and Calibration Procedures Used in Nuclear Power Plants, Revision 1, NUREG/CR-1369, U.S. Nuclear Regulatory Commission, Wash., DC, Sept. 1982.

A Past, Present and Future Look at Emergency Preparedness

Michael T. Jamgochian
Human Factors Branch
Division of Facility Operations
Office of Nuclear Regulatory Research
Nuclear Regulatory Commission

In June 1979, the Nuclear Regulatory Commission began a formal reconsideration of the role of emergency planning in ensuring the continued protection of the public health and safety in areas around nuclear power facilities. The Commission began this reconsideration in recognition of the need for more effective emergency planning and in response to the TMI accident and to reports issued by responsible offices of government and the NRC's Congressional oversight committees.

On December 19, 1979, the Nuclear Regulatory Commission published in the Federal Register proposed amendments to 10 CFR Part 50 and Appendix E to Part 50 of its regulations.

The regulation contains the following three major changes from past practices:

1. In order to continue operations or to receive an operating license an applicant/licensee will be required to submit its emergency plans, as well as State and local governmental emergency response plans, to NRC. The NRC will then make a finding as to whether the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

2. Emergency planning considerations must be extended to "Emergency Planning Zones," and

3. Detailed emergency planning implementing procedures of both licensees and applicants for operating licenses must be submitted to NRC for review.

   In addition, the Commission revised 10 CFR Part 50, Appendix E, "Emergency Plans for Production and Utilization Facilities," in order to clarify, expand, and upgrade the Commission's emergency planning regulations. Section of Appendix E that were expanded include:

1. Specification of "Emergency Action Levels" (Sections IV.B and C)

2. Dissemination to the public of basic emergency planning information (Section IV.D)

3. Provisions for the State and local governmental authorities to have a capability for rapid notification of the public during a serious reactor emergency, with a design objective of completing the initial notification within 15 minutes after notification by the licensee (Section IV.D)

4. A licensee onsite technical support center and a licensee near site emergency operations facility (Section IV.E)

5. Provisions for redundant communications systems (Section IV.E)

6. Requirement for specialized training (Section IV.F)

7. Provisions for up-to-date plan maintenance (Section IV.G) Applicants for a construction permit would also be required to submit more information as required in the new Section II of Appendix E.

Since then we have continued to clarify and update these regulations based on public input and the staff's experience with using the emergency preparedness regulations. A few examples of these modifications are:

1. Extending the implementation dates in the installation of the prompt public notification systems.

2. Extending the implementation date for complying with the regulations for research reactions.

3. The staff presentation to the Commission to relax the frequency of full scale emergency preparedness exercises.

4. Clarifying the applicable emergency preparedness requirements for low power operation and

5. Refining the list of events that are immediately reportable to the NRC.

Future projects that I'm anticipating include the following

1. Continue to update the basic emergency preparedness regulations.

2. Codify relaxing the frequency of exercices .

3. Development and codification of emergency preparedness regulations for fuel cycle and material licensees.

4. Research relating to the interfaces (Federal, State and local governments and the licensees) that are necessary in dealing with an emergency.

5. Research dealing with the human factors aspects of initiating the prompt public notification systems, and

6. Research dealing with methods for better assessing the magnitude and course of an accident.

A PWR HYBRID COMPUTER MODEL FOR ASSESSING THE SAFETY IMPLICATIONS OF CONTROL SYSTEMS

O. L. Smith, N. E. Clapp, C. W. Craddick,

F. C. Difilippo, D. G. Morris, J. P. Renier, and C. W. Weber

The ORNL study of safety-related aspects of control systems consists of two interrelated tasks, (1) an augmented failure modes and effects analysis that will, in part, identify single and multiple component failures that may lead to significant plant upsets, and (2) a hybrid computer model that uses these failures as initial conditions and traces the dynamic impact on the control system and remainder of the plant. The first of these tasks is described in a companion paper by R. S. Stone. The second is reported here.

The initial step in model development was to define a suitable interface between the FMEA and the computer simulation tasks. This involved identifying primary plant components that must be simulated in dynamic detail and secondary components that can be treated adequately by the FMEA alone. The list of components included in the model is given below.

As an example of interfacing, a variety of initiating disturbances in auxiliary systems such as demineralizers may result in feed-pump failure. Initiating events in the auxiliaries are traced in detail in the FMEA; the demineralizers are not modeled. The pumps are modeled; pump failure is an initial condition for dynamic simulation of this sequence and its consequences. The FMEA will in general explore broader spectra of initiating events that may dove-tail into a reduced number of computer runs.

A portion of the FMEA (undertaken at Sandia) includes power supply failures. Events of this type define initial conditions for transients that may feed back on the initiating causes, as for example in load shed and pickup during and following blackout. There may thus be an iterative relationship between the FMEA and the computer simulation.

MODELING APPROACH

Since the thrust of this program is to investigate control system behavior, the controls are modeled in detail to accurately reproduce characteristic response under normal and off-normal transients. The balance of the model, including neutronics, thermohydraulics and component submodels, is developed in sufficient detail to provide a suitable support for the control system.

The ORNL hybrid computer consists of two AD4 analog units with approximately 300 amplifiers and a DEC10 digital computer with one million bytes of fast memory. The control system is being patched on the analog units to exploit their interactive capabilities: operator actions can easily be simulated during computer runs and the consequences of acts of omission and commission studied. The balance of the model is assigned to the digital machine.

The overall approach uses existing advanced state-of-the-art procedures available in production codes or in the literature. Some 30 codes and methodologies were reviewed. The computer time required by large codes to do the extensive calculations needed, appeared to make their cost prohibitive. The fast memory requirements of these codes also exceed the capability of the available hybrid computer.

The all-digital codes do not appear to allow the fully interactive mode of operation needed in our work. At the expense of generality, attempts are being made to simplify and streamline programming in the hybrid model, tailor it to specific plants, and improve computational speed and maneuverability. Wherever practical, existing subroutines are incorporated with minor alterations. Where it is more cost effective, original coding is being done from published methodologies. The use of tried and tested techniques minimizes basic development and classic problems such as the numerical instabilities common to two-phase flow. The use of confirmed techniques also provides a leg up on verification.

The model will primarily address mild to moderate transients that can occur at least partially under action of the non-safety control system. Severe transients such as large break LOCAs have been studied extensively elsewhere. The tool will be used to screen a large number of cases involving potential system malfunctions. Whenever the transients exceed the model's limitations, further analysis will be made with broader spectrum codes such as RELAP and RETRAN.

OVERALL MODEL LAYOUT

The model includes all principal plant components between the heat source in the fuel pins and the ultimate heat sink. In the primary loop these are the core, control rods, coolant pumps, core flood tanks, pipes, high and low pressure injection pumps, borated water storage tank, residual heat removal heat exchanger, core letdown tank, steam generators, and pressurizer with safety valves, PORV, heaters and spray valves. At least two coolant loops are simulated to assess asymmetries.

Balance of plant components include the turbine-generator (sectioned into high and low pressure turbines), moisture separators, reheaters, condenser, upper surge tanks, condensate storage tanks, ultimate heat sink, high and low pressure heaters, flash tank, condensate-booster pumps, main feedwater pumps, emergency feedwater pumps (both motor and steam driven), safety valves, bypass valves, and governing valves.

The control system includes the digital and analog logic. In the case of ICS simulation, the separate emergency feedwater and flash tank level controls are added.

Modeling is initially focusing on a B&W design (Oconee Unit 1). To assure that this level of resolution is adequate for other plants that may be included later, schematics of a Combustion Engineering (Calvert Cliffs) and a Westinghouse (Turkey Point) design were also prepared. Components on the three schematics are generically similar, with differences primarily of arrangement. The control systems of these latter plants are expected to require appreciably less of the analog computer capacity.

STATE-OF-THE-ART SUBMODELS AND NUMERICAL PROCEDURES

Core Hydraulics

Requirements for the core hydraulics submodel include 1 and 2 phase flow with one-dimensional axial spatial resolution. For the large majority of calculations expected in this study, single-phase conditions will prevail. However, two-phase capability may be needed for upset conditions such as occurred at TMI. If warranted, two-dimensional spatial detail can be added; this is not presently planned.

Formalisms considered for this part of the model include RELAP4, RELAP5, TRAC and RETRAN. Possible hydraulic options were the drift flux model, the dynamic slip model of RETRAN, the nonequilibrium, unequal velocity model of RELAP5, and the homogeneous approximation of RELAP4 (included as an option in RELAP5 and RETRAN2). For the mild to moderate transients of this study, nonequilibrium conditions are expected to be significant only in the pressurizer. Further, slip is not expected to contribute significantly to control system evaluation in most cases. Therefore the homogeneous approximation is sufficient for the majority of calculations. Higher level models will be used when this approximation is unsuitable. For the modest number of cases requiring higher resolution it should be more cost effective to use the production codes directly.

Neutronics

The neutronics submodel includes the point kinetics approximation and the one-dimensional axial calculation of flux distribution for cases in which control rod movement, voiding, or other non-uniformities are important. Two neutron groups provide adequate representation of the reflector flux shape. Six delayed neutron groups are used.

Candidate Codes for this submodel include WIGL3, PDQ, and VENTURE. Since WIGL3 obtains an analytical solution of the 2-group equations without iteration, it was judged to have a running time advantage over more general codes that allow arbitrary energy groups and which require iteration. WIGL3 has built in options for control-rod motion and voiding. The point kinetics approximation is implemented by specifying a single axial node with reflected boundaries.

Steam Generator

This submodel treats 1 or 2 phase flow in both the primary and secondary loops. Nodalization is particularized to the plant studied, using a once-through arrangement for B&W plants and a two-pass design for other PWR's. The steam generator (and piping) formalism is the same as used elsewhere in the primary loop.

Balance of Plant

ORTURB, a production code for turbine-generator-condenser simulation, is the basis for balance of plant modeling. This code has had extensive application in studies of Ft. St. Vrain and other plants. The feedtrain simulation permits detailed modeling of steaming and condensation in heaters and uses the formalism developed by J. G. Delene which was extensively applied in the desalination program.

A number of modifications were made to ORTURB to accommodate specific needs of this program. Speed variation was added to properly characterize the interaction between the turbine-generator and the control system under varying load conditions.

Feedwater flow instabilities have occasionally been experienced in some B&W plants. The cause or causes have not been fully diagnosed but are thought to be associated with components in the feedtrain. Approximately one-third of the total flow passes through flash tanks that may contribute to the instability. In an attempt to allow sufficient detail in the model for this instability to manifest itself, the flash tanks have been added to ORTURB. Also added were moisture separators, reheaters, and hotwell, booster, and feedwater pumps.

Pumps

In modeling system pumps, it is necessary to decide whether coastdown is important. If it is, inertial effects must be included, and the pump characteristic curves are needed. The main feedwater pumps are in this category, as are the booster-hotwell pumps if coastdown at pressures below about 500 psi is to be considered. In addition to inertial effects, the main coolant pump simulation includes both 1 and 2 phase flow capability. The treatment of two-phase pump flow is the same as used in RETRAN and RELAP, with single-phase characteristics multiplied by a two-phase degradation factor. For normal pump operating conditions the curves provided by the manufacturer or utility are used. Under off-normal conditions such as two-phase flow, SEMISCALE data are used.

If coastdown is not important, inertial effects may be ignored and a simpler head versus flow curve used. In this category are the auxiliary feedwater, high and low pressure injection, and booster-hotwell pumps if not in the inertial category.

Two-phase flow

The treatment of two-phase flow is based on that of RETRAN. Five flow regimes--sub-cooled liquid, nucleate boiling, transition boiling, film boiling, and superheated vapor--may occur in either forced convection or natural circulation. There are thus ten heat transfer regimes with associated coefficients, plus one condensation mode. Computational switching among modes is based on flow rate or Reynolds number, quality, and wall versus fluid temperature. Correlations used to evaluate the transfer coefficients are taken from RETRAN and RELAP. Water properties such as density and viscosity are taken largely from TRAC.

Control System

The most important single part of the simulation is the plant control system. The B&W model 721 Integrated Control System is the first being patched on the analog computer. The ICS is conceptually partitioned into four main subsystems: unit load demand development, integrated master control, steam generator feedwater control, and reactor control. Each consists of an array of modules such as integrators, summers, and limiters. This level of resolution, approximately 120 modules, is used in the analog simulation. To gain further insight into basic ICS operation, we plan to run a simplified version with some modules and intercouplings omitted. This should aid our understanding of how the ICS functions under certain failure conditions.

MODEL VERIFICATION

Before beginning production runs with the model, a period of verification is planned. Ideally, testing will occur in three steps. First, model calculations will be compared with benchmark calculations from RELAP and RETRAN. These need not be based on experiment but may be any suitable examples that exercise the dynamics of each of the principle parts of the model. The second phase of testing will compare calculations with existing transient data from nuclear power plants, and possibly with LOFT and SEMISCALE. Although economic and other factors may preclude the third step, testing should include data taken specifically to fill gaps in existing information.

SUMMARY

The modeling task has as its central objective the development of a tool to assess the safety implications of malfunctions in the non-safety control systems of PWRs during mild to moderate transients, an area that has received much less attention than the behavior of safety systems during severe transients. In keeping with this objective, control systems receive the most elaboration in the hybrid model. There is less interest in flow dynamics, for example, and the remainder of the plant is described at a level of detail that permits accurate conclusions about control system response.

In order to minimize model development, heavy reliance is being placed on existing numerical procedures, either directly as subroutines or as original coding of published methodologies. The resulting model is a synthesis of parts, and like most codes makes no claim of total originality. It is not the intention of this program to reinvent the wheel, but rather to tailor others' wheels to present purposes.

Principal plant components are represented while these and finer details are treated in the FMEA. Output of the FMEA provides initial conditions for the model, which then tracks the development of the transient and assesses whether a design safety limit is ultimately violated.

The model is intended in part as a tool to screen a wide range of potentially significant failures. Its limitations are recognized. Where these are exceeded, transients will be submitted to the broader spectrum codes for further analysis.

An Augmented Failure Mode and Effects Analysis Method for the Assessment of the Safety Implications of Control Systems

by

R. S. Stone*

ABSTRACT

Control systems have in the past been considered noncontributors to safety concerns. This having been demonstrated to be not the case by a number of events in operating plants, a program has been mounted to assess the safety impact of single and double failures in reactor control systems, using an augmented failure mode and effects analysis. The methodology is explained, and sample results of a broad study presented.

*Oak Ridge National Laboratory

P. O. Box X

Oak Ridge, Tennessee 37830

Work supported by the U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under Interagency Agreement DOE 40-550-75 with the U. S. Department of Energy.

An Augmented Failure Mode and Effects Analysis Model for the Assessment of the Safety Implications of Control Systems

Introduction

In years past, conventional wisdom has assigned low importance to the Safety Implications of Control Systems. This disregard stems from two philosophies: first, if a control function is of primary safety significance it should be part of the safety system. In general, this mandate has been followed. Second, a carefully designed set of safety grade systems exists to assure emergency plant shutdown, heat removal, and isolation capabilities -- in principle countering any challenge posed by a failed element in a nonsafety control system.

Over the past few years a number of accidents and near accidents have been initiated or propagated by failures of nonsafety components, creating situations which the safety systems have been hard pressed to counter.

Three Mile Island started down the road to a destroyed core as a result of control system failures. This is not a perfect example because mitigating safety features were interfered with by operator action, but even these human errors were contributed to by the absence of needed operational information. In the current concern over the pressurized thermal shock issue, control system failures are prominently mentioned as potential causes of overcooling transients. Indeed, overcooling transients have occurred on several occasions, thereby demonstrating both the soundness of the reactor vessels affected and the potential vulnerability of less ductile structures.

Appropriate remedies have been taken or are being taken in the hazardous control failures which have come to light, but the thought persists that other control failures, as yet undiscovered, have the capability to create hazards outside of the envelope protected by the safety system. It is from this perception that our program had its beginning. Once control failures which challenge safety have been identified, there will be an obvious need to determine remedies and insure their application. Some of these will follow traditional approaches. Where a control action's nonfailure is shown to be essential to safety, the necessary actuator can be promoted to safety grade. An alternate approach is to permit the control failure but to upgrade the safety system, extending its envelope of protection to the new area which has become of concern. Under other circumstances, fixes which are less drastic than additions to the safety system will be sufficient.

The objective of our program is thus twofold: first, to find, on a plant-specific basis, control malfunctions with serious implications for safety; and second, to determine solutions appropriate to the severity and probability of the postulated failure.

Methodology

A Failure Mode and Effects Analysis (FMEA) is the standard method used for a systematic qualitative search for significant failures and their consequences. It has commonly been applied to elements of the reactor protection system, these being considered not only necessary but also sufficient for the security of the plant. It is this same formalism that we are extending to failures in control.

The first step in performing an FMEA is to define the system to be analyzed. In our case, this means an exhaustive list of every system in the nuclear plant under study. Until the systems under consideration are narrowed to the fine details of specific designs, all PWR's involve much the same functions. This has permitted us to create a so-called generic systems list. In this list the plant systems and components are grouped in seven major categories:

N. Nuclear Systems
S. Engineered Safety Systems
C. Containment Systems
E. Electrical Systems
P. Power Conversion Systems
W. Process Auxiliary Systems
X. Plant Auxiliary Systems

These categories can be broken down into generic lists of subcategories, as for instance, POWER CONVERSION SYSTEMS:

P01	Main Steam System
P02	Turbine-Generator System
P02.A	Electro-Hydraulic Control Subsystem
P02.B	Turbine Gland Seal Subsystem
P02.C	Turbine Lubrication Subsystem
P02.D	Stator (Hydrogen) Cooling Subsystem
P02.E	Hydrogen Seal Oil Subsystem
P03	Turbine Bypass System
P04	Condenser and Condensate System
P04.A	Condenser Evacuation System
P04.B	Condensate Cleanup/Polishing System
P04.C	Condensate Heater Drain Subsystem
P05	Feedwater System
P05.A	Feedwater Heater Drain Subsystem
P06	Circulating Water System
P07	Steam Generator Blowdown System
P08	Auxiliary Steam System

Each of these subcategories is further divisable into third order categories and components, but differences between plants become evident at that detailed a level and from there on one must proceed on a plant-specific basis. However, the operational characteristics of each of these subsystems are common to all plants and their functions should be individually identified on a generic basis. For example, P03 Turbine Bypass System

The Turbine Bypass System allows the NSSS to follow approximately a fifty (50) percent step load reduction to the turbine-generator without causing a reactor trip or lifting the main steam pressure relief valves. The Turbine Bypass System is a non-safety system.

The Turbine Bypass System consists of a pneumatically or electrically operated turbine bypass valve and controls, one or two isolation valves and controls, and associated piping for each main steam line. On the occurrence of a large reduction in electrical load the turbine bypass valves open, thereby relieving main steam directly to the condenser. Some designs use a group of small turbine bypass valves in parallel rather than a single large valve for each steam line. This helps prevent an uncontrollable cooldown if a valve sticks open. The turbine bypass valves are opened automatically by the turbine electro-hydraulic control (EHC) subsystem following a large load reduction. During a normal shutdown of the reactor the turbine bypass valves are opened manually to release steam generated by decay heat in the reactor. As cooldown continues the turbine bypass valves are throttled closed, eventually transferring the decay heat removed to the Residual Heat Removal/Low Pressure Safety Injection System.

The Turbine Bypass System interfaces with the following systems:

- Main Steam System
- Condenser
- EHC Subsystem
- Instrument Air System
- Plant AC Distribution System
  (for motor operated isolation valves)

Such descriptions and interface identifications are required for each system. On a plant-specific inquiry the listing continues to the finest system level.

All these lists and interfaces are crucial to the FMEA process. It is unlikely that a serious failure mode will be uncovered if the system affected is omitted during system definition.

Having defined the plant systems and described their operation, it is necessary to limit the cases examined to a finite set by identifying the failure categories which will be examined. This will in general eliminate broad classes of systems which are not involved in the failures of concern. Moreover, in the systems which are of concern, selection of specific failure classes will limit the scenarios which must be examined and hence remove from consideration large classes of failure modes. In limiting an otherwise infinite task this is another vital step.

At this point in our description of the FMEA process we have identified all of the systems in the plant, described their functions and interfaces, and selected the class of failures which will be addressed. This is the point at which the first judgmental decisions will be made. Those systems without input to the failure classes of interest are eliminated, using as basis the previously developed functional descriptions and interfaces.

The ultimate event which we seek to identify and prevent is a breach of containment with leakage of radioactive material beyond allowable limits. Failures which only lower the barriers against such breaches of containment are also, with lower priority, to be identified and prevented. (Failures which cause a safety limit to be exceeded or which disable a portion of the protection system are examples of "barrier lowering".)

- Each identified hazard will be examined to determine what system function failures might have contributed to it. The system function failures thus found will be examined:

  a. to determine the mechanism for failure. Single cause or common modalities for combinations will be sought.

  b. To quantify the consequences of failure.

  c. To bring out the effects on other systems to which the failed system interfaces.

- Combinations of system function failures will be examined when the procedure above indicates that multiple failures are required for the breach of protection under investigation; the list will be examined for common modalities and a reduced list produced.

- Those reduced lists which meet pre-stated requirements concerning the number and kinds of initiating events will then be evaluated for possible simulation, probability determination, and reporting.

The above sequence is the primary approach to the FMEA process, i.e., an orderly assessment of the failure consequences of each identified system in the plant. Where failure of a control element produces actual or potential compromises to safety, interfacing systems are examined for contributions to the failure or to the consequences. Where a double failure is required to produce the hazard in question, common sources for the two failures will be sought, particularly in control logic sequence.

Systems which have a capability to impact the chosen failure classes are systematically examined for failure modes and the resulting first order effects. By "first order" we refer to those consequences which can be determined by logical inspection. For example, on a systems basis we may postulate a malfunction in which a turbine bypass valve fails open, allowing the full steam flow to bypass the turbines. The effect on a first order basis will be loss of pressure in the steam generator with possible overcooling. There may be other effects related to the condenser. Quantitative outcomes describing specific pressures and temperatures in various parts of the system will not in general be available from this type of deductive analysis.

For quantitative results, particularly for scenarios in which the affected system feeds back altered input conditions to the initiating event, failure effects must be determined by computer analysis. In the preceding paper, O. L. Smith described the modeling program which will be used for this purpose, providing what we have referred to as an augmented failure mode and effects analysis. The conventional exhaustive examination of individual systems will be performed by project personnel to determine failure modes. In the majority of cases the effects corresponding to these modes will be obvious, either benign or clearly unacceptable. The remaining ambiguous cases will serve as inputs to the computer program. The plant conditions predicted by the dynamic analysis become input data for the "effects" category of the FMEA.

Probability Considerations

The classical FMEA does not concern itself with the likelihood of occurrence of the events whose effects it predicts. Such probabilities are examined by companion methods of quantitative analysis, the FMEA being a qualitative technique. If the examination is restricted to single failures in protection systems, probabilities may not even be addressed; the single failure criterion requires the protection system to function properly despite any single failure within itself, whatever the probability. Even in control systems, single failures whose occurrences are found to produce safety crises may well be considered to command the same need for resolution, regardless of probability, as do single failures in the protection system itself.

In the case of multiple failures there is no such mandate in the General Design Criteria. Concern for multiple failures has generally been confined to common cause or cascade failures, considered to be special cases of single failures. The anticipated-transients-without-scram (ATWS) issue, which represents a double failure, has aroused concern because of the high probability of one of the two failures, experienced in most plants many times per year. Since, in the augmented FMEA, we intend to address multiple failures, the question of probability assumes an importance not present in most failure modes investigations.

The motive for including multiple failures in our analysis is a determination not to neglect safety hazards which may have until now successfully hidden behind the single failure criterion. There have been reactor incidents, some of them serious, involving multiple failures. We intend to search for undiscovered possibilities in this class.

As the number of multiple failures increases, the probability of simultaneous occurrences diminishes and the unwieldiness of the task soars. For this reason we shall limit our consideration to no more than two simultaneous failures. Conditions perceived to be hazardous--such as overcooling the reactor vessel or overheating the core--will be postulated. Single or double failures with the capability of causing the undesired result are then investigated. Where a double failure does in fact lead to danger, further procedures must be undertaken. The first approach will be to examine interfaces between the two failures in search of common elements. Where common cause or cascade failures can be demonstrated, the double failure is reduced to one. In these cases, probability need not be an issue.

Where such a reduction cannot be made and two independent failures remain a requirement, some measure of joint probability must be assessed. To maintain credibility, one or both failures should be of frequent occurrence, as in the ATWS case, so that the joint failure of both is of the same order of likelihood as the failure of a single channel of safety grade instrumentation. This project is not constituted as a probability investigation, but in multiple failures that aspect must be addressed. It is not anticipated that multiple failures of safety concern will be present in large numbers. Since this task is being pursued on a plant-specific basis, the best source for failure records of the control elements in question are plant repair records. Where these are not available, LER's and other failure records can be used to estimate the required failure rates on a generic basis.

The Plant-Specific Approach

The early stages of an FMEA are in progress for the first nuclear plant to be examined. The plant selected is Duke Power's Oconee-1, chosen largely because of its ongoing involvement with a related investigation. This is a 2,568 MWth pressurized water reactor of Babcock and Wilcox design, put into operation in 1973.

For the initial investigation, failures capable of producing pressurized thermal shock (PTS) were the target events. The generic list of plant systems described previously was first examined to select those elements which are capable of creating overcooling and/or overpressure phenomena.

PTS transients are defined to comprise the simultaneous occurrence of high reactor coolant system (RCS) pressure and low vessel wall temperature in high irradiation sections of the reactor vessel (belt line region). Typically, RCS pressures greater than 2000 psi and RCS temperatures less than 300°F are thought to be of significant concern.

Overcooling capabilities were the first characteristics we looked for. Three heat transport mechanisms could be involved in reducing the temperature of the central vessel wall. These are External Heat Transport through the reactor vessel insulation, Heat Transport to Fluids in the Vessel Downcomer, and longitudinal Conduction to Low Temperature Metal Sections. In Fig. 1, the systems involved in these heat transport modes are shown in their functional relationships.

The first mechanism, External Heat Transport, appears to require simultaneous removal of vessel insulation and water flooding of the reactor cavity. Several potential sources of water exist in the containment. These include the containment spray system, plant water lines, and feedwater lines. The potential for rupturing one of these lines or inadvertently actuating the spray system and subsequently flooding the reactor cavity should be investigated to assess the feasibility and time scale of these postulated events.

The second mechanism, Heat Transport to Downcomer Fluids, has a number of potential initiators. The principal mode of cold water injection into the downcomer is from an external source, i.e., from the core flood (accumulation) tanks (CF) or from the borated (refueling) water storage tank (BWST).

Two credible events can be postulated which result in CF tank injection: the operator failing to isolate and depressurize the tank during a controlled RCS shutdown and an uncontrolled depressurization due to a transient. Although possible, deliberate repressurization of the CF tanks and opening the isolation valves during shutdown are not considered credible.

The effect of CF tank injection during a rapid depressurization of the RCS is expected to be more significant. Following medium to large steam line breaks (SLB) or loss of coolant accidents (LOCA), the RCS pressure will rapidly drop below 600 psi and high CF tank injection flow rates can be expected. The effect of this injection on the vessel wall temperature has not been completely analyzed. As above, the wall temperature decrease will depend on the CF and reactor inlet flow rates, the mixing rates of the two fluids, and heat conduction in the vessel wall.

Injection of coolant from the BWST can occur through the low pressure injection system (LPI) pumping into the CF/LPI nozzles or through the high pressure injection (HPI) system pumping into the four inlet pipes. The water temperature of the BWST is typically in the 40°F to 80°F range.

The LPI system is initiated by the engineered safety features actuation system (ESFAS) based on low RCS or high containment pressures. Due to the 200 psi shutoff head of the LPI pumps, however, the LPI system will be capable of injecting coolant into the downcomer only following significant RCS depressurization. [The LPI pumps are also used in the residual heat removal (RHR) system.] With the RCS fully depressurized, the LPI is capable of pumping approximately 6000 gpm through the two CF/LPI nozzles.

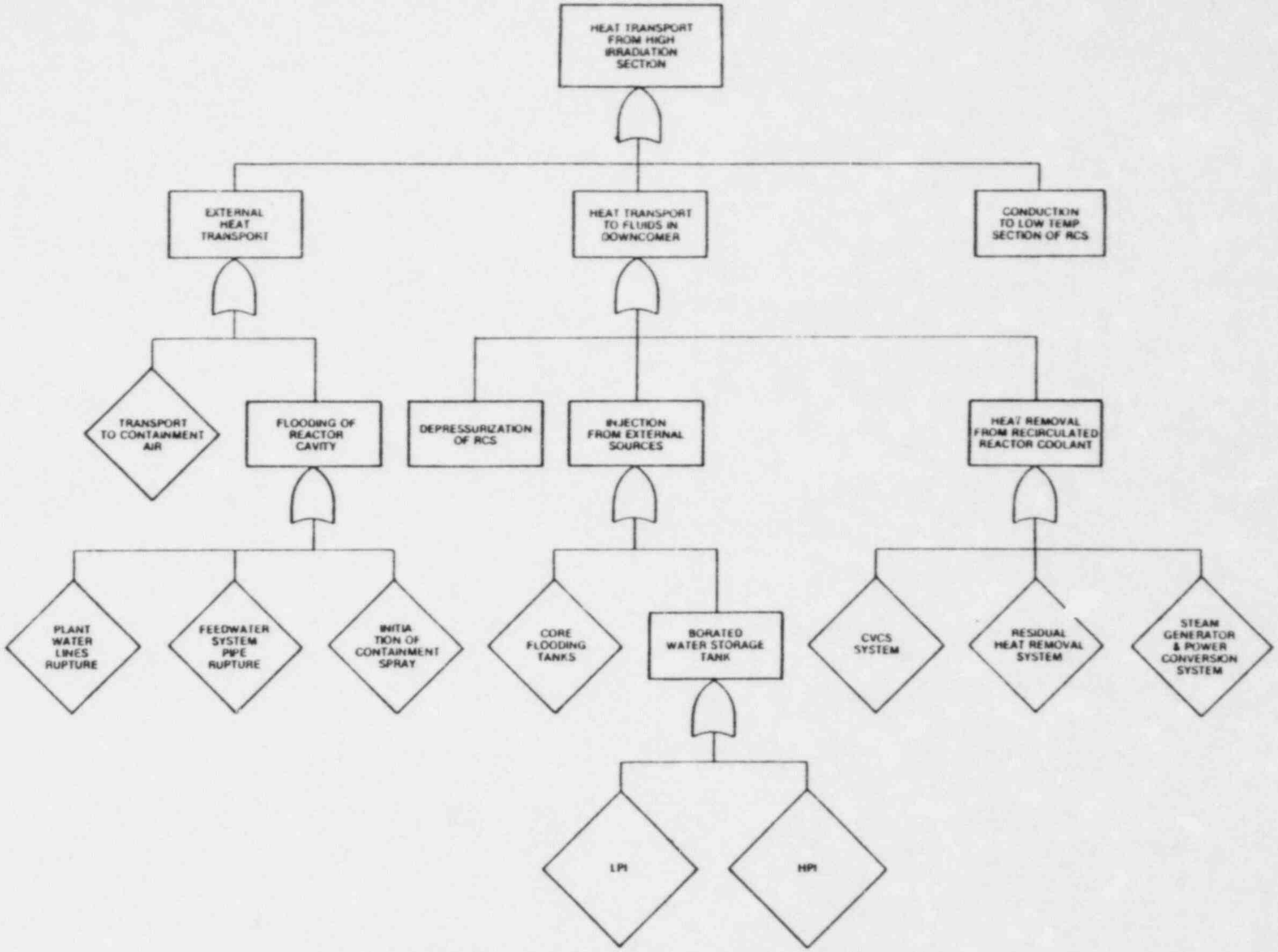


FIGURE 1. SYSTEMS INVOLVED IN HEAT TRANSPORT FROM HIGH IRRADIATION SECTION OF R. V.

The HPI system is actuated by the ESFAS on low RCS or high containment pressure following small to large SLB's or LOCA's or RCS overcooling. The three HPI pumps have a shutoff head exceeding 3000 psi. The injection flow rate will vary from less than 300 gpm per pump at operating RCS pressures to more than 500 gpm per pump at RCS pressures of 600 psi or less. The effect of HPI on vessel wall temperature will vary with HPI flow rate, the reactor coolant flow rate in the inlet lines, and the degree of mixing and heat conduction in the vessel wall.

The above approach is illustrative of the methods used to determine which plant systems can give rise to a specified hazard, and to determine the malfunctions of those systems which lead to the target result. As a result of these procedures, 19 generic systems were identified as having capability for overcooling involvement, with investigation of failures of these 19 systems the next objective.

Figure 2 displays the 19 critical systems, showing their interfaces with each other and with the reactor core and steam generator. The first step in the broad FMEA process is a study of systems level single failures, in which each of the 19 systems with potential PTS impact is failed in each of the modes in which it is capable of failure. Many of these failures are found to be benign, others are potential sources of overcooling, others require computer analysis to determine what the effects will be. Failures in one category, the ESFAS, are shown below to illustrate typical output from this process.

System: Engineered Safety Features Actuation System

Failure Mode	Effect and Remarks
Fail off	No effect in single failure mode.
Fail LPI system on	At high pressure, pumps deadhead into check valves. May damage pumps.
Fail HPI system on	Borated water injected, decreasing power. Coolant pressure increased until safety valves release. Causes reactor high pressure trip.
Fail Emergency Feedwater system on	May lead to overcooling and reactor trip.
Fail Containment Spray system on	Water in containment. May lead to overcooling. Can put water on outside of vessel.

Such single failures as appear to offer cause for concern are set aside for later study in the computational stage of the process.

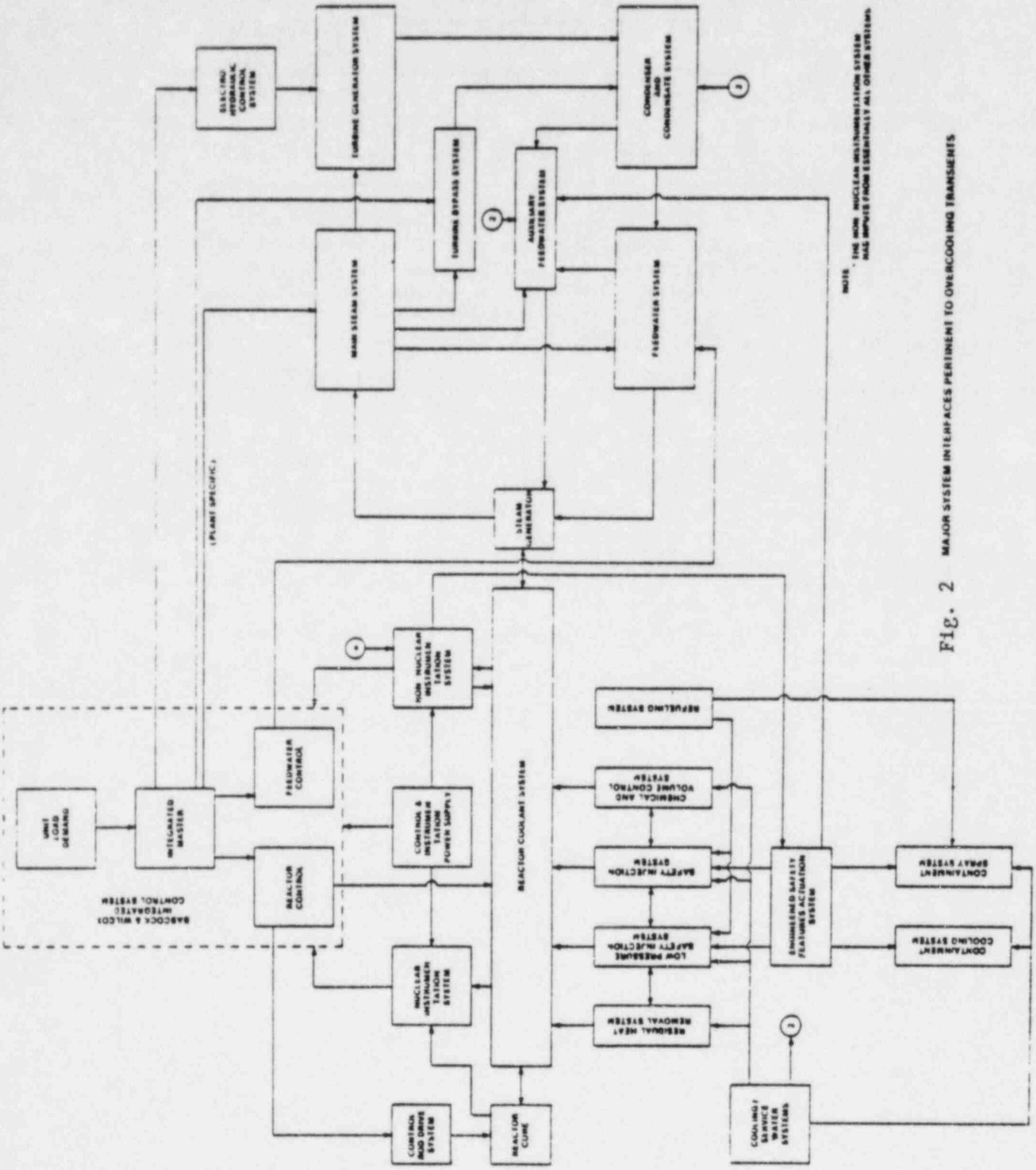


Fig. 2 MAJOR SYSTEM INTERFACES PERTINENT TO OVERCOOLING TRANSIENTS

The next phase of the broad FMEA augments the single failure investigation by allowing double failures, in this case combinations of the 19 critical systems. Those combined failures which produce effects essentially equal to the sum of their single failures (superposition) or are series failures where the last one dominates, are not presented. Only those failures which can produce significantly different states than the superposition of the single failures are considered. As in the case of single failures, a typical example of double failure is presented below.

Systems	Failure Modes	Joint Effects and Remarks
Condenser and Condensate System	Loss of flow	Partial loss of primary heat sink. Loss of primary coolant pump seal cooling. Operator will have to use emergency feedwater to establish natural circulation if not depressurize to use low pressure safety injection system. Possibility of core overheating and vessel overcooling. Requires simulation.
Plus		
High Pressure Safety Injection System	Flow fails low	

Again, combinations in this class which appear to offer interesting scenarios are set aside for later examination in depth. Examination of operator actions may be required, since the ultimate safety of a nuclear power station is in the hands of the operator. As has been seen, most transients result in reactor protection system actuation after which the operator must control the plant between core overheating and system overcooling. He has many paths of action and has been shown to take acceptable courses in the vast majority of cases. The cases where cascading events have occurred coupled with operator actions of commission or omission have been characterized by equipment in the wrong state (e.g., valves closed); equipment in "manual", preventing automatic action; failures which present incorrect information to the operator; and the operator misinterpreting information he is given. These are in addition to outright failures.

Portions of the process still to be accomplished are computer analysis of interesting cases and probability assessment where multiple failures are concerned. The overall objective is to identify previously unrecognized hazards, if each exist, and to suggest the most expeditious means for their resolution.

SAFETY IMPLICATIONS OF USING PROGRAMMABLE DIGITAL COMPUTERS IN NUCLEAR SAFETY AND CONTROL SYSTEMS[a]

D. M. Adams and R. R. Rohrdanz
EG&G Idaho, Inc.

This paper describes the activities being conducted at the Idaho National Engineering Laboratory associated with the use of stored-program computers for protection and control systems. This project has recently been initiated and a preliminary report will be available. The use of computers in plant control and protection (and more generally in systems important to safety) represents a major departure from the systems which have been used in the past. The design, development, and audit methods used for these systems are significantly different, thus requiring different skills and different perspectives.

Considerable concern has been expressed involving the use of computers at nuclear power stations, especially in 1E equipment. In fact, some computer manufacturing companies are discouraging their use. Others feel that although operators need better information displays, the use of computers constitutes a high "nifty factor." To remain competitive, vendors are turning to computer systems for use in new plant designs and for retrofitting existing plant designs.

The first concern is to determine if stored-program digital systems are needed. On the basis of the TMI-2 Lessons Learned Task Forces's recommendations to implement emergency response capabilities (Technical Support Center, Operational Support Center, and Emergency Operators Facility) and control room improvements (Safety Parameter Display Systems and Instruments for Accidents--Regulatory Guide 1.97), there is a need to

---

a. Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570.

process more information. Currently, control rooms have indicators on about 18 yards of control panels in three tiers, with very little integration. Operators are required to respond within a few minutes to extraneous power plant conditions. Using stored-program digital technology, the system designer can spend months designing the responses to these same extraneous power plant conditions.

These requirements indicate a need to process more information and, hopefully, enhance the operation of the plant. Computer technology offers a flexible, cost effective solution to these information and control problems. Fundamentally, computers can offer a number of advantages for control applications, including the capability to process and integrate large quantities of information, to perform calculations with fixed coefficients, and to perform extensive real-time diagnostics to ensure the integrity of the system.

Unfortunately, programmable digital devices also have some significant disadvantages. Probably the greatest disadvantage is that computers in a process environment are new to nuclear power stations and require that a new "bag of tricks" or new skills be used to implement designs. Problems with digital computers are often difficult to detect. The frequency response is dependent on the number of instructions processed. With larger systems, there are too many system states for conclusive testing, and residual errors are probable.

Given the apparent need for stored-program devices, their capabilities and their disadvantages, the concern of this task has been to identify the design issues that in some cases or under certain conditions constitute or precipitate a safety problem. These issues were identified using previous NRC reviews of digital systems, reviewing current standards and regulations, and through an extensive literature search. The design requirements edicted by law, standards, and regulations are, for the most part, fundamentally sound. Their implementation requires new skills, associations, and perhaps guidance in their design and evaluation. As with any design, the design of digital systems requires a structured design process (design method) that emphasizes the realization of functional requirements. Within that design method, provisions should be made for the "defense in depth" concept, the susceptibility and the reliability of digital systems.

The defense in depth principle is firmly established in the safety design of nuclear power plants as a means of protection for common-mode failures. Defense in depth consists of diversity, redundancy, and isolation (electrical and informational).

Programmable digital systems are susceptible to electromagnetic interference, configuration management problems, software practices, timing of functions (frequency problems), single failures that may bring the system down, and special power considerations.

The reliability of programmable digital systems can be enhanced through error detection/corrective action, quality requirements and equipment qualification, verification and system validation, system maintenance, system architecture, and planning for system obsolescence.

The evaluation of digital systems should (a) judge each of the design steps to ensure that the functional requirements are established and accomplished by the system, (b) establish adherence to the defense in depth concept, (c) determine the provisions made for the susceptibility of digital systems, and (d) establish provisions to verify the reliability of the system. The evaluation should be supported by testing and analysis for each functional requirement, each design issue, each design basis event, and their combination.

In looking at work done by others, the National Aeronautics and Space Administration and the Department of Defense have shown a heavy reliance on stored-program computers. Both organizations have funded extensive programs for the development of hardware, software, and standards. Modern process control systems allow tighter control processes and implementation of control strategies that could not have been accomplished just a few years ago. Most new high performance aircraft are "fly by wire systems" (no hydraulics). The Federal Aviation Administration has recently certified the airworthiness of the Boeing 757/767, the first commercial aircraft to be certified in about 10 years. The major concern during the certification process was the use of 180 microprocessors, some of which control critical aircraft functions. Boeing's decision to go ahead with digital rather than analog controllers was made primarily as a result of requirements for more built-in testing.

In general, the trend is that stored-program digital systems greatly improve the capabilities of systems, with about the same or better mean-time between failures.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, May 1980.

2. U.S. Nuclear Regulatory Commission, "Functional Criteria for Emergency Response Facilities," NUREG-0696, February 1981.

3. A. D. Friedman, Logical Design of Digital Systems, Comupter Science Press, Inc., 1975.

4. C. C. Fosts, Computer Architecture, Second Edition, Van Nostrand Reinhold Company, 1976.

5. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2," NUREG-0308, November 1977.

6. "Inside N.R.C.," June 14, 1982, page 9.

7. Mikros Systems Corp., Sales Information, Copyright 1981.

8. "The M68000 Family," Motorola Inc., Semiconductor Products, Inc., M68KFM-2.

9. S. Baker, "Long-Delayed High-End Micro P Family Finally Available from National," Electronic Engineering Times, May 24, 1982.

10. R. C. Camp, T. A. Smay, C. J. Triska, Microprocessor Systems Engineering, Matrix Publishing, Inc., Beaverton, Oregon, 1979.

11. A. V. Avizienis, "Arithmetic Error Codes: Cost and Effectiveness Studies for Application in Digital System Design," IEEE Transactions on Computers, Vol. C-20, No. 11, November 1971.

12. J. D. Ichbiah et al., "Reference Manual for the Ada Programming Language," Honeywell Systems and Research Center, Minneapolis, Minnesota, and Cii Honeywell Bull, Paris, France, July 1980.

13. G. K. Maki, "A Self Checking Microprocessor Design," Computer Sciences Press, Inc., 1978.

14. Combustion Engineering, Inc., "Topical Report, CENPD-172-A, Plant Protecion System With Integrated Automatic Test System," January 1978. (This contains the NRC Approval Letter and Topical Report Evaluation.)

15. U.S. Nuclear Regulatory Commission, "Advisory Committee on Reactor Safeguards," Transcript of Meeting, Friday, March 30, 1979 in Phoenix, Arizona.

16. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related To The Preliminary Design Of The Standard Reference System CESSAR System 80," NUREG 75/112, December 1975.

17. Westinghouse Electric Corp. "The RESAR-414 Integrated Protection System" with Amendments 1 through 19, October 8, 1976 to November 15, 1978.

18. IEEE Standard 603, "Standard Criteria for Safety Systems for Nuclear Power Generating Station," 1980.

19. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related To The Preliminary Design Of The Standard References System, RESEAR-414," NUREG-0491, November 1978.

20. U.S. Nuclear Regulatory Commission, "A Defense-In-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," NUREG-0493, March 1979.

21. "Boeing Refines 757/767 Digital Avionics," Aviation Week and Space Technology, April 19, 1982.

22. N. C. Thomas and E. F. Dowling, "Verification and Validation for Systems Important to Safety," Babcok & Wilcox, P.O. Box 239, Lynchburg, Virginia.

23. "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems," ANSI/IEEE Std. 352-1975.

24. "SDM/70, Systems Development Methodology," a proprietary product of Atlantic Software Inc., Philadelphia, PA.

25. "FAA, Industry Formalize Softwar. Design Process," Radio Technical Commission for Aeronautics, Special Committee 145, Document DO-. November 1981, as reported in Aviation Week and Space Technology, April 19, 1982.

26. "New Avionics Enhance USAF Inventory Reliability." Aviation Week and Space Technology, April 19, 1982.

27. "Definitions of Terms Used in IEEE Standards on Nuclear Power Generating Stations," IEEE Std 380-1975.

28. J. L. Norman Violette, "Electromagnetic Interference Control in Industrial Facilities," EMC Technology and Interference Control News, January 1982.

29. "Computer Surety--Computer System Inspection Guide," Lawrence Livermore National Laboratory, NUREG/CR-2288, UCID-18975, March 1982.

30. "Software for Computers in the Safety System of Nuclear Power Stations," Draft, International Electromechanical Commission, Technical Committee No. 45: Nuclear Instrumentation, Sub-Committee 45A: Reactor Instrumentation, October 1981.

31. W. Wesley Peterson, Error-Correcting Codes, MIT Press, 1961, Fourth Printing, 1968.

32. U.S. Nuclear Regulatory Commission, Memorandum from L. Beltracchi to R. M. Satterfield, "Audit of RESAR-414 IPS Development January 23-25, 1979," April 8, 1979.

33. F. Lynott, "Control System Dependability Considerations Encompass Availability, Integrity, Security," Control Engineering, January 1982.

34. A. V. Avizienis, "Fault Tolerant Systems," IEEE Transactions on Computers, Vol. C-25, No. 12, December 1976.

35. "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," IEEE Std P742/ANS 4.3.2/010-A, Draft, November 12, 1980.

36. J. A. Clapp, "Designing Software for Maintainability." Computer Design, September 1981.

37. J. A. McHaffey, "Reactor Safety System Design Using Hardened Computers," Georgia Institute of Technology, Engineering Experiment Station, prepared for EG&G Idaho, January 1981.

38. J. H. Wensley, "Fault-Tolerant Computers Ensure Reliable Industrial Controls," Electronic Design, June 25, 1981.

39. J. Hemenway, "Extended μC Capability via Multiprocessing," Engineering Design News, April 14, 1982.

Safety Implications of Using Programmable Digital Computers in Nuclear Safety and Control Systems

D.M. Adams

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

52 3439

Outline

Purpose →

Methods →

Design Issues →

Status

52 3438

Purpose

- Stored-program computers in protection and control systems
- Computers in systems "important to safety"

→ Methods

52 3437

Methods

- Need
- Capabilities
- Evaluation criteria
- Related work (NASA, DOD, FAA)

→ Design Issues

52 3436

Methods
(cont'd)

- Need
  - — Emergency response
  - — Control room improvements
    - - 18 yards of panels in three tiers not much integration
  - — New reactor designs

Design Issues

S2 3435

Methods
(cont'd)

- Capabilities
  - — Calculations made with fixed coefficients
  - — Errors detected and isolated
  - — Extensive operations (through-put)
  - — Cost

Design Issues

S2 3434

Methods
(cont'd)

- Capabilities (cont'd)
  - — Disadvantages
    - - New (new bag of tricks)
    - - Frequency response
    - - Conclusive testing is impossible (residual errors are probable)

Design Issues

S2 3433

Design Issues

- — Design steps
- — Defense in depth
- — Susceptibility
- — Reliability

Status

S2 3432

Design Issues
(cont'd)

- Evaluation criteria
  - Demonstration of adherence to design issues
  - Extensive testing

Status

52 3431

Design Issues
(cont'd)

- Related work (NASA, DOD, FAA)
  - Extensive funding
  - Improved systems
  - Comprehensive standards
  - Trends
    - Enhanced capability
    - Better mean time between failures

Status

52 3430

Status

- Preliminary report will be available
- Interim criteria Jan. 1983
- Manufacturers
  - 1E systems
  - Air traffic control

Questions

52 3429

Demonstration of a Noise Surveillance System at a PWR*

C. M. Smith
Instrumentation and Controls Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

INTRODUCTION

As a first step in demonstrating the practicality of performing continuous on-line surveillance of nuclear plants using noise related techniques, Oak Ridge National Laboratory is operating a computerized noise signal processing and data acquisition system at the Sequoyah Unit 1 Nuclear Plant, an 1148-MWe Westinghouse pressurized water reactor (PWR). The principal objective is to establish the long term signal characteristics of neutron and process signal sensors in order to evaluate the feasibility of detecting and diagnosing anomalous reactor conditions using these signals. The system is designed to screen the gathered data and identify for the noise analyst the data which differs statistically from norms which the system previously established.[1] Currently the system aids the noise analyst although it is intended to eventually aid the plant operator.

The system has access to 22 plant signals (seven ex-core neutron flux signals and several reactor coolant system flow, pressure temperature, liquid level, and vibrational signals). The system has monitored these signals during the first fuel cycle and the collected data base is being studied to determine the noise signal character over this fuel cycle.

MONITORING

Monitoring occurred primarily during 100% power operation. However, significant amounts of data were obtained at other than full power conditions. Also, magnetic tape recordings of 14 signals were made periodically throughout the fuel cycle to provide supplemental data to aid in interpretation of changes in the noise signatures detected by the surveillance system.

DATA ANALYSIS

The first fuel cycle has just terminated and the data base is still undergoing analysis. However, the noise signatures have already aided noise analysts at ORNL in understanding the changes in neutron noise caused by fuel element vibrations. For example, theoretical predictions of the increase in fuel element induced neutron noise as a function of

---

*Operated by Union Carbide Corporation under contract W-7405-eng-26 with the U.S. Department of Energy.

burnup and boron concentration reduction compare well with the changes in neutron noise recorded by the surveillance system over the fuel cycle.[2] Also, tape recorded data has been used to show that the relationship of ex-core neutron noise and core exit temperature noise shows potential as a means of monitoring flow in the core.[3]

SYSTEM PERFORMANCE

While monitoring Sequoyah's first fuel cycle, the system was evaluated to determine its performance in three areas: (1) how the system performed in tracking the different operating states of a commercial nuclear reactor; (2) how the system performed in adapting its screening to signals with different statistical characteristics; and (3) how the system performed in screening data to detect changed conditions.

Startup procedures, instrument calibration, and changes in power level are normal operating events in a commercial nuclear plant which must be handled by an automatic surveillance system. While this system is designed to adapt its monitoring to compensate for the occurrence of these events, this ability to adapt to unforeseen situations had to be evaluated during actual monitoring. Because the system makes initial assumptions about the statistical character of the signals which it monitors, it has procedures to modify the effects of these assumptions based on the data collected during monitoring. These modification procedures were evaluated for each reactor signal during the first fuel cycle. Also, the set of discriminants which detect spectral changes were evaluated to determine their sensitivity and usefulness.

CONCLUSION

The automated surveillance system has monitored the Sequoyah Nuclear Plant during its first fuel cycle. The system was able to acceptably adapt to different plant operating conditions. While evaluations are still ongoing, results indicate that the system was able to adapt to signals with different statistical character and that the discriminants are useful in detecting spectral changes. The system monitored long-term noise behavior, detected spectra that differ from what is considered normal, and provided concise storage of spectra together with the plant operating condition associated with the stored spectra.

REFERENCES

1. C. M. Smith, "A Description of the Hardware and Software of the PSDREC Continuous, On-Line Reactor Surveillance System," ORNL-TM-430, (To Be Published).

2. F. J. Sweeney and J. P. Renier, "Sensitivity of Ex-Core Neutron Noise to Vibration of PWR Fuel Assemblies," 7th International Conference on Structural Mechanics In Reactor Technology, Chicago, Ill, August 22-26, 1982.

3. F. J. Sweeney and B. R. Upadahyaya, "Measurement of Core Coolant Flow Velocities In PWR's Using Temperature - Neutron Noise Cross Correlation," American Nuclear Society Winter Meeting Proceeding, 1982.

PERFORMANCE OF AN AUTOMATED NOISE SURVEILLANCE SYSTEM AT SEQUOYAH-1

C. M. SMITH

SURVEILLANCE AND DIAGNOSTICS GROUP
INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY

PRESENTED AT

TENTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING
NATIONAL BUREAU OF STANDARDS
GAITHERSBURG, MARYLAND

OCTOBER 13, 1982

OAK RIDGE NATIONAL LABORATORY IS DEMONSTRATING A CONTINUOUS ON-LINE SURVEILLANCE SYSTEM AT TVA'S SEQUOYAH NUCLEAR PLANT.

- NOISE ANALYSIS TECHNIQUES HAVE BEEN APPLIED TO DETECT AND DIAGNOSE ANOMALOUS CONDITIONS IN REACTORS.
- THE SYSTEM DEMONSTRATES THE ABILITY TO CONTINUOUSLY MONITOR A REACTOR USING NOISE ANALYSIS TECHNIQUES.
- THE DEMONSTRATION STUDIES THE BEHAVIOR OF NOISE SIGNATURES DURING A FUEL CYCLE AND ESTABLISHES A DATA BASE OF THESE SIGNATURES AT VARIOUS OPERATING CONDITIONS.
- THE RESULTS OF THE DEMONSTRATION ALLOW THE DEVELOPMENT OF LIMITED DIAGNOSTIC CAPABILITY.

THE SYSTEM HAD ACCESS TO 22 SIGNALS FOR MONITORING OVER THE TWO YEAR FIRST FUEL CYCLE (JULY 1980 - SEPTEMBER 1982).

- HARDWARE LIMITED DURING FIRST YEAR TO MONITORING ONLY FOUR SIGNALS
  - 3 LOWER POWER RANGE EX-CORE NEUTRON CHAMBERS
  - 1 REACTOR COOLANT SYSTEM PRESSURE
- HARDWARE EXPANDED DURING SECOND YEAR TO MONITOR SIXTEEN SIGNALS
  - 4 LOWER POWER RANGE EX-CORE NEUTRON CHAMBERS
  - 1 AVERAGE POWER RANGE EX-CORE NEUTRON CHAMBER
  - 1 REACTOR COOLANT SYSTEM PRESSURE
  - 1 REACTOR COOLANT SYSTEM FLOW
  - 1 PRESSURIZER LEVEL
  - 2 CORE-EXIT THERMOCOUPLES
  - 1 REACTOR COOLANT SYSTEM HOT LEG TEMEPRATURE
  - 1 REACTOR COOLANT SYSTEM COLD LEG TEMPERATURE
  - 1 STEAM GENERATOR STEAM FLOW
  - 1 STEAM GENERATOR FEEDWATER FLOW
  - 1 STEAM GENERATOR WATER LEVEL
  - 1 STEAM GENERATOR PRESSURE
- SOFTWARE MODIFIED TO INCLUDE FOUR ADDITIONAL SIGNALS FOR THE LAST FIVE MONTHS OF THE FUEL CYCLE
  - 1 AVERAGE POWER RANGE EX-CORE NEUTRON CHAMBER
  - 1 INTERMEDIATE RANGE EX-CORE NEUTRON CHAMBER
  - 1 CORE EXIT THERMOCOUPLE
  - 1 REACTOR HEAD ACCELEROMETER

DURING THE FIRST FUEL CYCLE, THE PERFORMANCE OF THE SYSTEM WAS EVALUATED.

- HOW THE SYSTEM PERFORMED IN TRACKING THE VARIOUS OPERATING STATES OF A COMMERCIAL NUCLEAR REACTOR
- HOW THE SYSTEM PERFORMED IN ADAPTING ITS SCREENING TO SIGNALS WITH DIFFERENT STATISTICAL CHARACTERISTICS
- HOW THE SYSTEM DISCRIMINANTS PERFORMED IN SCREENING THE DATA TO DETECT CHANGED CONDITIONS
- HOW THE SYSTEM PERFORMED IN RECORDING THE RELATIVE CHANGES IN SPECTRA

THE SURVEILLANCE SYSTEM MUST BE ABLE TO MONITOR DURING NORMAL REACTOR OPERATIONAL CHANGES.

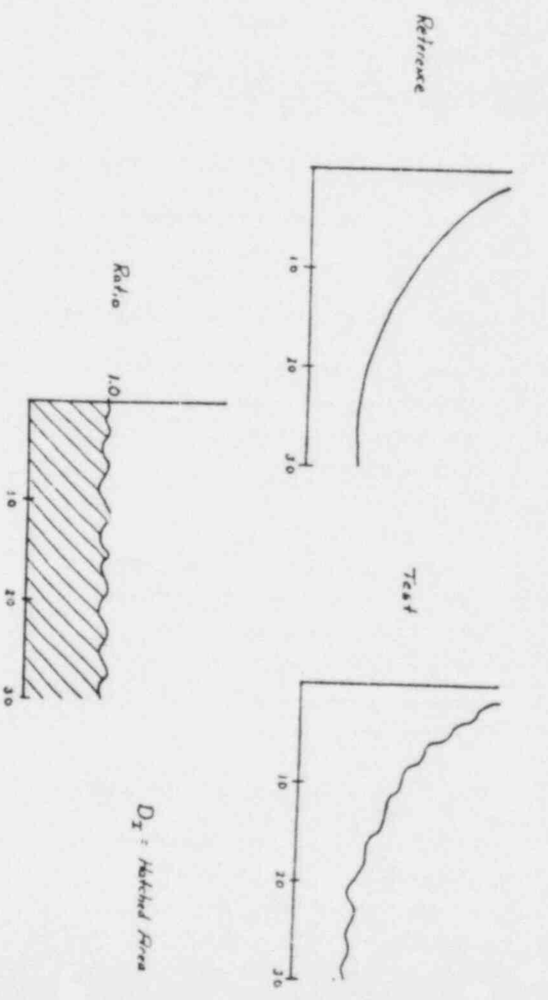
- STARTUP, SHUTDOWN, AND POWER LEVEL CHANGES REQUIRE SPECIAL HANDLING BY THE SYSTEM.
- AN OUT-OF-RANGE SIGNAL BECAUSE OF EITHER CALIBRATON OR A FAILED SENSOR REQUIRE DELETION OF THAT SIGNAL FROM ANALYSIS AND STEADY STATE DETERMINATION.
- LONG-TERM STORAGE OF SPECTRA MUST ACCOMMODATE AN EXPANDING AMOUNT OF DATA.

THE SURVEILLANCE SYSTEM MUST BE ABLE TO ADAPT TO SIGNALS WITH DIFFERENT STATISTICAL CHARACTERISTICS.

- INITIALLY ASSUME
  - (1) GAUSSIAN AMPLITUDE DISTRIBUTION
  - (2) INDIVIDUAL PSD ESTIMATES ARE INDEPENDENT
- CALCULATE LIMITING CRITERIA BASED ON ASSUMPTIONS
- MODIFY LIMITING CRITERIA BASED ON DATA COLLECTED DURING THE LEARNING PERIOD

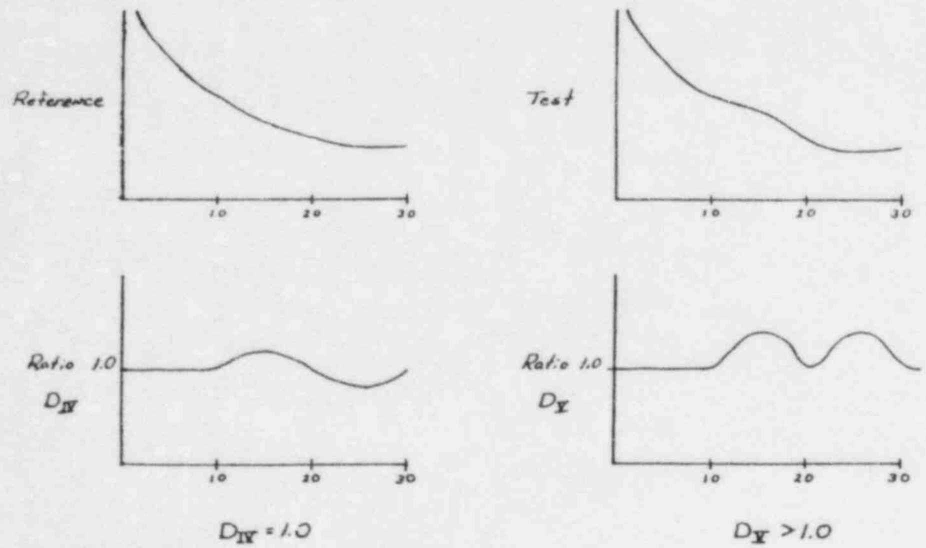
THE SYSTEM USES EIGHT DISCRIMINANTS TO DETECT A CHANGE IN THE SPECTRUM OF A SIGNAL.

- D_I INTEGRAL POWER RATIO
- D_{II} MINIMUM RATIO
- D_{III} MAXIMUM RATIO
- D_{IV} SUM OF LOG RATIO
- D_V SUM OF SQUARED LOG RATIO
- D_{VI} SIGN TEST
- D_{VII} NUMBER OF RUNS TEST
- D_{VIII} LONGEST RUN TEST



The Integral Power Ratio Is An Overall Measure Of The Difference Between The Test Spectrum And The Reference Spectrum

The Sum of Log Ratio And Sum of Squared Log Ratio Detect Spectral Changes Over A Relatively Narrow Frequency Region



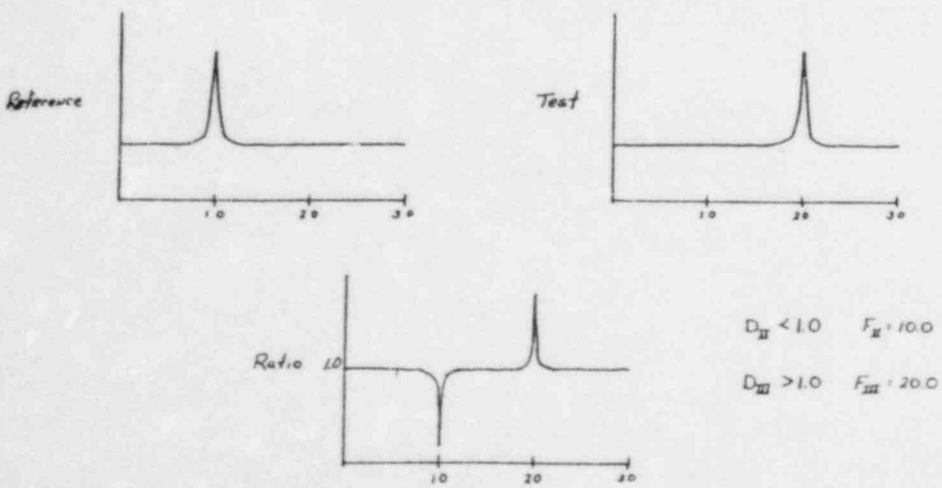
THE SURVEILLANCE SYSTEM MUST DETERMINE THE RELATIVE CHANGES IN SPECTRA.

- ABSOLUTE BASELINE DEFINES SPECTRA AT A GIVEN REACTOR STATE.
- LONG-TERM TREND MONITORS SLOWLY CHANGING SPECTRA CHARACTERISTICS AGAINST THE ABSOLUTE BASELINE.
- SHORT-TERM TREND MONITORS QUICKLY CHANGING SPECTRA CHARACTERISTICS AGAINST THE LONG-TERM TREND.

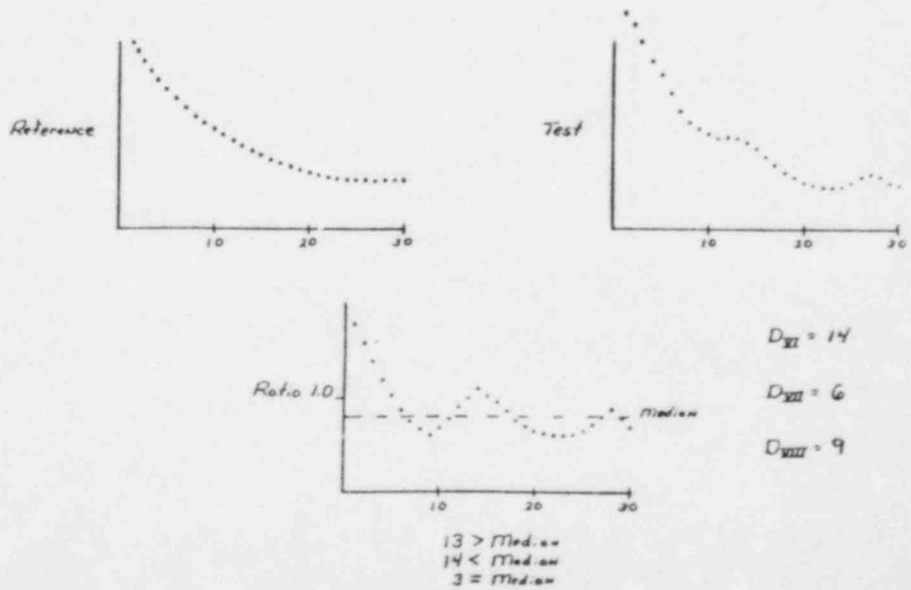
RELATIVE CHANGES IN SPECTRA ARE TRACKED BY:

- TEMPORARILY STORING SHORT-TERM TRENDS WHENEVER A CHANGE IS INDICATED.
- PERMANENTLY STORING BASELINES, ANOMALOUS LONG-TERM TRENDS, AND LONG-TERM TRENDS WHICH ARE DIFFERENT FROM THE BASELINE.

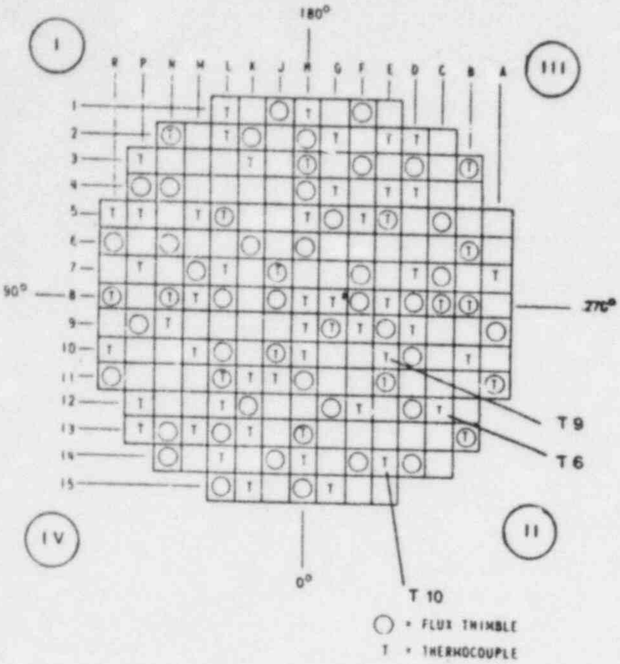
The Maximum And Minimum Ratios Detect The Appearance Or Disappearance Respectively Of A Discrete Spectral Peak In The Test Spectrum

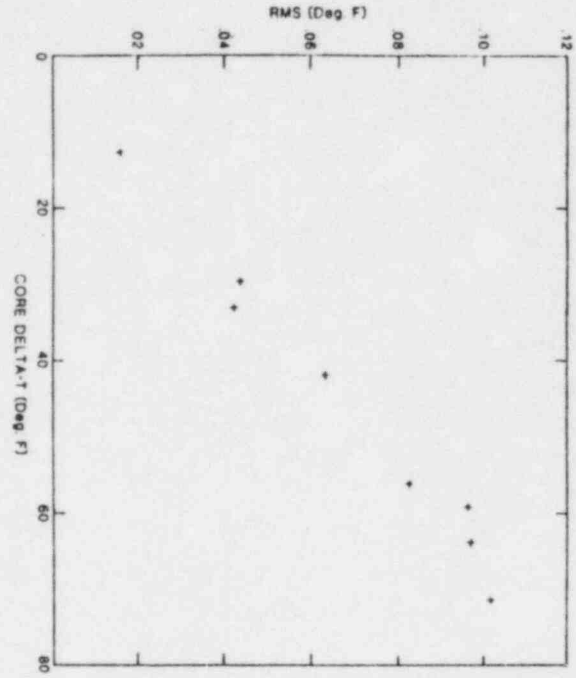


The Sign Test, Number of Runs Test, And Longest Run Test Detect Spectral Changes Over A Relatively Broad Frequency Region



[illegible] TEMPERATURE NOISE [illegible] AT SEQUOYAH-1 USING THE
AUTOMATED SYSTEM DURING A PLANT STARTUP.





THE [illegible] DATA SHOWED A LINEAR RELATIONSHIP BETWEEN THE RMS TEMPERATURE NOISE AND THE CORE ΔT.

IN CONCLUSION

- THE SYSTEM CONTINUOUSLY MONITORED THE SEQUOYAH NUCLEAR PLANT THROUGH VARIOUS NORMAL OPERATING STATES.
- THE SYSTEM ADAPTED ITS SCREENING TO SIGNALS WITH DIFFERENT STATISTICAL CHARACTERISTICS.
    - SIGNALS WITH STATISTICAL CHARACTER CLOSE TO THE ASSUMPTIONS REQUIRED A SMALL AMOUNT OF LEARNING.
    - SIGNALS WITH STATISTICAL CHARACTER SIGNIFICANTLY DIFFERENT FROM THE ASSUMPTIONS REQUIRED LONGER LEARNING.
- THE DISCRIMINANTS DETECTED CHANGES IN THE SPECTRA.
    - SENSITIVITY OF DISCRIMINANTS IS SUFFICIENT.
    - SUBSETS OF THE DISCRIMINANTS MAY BE USED WITH SPECIFIC SIGNALS.
- THE PERFORMANCE OF THE SYSTEM IN RECORDING RELATIVE CHANGES IN THE SPECTRA HAS NOT YET BEEN FULLY EVALUATED.

NUCLEAR POWER PLANT INSTRUMENTATION EVALUATION[a]

J. A. Rose
EG&G Idaho, Inc.

The Nuclear Power Plant Instrumentation Evaluation (NPPIE) Program is funded by the NRC office of Nuclear Regulatory Research with full concurrence of the Office of Nuclear Reactor Regulation. The purpose is to make a completely unbiased assessment of the ability of the U.S. nuclear industry to meet the intent of Regulatory Guide (RG) 1.97, Revision 2--"Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Condition During and Following an Accident." This, of course, cannot be done without some judgments on the Regulatory Guide itself.

This discussion gives the NPPIE Program objectives, a brief outline of the approach taken to meet those objectives, and some of the technical concerns formulted as a result of a preliminary assessment of the ways the industry intends to meet RG 1.97.

The first objective is to review and understand the current version of RG 1.97 from a technical standpoint. The second is to determine the technical problems facing the industry in the effort to meet the intent of the Guide. The final objective is to provide practical solutions to those problems.

The approach taken to achieve the stated objectives was to first, fully understand the intent of the Regulatory Guide from both a philosophical point of view and relative to the requirements for each measurement channel. The second approach was to compare the RG 1.97 requirements o

---

a. Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-ID01570.

the current industry practices in meeting the Guide and to the current state of the art for each of the required measurements. From this comparison and inputs from both the NRC and industry, the real technical problems will be determined. The deficient instrumentation systems will then be fully evaluated by detailed analysis, by test, or by both. When this is completed, the impact to the industry will be assessed and compared to the probable benefit to be derived. The final result will be recommendations of practical solutions to those problem areas.

A preliminary assessment of current industry practices has been made and compared to the intent of the Guide. From this, some 45 separate concerns have come to light. The five areas of primary concern are: reactor coolant level instrumentation, core exit thermocouples, containment area radiation monitors, halogen and particulate sampling, and coolant activity measurements.

It is recognized that there has been and continues to be a great amount of work done on reactor coolant level instrumentation. Our concern is that all the data being generated have not been fully evaluated for completeness, nor have they been compared, in total, to the requirements or intent of the Regulatory Guide. Further, the test data should be reviewed and the instrumentation studied for its ability to meet environmental qualification as well as to withstand, physically, the stresses probable during accident conditions.

The concern with core exit thermocouples (TCs) centers around two points. The first is the reduction in range, accuracy, and expected lifetime that can and does exist as a function of the materials and assembly procedures used. However, the largest factor effecting these things is the actual installation in the plant. We have seen installations that would certainly compromise thermocouple life and accuracy. The second concern is in qualifying these units to Category 1 requirements. The typical TC used has a 62-mil outside diameter. It is extremely doubtful this can be qualified to the temperatures required by RG 1.97. In addition, some work done at the Idaho National Engineering Laboratory relative to TC materials compatibility strongly indicates that these TCs can fail at accident temperatures in a way that can give potentially unsafe output indications.

The requirement for the high range containment area monitor is that it measure radiation levels to 10^7 R/h. The concerns here are the lack of available hardware, the inability to test the monitors on-site to these levels, and the potential inability of the system to withstand high radiation levels without either malfunctioning or providing grossly inaccurate readings.

Halogen and particulate sampling is a problem area because the research has not been conducted to determine if isokinetic sampling is really required. To design and install an accurate isokinetic sample system is very difficult, if not impossible, primarily because the flow regimes, particle compositions, and size distributions must be known for all conditions of interest. In addition, operating and maintaining a good system is very expensive. The need for such a system at any release point should be examined before the requirement is made, and certainly before it is met.

Primary coolant activity can be found by measuring either the radiation level or the radioactive isotopic concentration. The hardware to make these measurements is available, but its ability to be qualified to Category 1 is in question. With radiation level measurements, the critical parameter is detector location on the primary system. It must be positioned so that positive correlation between system output and fuel damage can be achieved. A definition of the real criteria to be used in locating the instrumentation is yet to be formulated.

The system design criteria for an isotopic concentration measuring system that will detect a breach in the fuel of some INEL test reactors have been developed by EG&G Idaho. An adaptation of these criteria for commercial reactor systems remains to be done. However, a correlation between on-line measurements of isotopic concentration and the severity of fuel damage has not been seriously attempted. If a need for this type of measurement is indicated, much more work is required.

Again, the overall goal of the Nuclear Power Plant Instrumentation Evaluation Program is to make an unbiased assessment of the ability of the U.S. nuclear power industry to meet the intent of Regulatory Guide 1.97, Revision 2. To do this, we must thoroughly understand both the intent and application of the RG 1.97 requirements and we must understand the real technical problems faced by the industry in meeting the Guide. Then, and only then, will we be able to provide practical solutions to those problems.

Nuclear Power Plant Instrumentation Evaluation

J.A. Rose

IDAHO NATIONAL ENGINEERING LABORATORY

EG&G Idaho, Inc.

Nuclear Power Plant Instrumentation Evaluation

- Objectives
- Approach
- Technical concerns
- Summary

52 3440

Objectives

- Review of RG 1.97
- Determine technical problems
- Provide solutions

52 3441

Approach

- Understand intent of RG 1.97
  - — Measurement philosophy
  - — Channel requirements
- Compare RG 1.97 requirements to:
  - — Current practices
  - — State-of-the-art

52 2808

Approach (cont'd)

- Determine real technical problems
- Evaluate deficient instrumentation
  - Analysis
  - Testing
- Assess impacts
- Recommend practical solutions

S2 2810

Technical Concerns

- Reactor coolant level
- Core exit temperature
- Containment area radiation
- Plant release sampling
- Primary coolant activity

S2 3442

Reactor Coolant Level

- Data evaluation
  - Completeness
  - Requirements
- Instrumentation review
  - Qualification
  - Robustness

S2 3443

Core Exit Temperature

- Reduced range, accuracy and life
  - Materials and assembly
  - Installation
- Qualification to Category 1
  - Physical size
  - Materials compatability

S2 3444

Containment Area Radiation (High Range)

- Testing to 10^7 R/hr
  - Detector damage
  - Inaccuracy
- Qualification to Category 1
  - System radiation effects

S2 3445

Primary Coolant Activity

- Radiation level
  - Measurement locations
    - Criteria
    - Installation
  - Qualification to Category 1
    - Methods
    - Procedures

S2 3446

Primary Coolant Activity (cont'd)

- Radioactivity concentration
  - Detection of breach criteria developed at INEL
  - Degree of fuel damage
    - Damage versus concentration
    - System design criteria
  - Qualification to Category 1

S2 3447

Plant Release Sampling (Halogens and Particulates)

- Isokinetic samples may be required
  - Large particles
  - Dose contribution
  - Accuracy requirements
- To design for accuracy - need:
  - Temperature and pressure
  - Flow regimes
  - Particle sizes and composition

S2 3450

Summary

- Program objectives
  - Assess ability of industry to meet intent of RG 1.97
  - Provide practical solutions
- Technical approach
  - Determine requirements
  - Determine industry methods
  - Evaluate performance

82 3448

Summary (cont'd)

- Technical concerns
  - Level measurements
  - Core thermocouples
  - Area monitors
  - Particulate sampling
  - Activity monitors

82 3449

A TORSIONAL ULTRASONIC TECHNIQUE for LWR LIQUID LEVEL MEASUREMENT

W. B. Dress

Union Carbide Corporation
Oak Ridge National Laboratory
P. O. Box X, Building 3500
Oak Ridge, Tennessee 37830

In the late 1970's, a technique for determining the mean density of a fluid surrounding a waveguide of non-circular cross section was developed using slow torsional ultrasonic pulses. With the waveguide only partially inserted in a liquid of known density, the length of the probe immersed in the liquid can be derived, and hence the level. (See, for example, references 1 through 3). The loss-of-cooling accident at TMI-2 in 1979 pointed out the need for a sensor to monitor continuously reactor vessel coolant level and density (i.e., void fractions). Devices utilizing torsional ultrasonic pulses can provide continuous readouts, in appropriate units, of level, density (or void fraction), and temperature along a chosen path in the reactor vessel, in a number of steps (limited to about 20 sections per probe in practice).

Additional advantages of the torsional-wave level probe include ruggedness and long life provided by the simplicity of the waveguide structure -- only a ribbon of stainless steel immersed in the reactor vessel. Thus the probe is expected to have a long life in the high neutron and gamma fluxes as well as being adaptable to current reactor designs.

Under normal reactor operation the torsional wave sensor will provide a display of level (if it exists, as in a BWR), as well as density, and temperature profiles. These indications can be correlated with measurements from other plant sensors providing confirmation of the indications of plant instrumentation. Under accident conditions, the probe would be expected to provide indications of the event to the point of destruction of the stainless steel itself. Thus, a continuous reading of temperature, level, and density would be available, providing a log of the event.

Since 1980, the Instrumentation and Controls Division of ORNL has been involved in the evaluation of sensors using torsional ultrasonic pulses for level and density measurements, and extensional ultrasonic pulses for temperature measurement and correction. Work at ORNL through 1981 demonstrated the feasibility of transmitting torsional pulses in a ribbon of stainless steel and successfully detected the level of a steam/water interface at 550F and 1550 psi. Subsequent work under the Advanced Two-Phase Instrumentation Program has been directed towards an operational prototype instrument that can be installed in functional power reactor.

To this end, we have pursued a number of sub-goals whose successful completion is necessary for a fully functional device. The first such step was to demonstrate the transmission of stress pulses over long distances such as would be required in a typical application. Figure 1 shows the attenuation of signal strength at the transducer coil as a function of distance. The loss was about 12% per meter; however signals of a few tens of millivolts are quite usable and a probe as much as 40 meters long will still have an adequate echo from its far end. Attenuation in a circular rod is somewhat less -- around 10%, so the portion of the waveguide leading to the active section of the probe should be circular. In a related experiment, the electrical signals were transmitted through a 400 foot length of coaxial cable with no significant attenuation.

Figure 2 shows a block diagram of the instrumentation used with the probe. Work has been concentrated primarily in transducer design -- including means for alternately providing torsional and extensional biasing for the two pulse modes; production of fast, high-energy pulses for strong, narrow signals; a signal-conditioning circuit to find the peak of the echo pulses; and a precise digital timer that is interfaced directly to a microcomputer. The use of a microcomputer has a number of advantages: (1) control of the measurement cycle allowing alternate torsional and extensional pulses with the appropriate magnetic biasing; (2) on-line calculation of the data using calibration tables and appropriate algorithms; and (3) graphic display of the level along with density and temperature profiles showing any sections where voids may be present. Figure 4 is an example of a computer-generated display showing level and density changes resulting from adding a brine solution to alcohol.

Previously, a severe restriction to the generation of torsional signals has been the loss of permanent magnetic bias in the magnetostrictive materials at elevated temperatures. The azimuthal magnetic field used to generate the torsional waves disappears completely from Remendur around 350 to 450°F. Earlier solutions included isolation or cooling of the transducer section. There are drawbacks, however, because of the loss of signal through any isolating walls or the additional complications due to the presence of cooling coils. Two solutions have been devised to overcome this problem. One method used by Arave in the in-vessel LOFT density probe [3], is to use "mode conversion." That is, mechanically convert extensional stress pulses to torsional pulses -- Photo 1 includes an example of such a probe. The second solution, which we employed, is to apply a torsional bias as mentioned above. The azimuthal magnetic field was established by providing an axial electric current in the magnetostrictive material. Examples of applying several values of such currents on the torsional amplitude are shown in Figure 3 for the range of 150 to 700°F.

The configuration of the apparatus for a level/density demonstration is shown in Photo 2. The probe and liquid are in the center of the photograph; the electronics above left, and the computer display to the left. In this demonstration, the computer generated a start pulse to initiate the measurement cycle. The start pulse turned on the torsional bias supply and gated the pulser on. Blanking was also provided by the computer so that echos not of interest were ignored. The correct echos, corresponding to the section of the probe under consideration, were then selected to start and stop a scaler which counted stable clock pulses at a 10 MHz rate. Upon the stop signal, the counts were then read by the computer and averaged in an appropriate fashion. Averaging over 100 cycles of about 10 ms each resulted in a time resolution of 10 ns which is adequate for a temperature resolution of 2°F and 0.1% density and level determination (relative to the length of the section under consideration).

Figure 4 shows the results of slowly adding a brine solution (density of about 1.5) to alcohol (density of about 0.8), and measuring the change in the resulting mixtures with a two-section probe. The keyed points on the plot show where: (1) all of the lower section of the probe was immersed in the alcohol; (2) a portion of the upper section was immersed; (3) initial introduction of brine by pouring it down a tube leading to the bottom of the vessel so that the less dense alcohol tended to "float" on top; (4) brine reached the upper section -- here the level information was no longer valid since it was based on an assumed density for alcohol, however, the sudden and sharp change in "level" indicated that the denser medium was forcing out the lighter alcohol; (5) alcohol completely overflowed (the level remained constant thereafter); (6) probe withdrawn exposing upper section; (7) upper section drained of brine; (8) upper section cleaned of salt residue.

In conclusion, we have successfully met several major milestones on the way to a functional untrasonic sensor for light water reactor instrumentation. In particular, we have overcome the problem of temperature on the torsional signals and shown that transmission of useful stress pulses over long distances is feasible. A simple demonstration of level and density measurements was presented graphically, showing the integrated instrument including: the level transducer with the torsional and extensional bias controlled by the microcomputer, high-speed electronics for precise time measurements, and on-line display of the resulting data shown graphically.

References

. Lynnworth, L. C., "Slow Torsional Wave Sensors", pp. 29-34, IEEE Ultrasonics Symposium Proceedings, Cat. No. 77CH1264-1SU (1977).

?. Lynnworth, L. C., "New Designs for Magnetostrictive Probes Using Extensional, Torsional and Flexural Waves", pp. 300-304, IEEE Ultrasonics Symposium Proceedings, Cat No. 78CH 1344-1SU (1978).

3. Arave, A. E., "Ultrasonic Densitometer Development", NRC Advanced Reactor Instrumentation Review Group Meeting, July 24-26, 1979 (unpublished).

4. Miller, G. N., Anderson, R. L., Rogers, S. C., et al., "High Temperature, High Pressure Water Level Sensor," Proc. 1980 Ultrasonics Symp., IEEE, New York, NY, pp. 877-881.

5. Rogers, S. C. and Miller, G. N., "Ultrasonic Level, Temperature and Density Sensor," IEEE Transactions on Nuclear Science, NU-29, 665, (1982).

6. Hardy, J. E., Miller, G. N., Rogers, S. C., and Zabriskie, W. L., "Advanced Two-Phase Flow Instrumentation, Quarterly Report," NUREG/CR-2752, Vol. 1, ORNL-TM-8365, 1982.

Pulse Attenuation in Flat Ribbon

Amplitude versus Distance

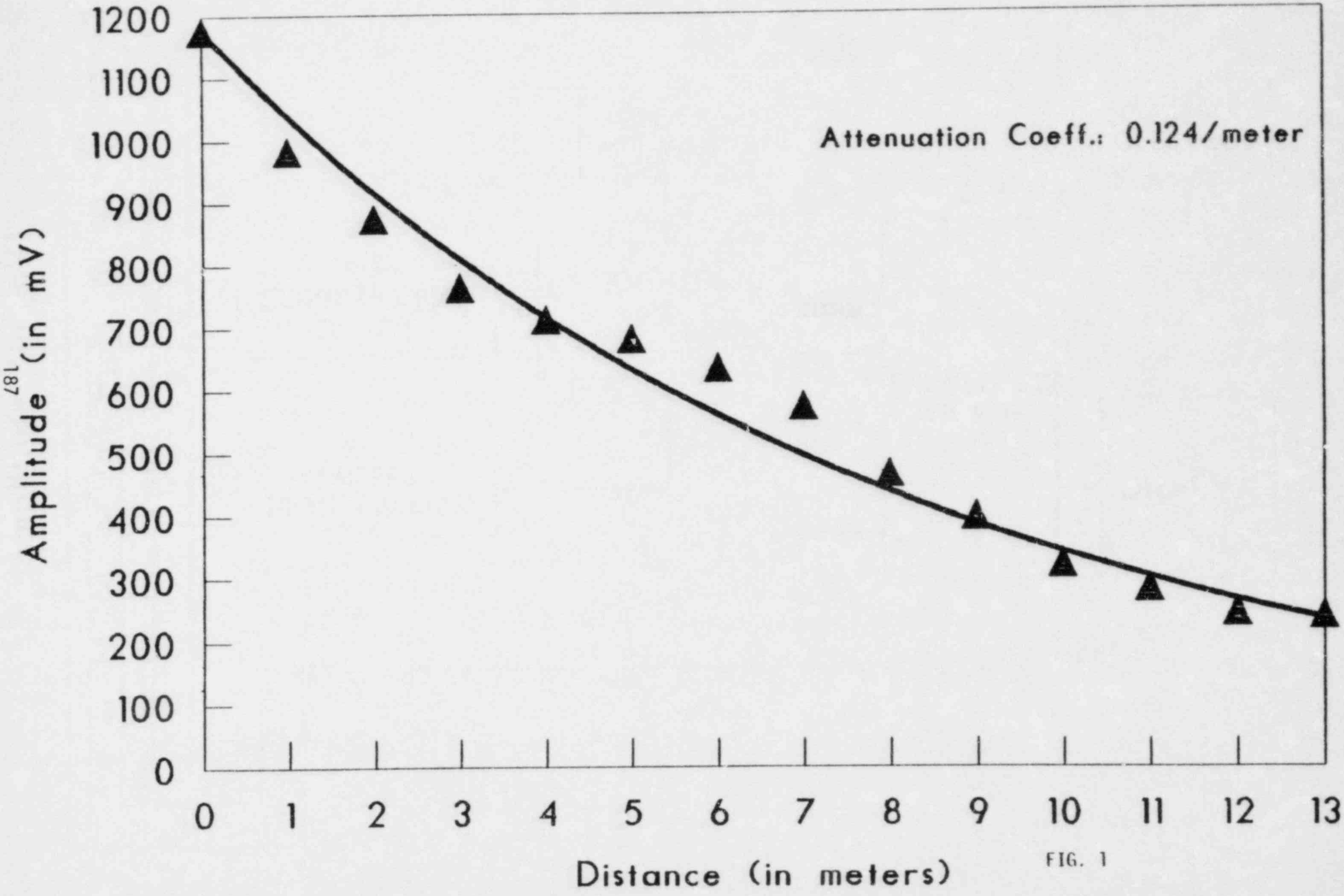


FIG. 1

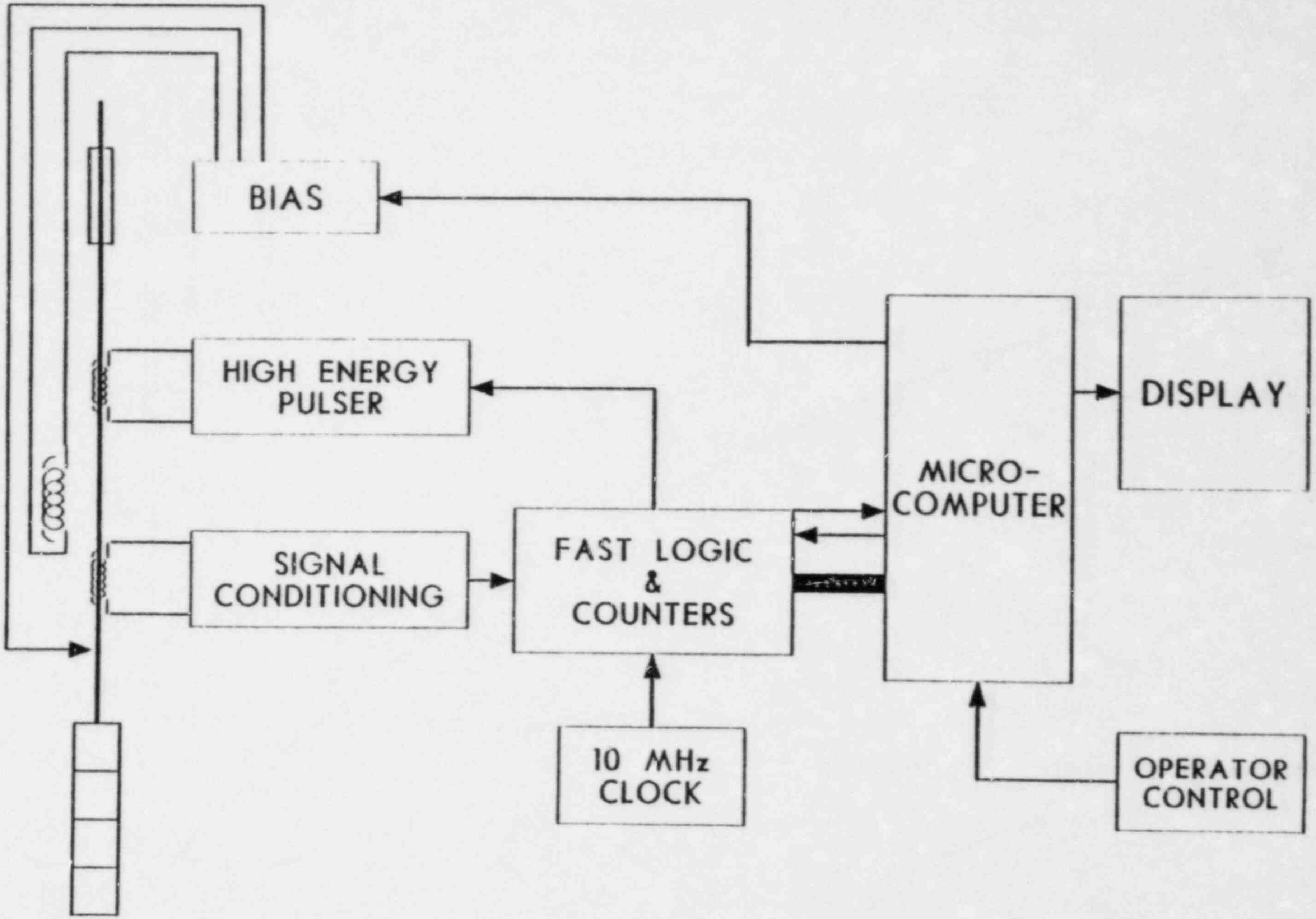

BIAS
HIGH ENERGY PULSER
SIGNAL CONDITIONING
FAST LOGIC & COUNTERS
10 MHz CLOCK
MICRO-COMPUTER
DISPLAY
OPERATOR CONTROL

FIG. 2

Temperature Effects on Torsional Signal Amplitude with Bias Currents

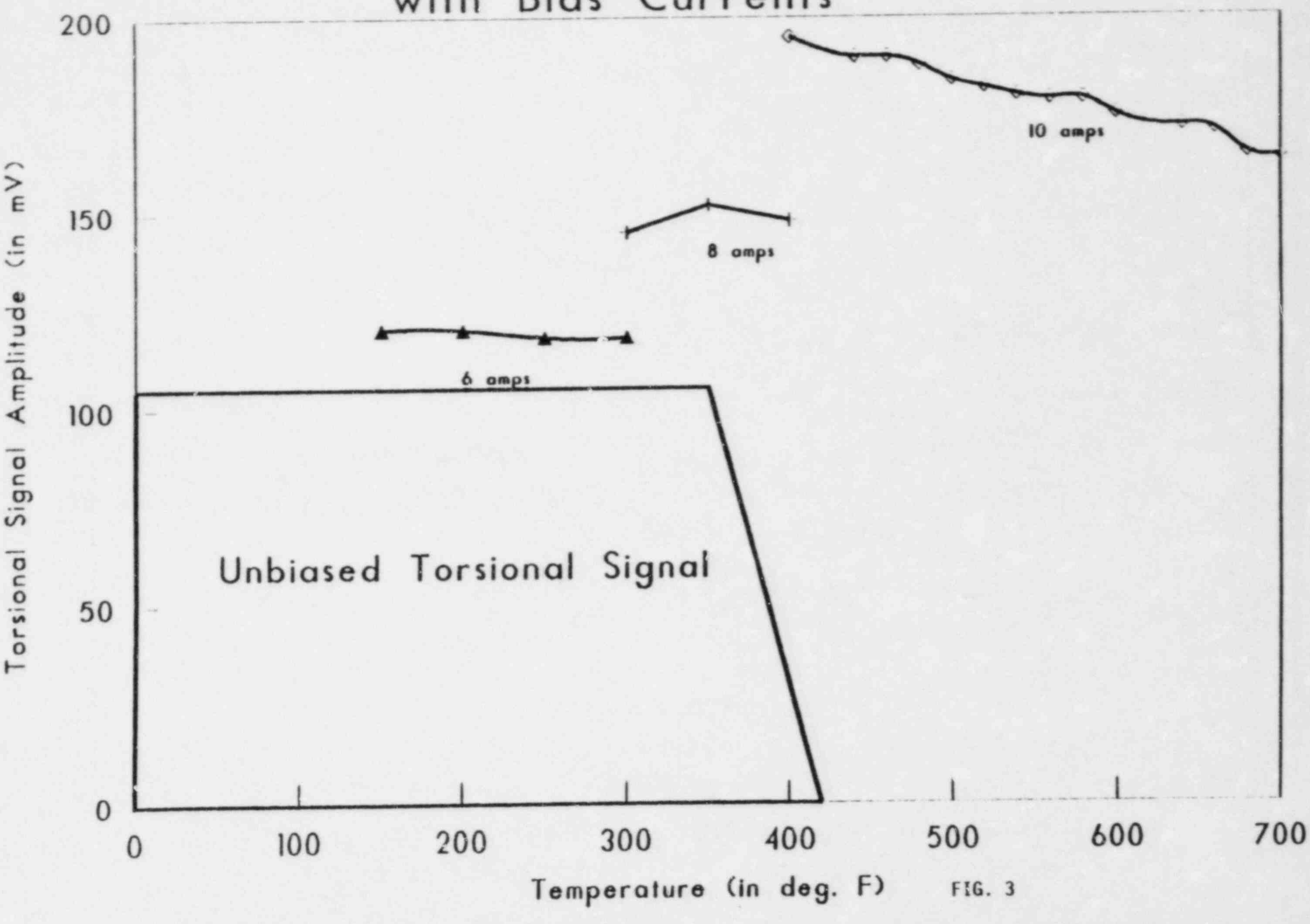


FIG. 3

Level and Density Indications While Adding Brine Solution to Ethyl Alcohol

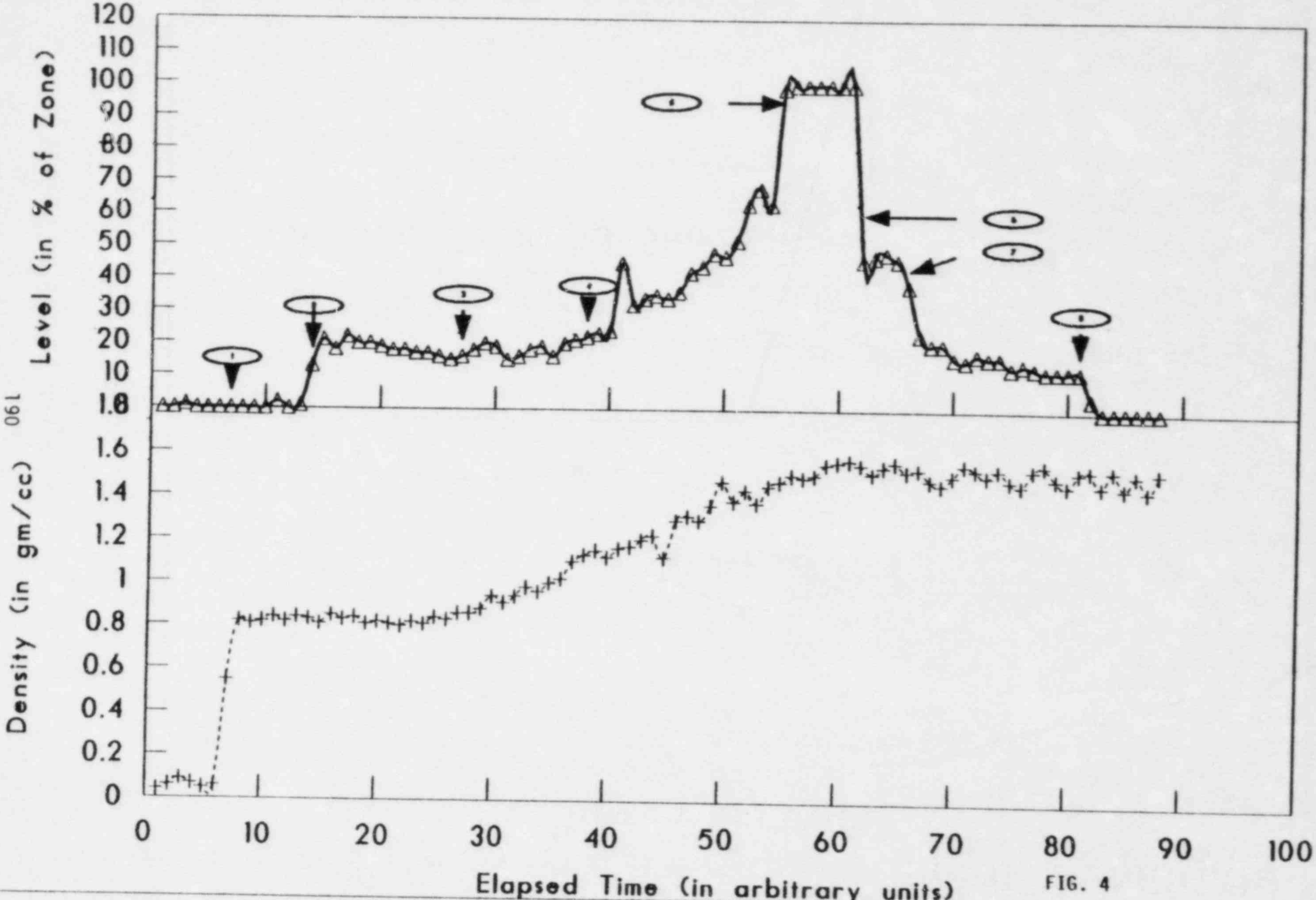


FIG. 4

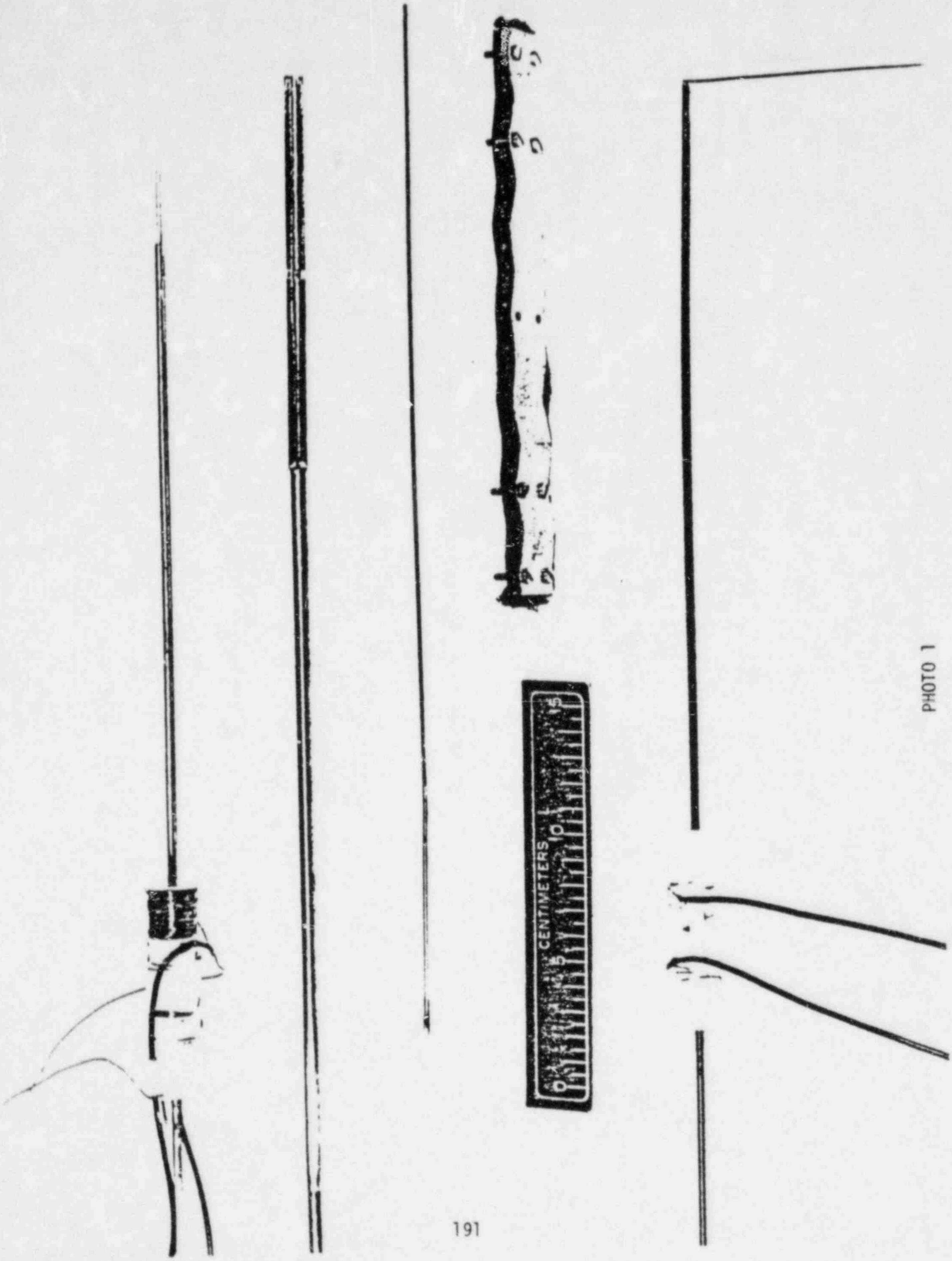


PHOTO 1

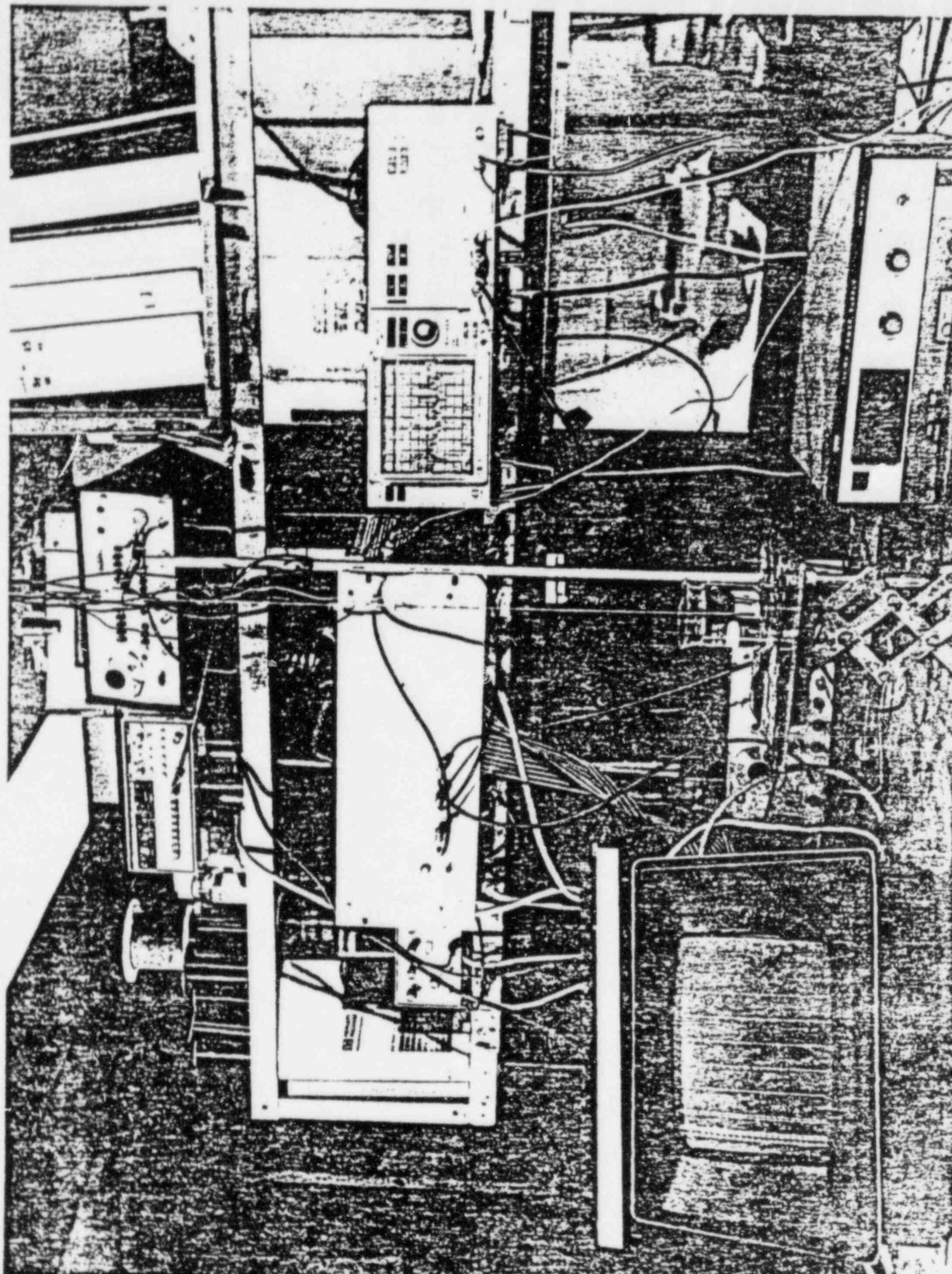


PHOTO 2

The Development of a Non-invasive Liquid Level and Density Gauge for Nuclear Power Reactors (1981-82 Progress Report)

by

W. A. Jester
A. J. Baratta, Jr.
G. Imel
A. H. Foderaro
E. S. Kenney
I. B. McMaster
E. Okyere
M. L. Gundy

Nuclear Engineering Department
The Pennsylvania State University

I. INTRODUCTION

The accident at Three Mile Island Unit-2 (TMI-2) demonstrated the need for pressure vessel water level instrumentation. In response to this need, a number of techniques have been proposed for level and/or density measurements. This includes heated thermocouples, differential pressure detectors and a variety of other methods. Each of these have serious limitations which the authors feel limit their usefulness under the multitude of conditions found during normal and accident conditions. In this paper we describe a non-invasive system which can be used with the reactor at power or shutdown. In addition, to level information, the proposed system has been shown to be capable of density measurement.

Analysis of the TMI-2 source range detector (SRD) revealed that the instruments were sensitive to both changes in coolant density and level during the accident period.[1,2] Analysis of the detector output prompted the authors to perform a series of tests at The Pennsylvania State University's Breazeale Nuclear Reactor (BNR). Their early tests show that neutron measurements can detect water level variations external to a reactor such as can occur in the downcomer of a pressure vessel.[2,3]

This project was designed to further explore the feasibility of constructing a power reactor density and level gauge capable of using the neutrons which penetrate a reactor's pressure vessel. Experimental tests conducted at the BNRL and at the LOFT facility and theoretical work and computer modeling conducted using the computational facilities of both Penn State and Argonne National Laboratory have further demonstrated the feasibility of such a system.

II. RELEVANT INFORMATION FROM THE TMI-2 ACCIDENT AND THE NSAC ANALYSIS OF ACCIDENT CONDITIONS[2]

The data collected by the TMI-2 SRD during the accident illustrates the type of data which could be obtained from one of the detectors in the proposed gauge under loss of coolant conditions. Figure 1 shows the SRD readings as a function of time after the turbine trip.[4] During the first 20 minutes, the SRD indicated a normal decay curve, and thus was seeing no effect from the loss of coolant to this point. Starting at approximately the 20 minute point, point A in Figure 1, the signal began to depart from the normal decay curve by the initiation of steam voids in the system, thereby increasing the fast neutron leakage. From this point (A) until the time (B) when the reactor coolant pumps were turned off about 100 minutes after the trip, the SRD acted as a density gauge. Figure 2 shows the relationship between the SRD response and the percent voids in the core and downcomer region as computed by the Nuclear Safety Analysis Center (NSAC).[5] This figure shows the type of response characteristic of a transmission-type density gauge. It should be noted that above 80% voiding, fast neutrons from the ABC start-up sources rather than the photoneutrons, would account for the origin of most of the monitored neutrons.

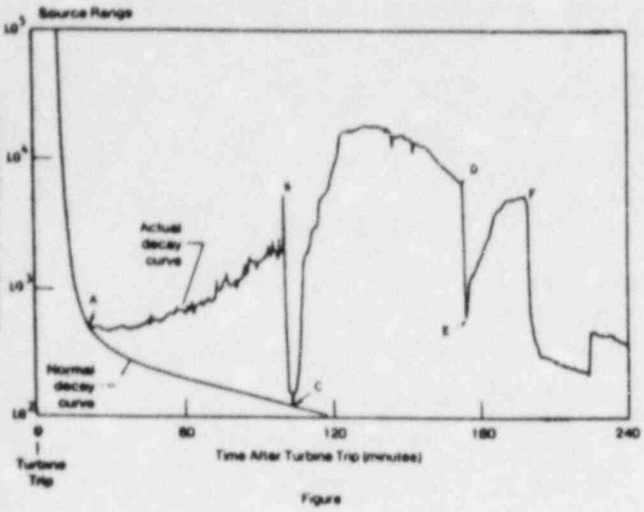


FIGURE 1. SOURCE RANGE DETECTOR READING AS A FUNCTION OF TIME AFTER TURBINE TRIP(4)

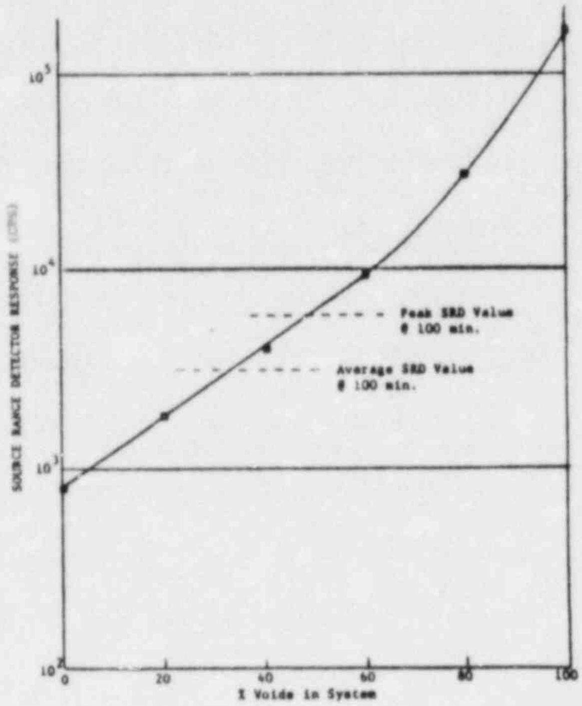


FIGURE 2. SOURCE RANGE DETECTOR RESPONSE TO HOMOGENEOUS VOIDING(5)

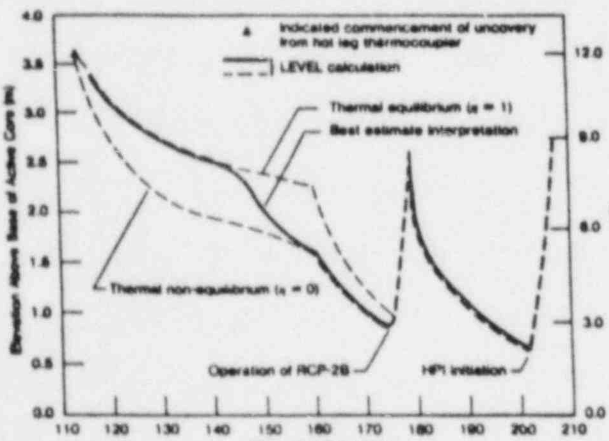


Figure 3. Calculated level path during uncovery of TMI-2 core. Solid curve is best estimate calculation.(6)

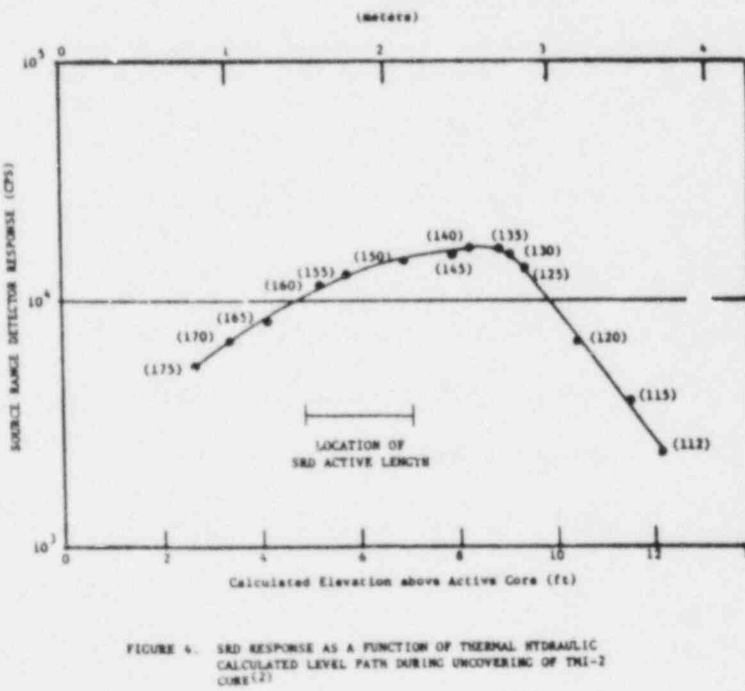


FIGURE 4. SRD RESPONSE AS A FUNCTION OF THERMAL HYDRAULIC CALCULATED LEVEL PATH DURING UNCOVERING OF TMI-2 CORE(2)

When the last of the operating coolant pumps was shutdown at 100 minutes into the turbine trip, there was a rapid drop in the SRD reading (B and C in Figure 1). The SRD reading momentarily returned to that of the normal shutdown curve. At this point, the voids rose to the top and coolant flowed into the core from the hot legs, producing a solid water condition as seen by the SRD.

During the next 75 minutes (C and D), the water level in the core region and downcomer was steadily being lowered as the water was being boiled off. During this period, the water in the downcomer, being cooler and thus more dense, would be somewhat lower than that in the core region. Figure 3 shows the best estimate of the actual core water level as computed by NSAC from thermal-hydraulic consideration from 112 to 205 minutes after turbine trip.[6] It should be noted that core uncovery did not occur until 112 minutes into the accident. Thus Figure 1 shows that the SRD began to respond to liquid level change when the coolant was still above the core.

If one takes NSAC's best estimated coolant level between 112 and 175 minutes after the trip and plots this versus the source range detector response during this same period, one obtains the curve shown in Figure 4.[2]

This curve indicates that the number of fast neutrons reaching the SRD increases exponentially until the water falls to a level near the upper level of the sensitive region of the detector. It remains about constant as the water level passes the detector, and then rapidly falls off again as the water level drops below the SRD.

It was proposed by the authors of this paper that the initial rise in measured activity was caused by the removal of the water moderator from between the primary sources of fast neutrons and the SRD. The fall-off of the signal as the water falls below the detector is believed to be due to the movement of the fast neutron fission and photoneutron sources away from the detector. Thus, the output of a single detector such as the SRD would produce an ambiguous signal since one might not know which side of the response curve one was on and could not tell whether or not the coolant level was rising or falling.

At about 175 minutes after the trip, the operators, for a brief period, started a reactor coolant pump, sending a slug of cold water into the downcomer and essentially filling it. This caused a rapid drop in the SRD reading with the return of a moderator to the downcomer (D-E in Figure 1.). During the period of time between 173 minutes after turbine trip and the 200 minute point (E-F), at which time the high pressure injection flow was initiated, the water level in the downcomer was not in equilibrium with the water level in the reactor vessel and the SRD response did not follow that indicated by Figure 4. This situation poses the question of how to interpret the output of the proposed gauging system when the downcomer water level is not in hydraulic equilibrium with that in the core.

III. NON-INVASIVE LIQUID LEVEL AND DENSITY GAUGE CONCEPT[2]

Briefly the basic concept on which this work is based is the use of a series of neutron detectors (fission chambers) positioned vertically in an existing instrument well of a power reactor. These detectors would be shielded from low energy scattered neutrons streaming within the air-filled annulus surrounding the pressure vessel and made sensitive primarily to the fast neutrons penetrating directly from the reactor region. In conjunction with existing reactor instrumentation, these detectors should give an early warning of an accident by detecting abnormal boiling in the reactor. The system should be able to resolve the ambiguity caused by the change of in-core liquid level since the output of adjacent detectors should allow the determination of where one is on a response curve of a specific detector. Each detector should have a different response curve depending on the neutronic and hydraulic condition of the core. The response curve for a given core condition should vary depending on the detector's vertical position in respect to the core.

In this concept, systematic changes such as changes in coolant boron concentration, reactor neutron source strength, and density changes due to changes in coolant temperature would be compensated for by normalizing the output of the higher positioned neutron detectors against that of the bottom-most detector. This reference detector would respond to the various systematic changes but would be the last detector to see changes in water density or level.

The object of the authors' research effort to date has been to verify this concept.

IV. COMPUTATIONAL MODELING OF THE TMI-2 CORE

In an attempt to analytically describe the response of the proposed water level gauge to the TMI-2 accident, project personnel conducted a series of calculations using an analytical model of the TMI-2 reactor developed by the Applied Physics Division of Argonne National Laboratory.[7] These calculations were utilized by project personnel to determine how a vertical string of three detectors would respond to changes in water level in the reactor core and/or downcomer. The three detectors modeled were axially positioned in an instrument well so that one was adjacent to the bottom of the core, another adjacent to the midpoint of the core, and a third adjacent to the top of the core. In these calculations, the detectors were assumed to be sensitive to epithermal neutrons with energies greater than 1.85 eV. Figure 5 shows the computed detector response for both downcomer and core region voiding while Figure 6 shows detector response for only downcomer voiding. A series of point kernel calculations for the same conditions were also conducted by project staff in an effort to understand the behavior of the responses of the three detectors in the ANL calculations. As seen in Figures 7 and 8, these response curves are similar to those shown in Figures 5 and 6 with the exception that they do not show a peak at about the 300 cm level for the lower detector. It is now believed that this segment of the ANL calculated response curves may result from a deficiency in the Argonne model.

These calculations confirm the initial concept that the exponential rise initially seen in both the core and downcomer voiding and the solely downcomer voiding cases is purely a shielding effect caused by the loss of water between the core and the detector. For the downcomer voiding case, this levels off when all of the direct neutron pathways are opened up. For the case of core and downcomer voiding, the drop-off is primarily caused by the loss of neutron source strength in the core region starting with the top of the core.

V. EXPERIMENTAL WORK CONDUCTED AT THE BREAZEALE NUCLEAR REACTOR

An experimental level gauge apparatus was constructed which would be positioned adjacent to the BNR TRIGA Reactor Core (Figure 9). The apparatus with the reactor simulates a segment of a power reactor out to its biological shield.[8] The water level can be varied in Chamber A which simulated changes in downcomer and to some extent core water level. A boral plate was placed between Chambers A & B to simulate the thermal neutron absorption caused by the pressure vessel wall. Chamber B simulates

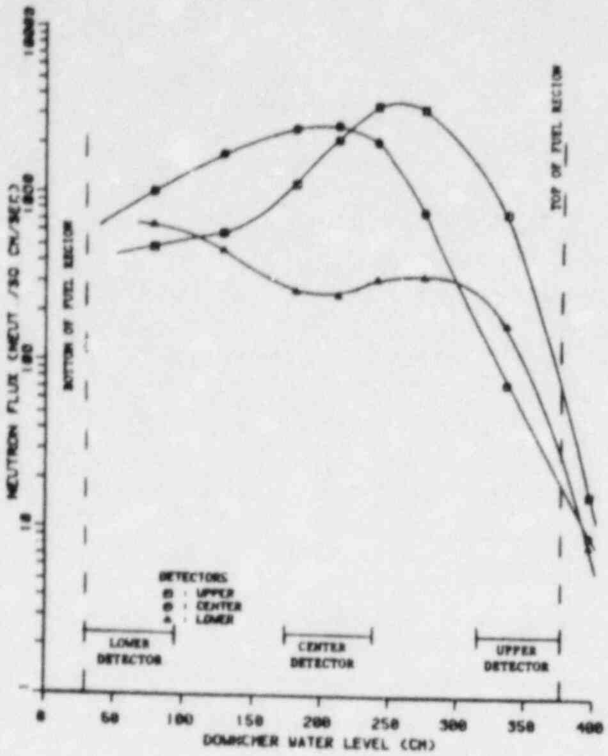


FIGURE 5. EPITHERMAL NEUTRON FLUX LEVELS RESULTING FROM CORE AND DOWNCOMER VOIDING OF TMI-2 (ANL CALCULATIONS)

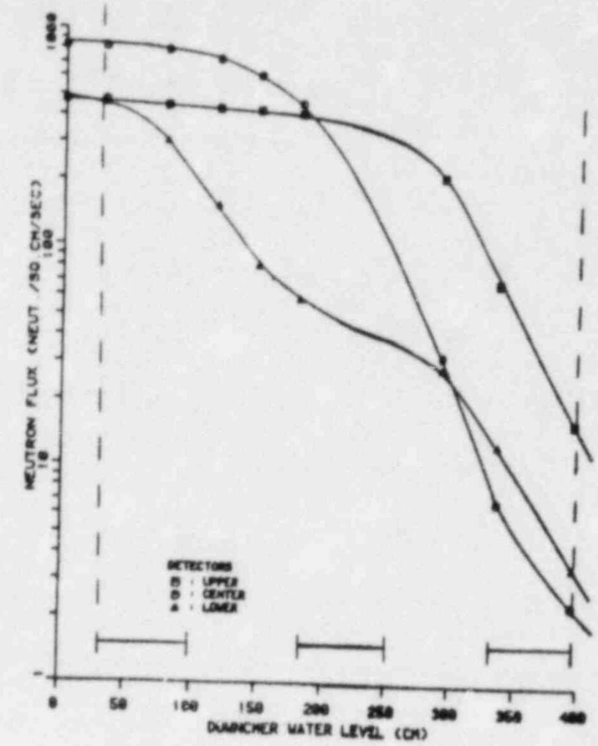


FIGURE 6. EPITHERMAL NEUTRON FLUX LEVELS RESULTING FROM DOWNCOMER VOIDING OF TMI-2 (ANL CALCULATIONS)

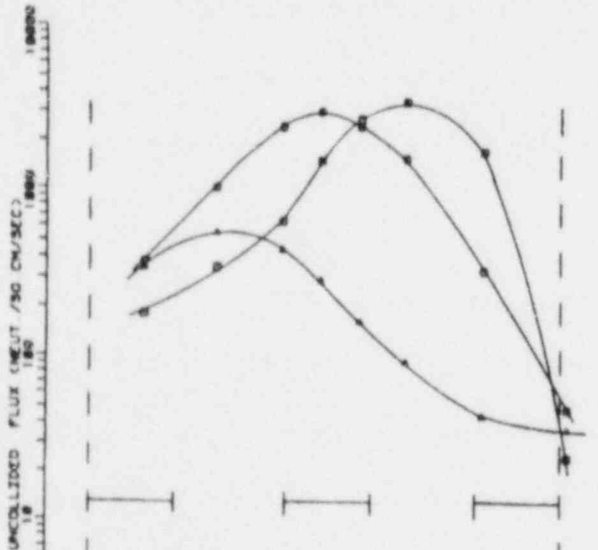


FIGURE 7. UNCOLLIDED NEUTRON FLUX LEVELS RE...TING FROM CORE AND DOWNCOMER VOIDING OF TMI-2 (PS. CALCULATIONS)

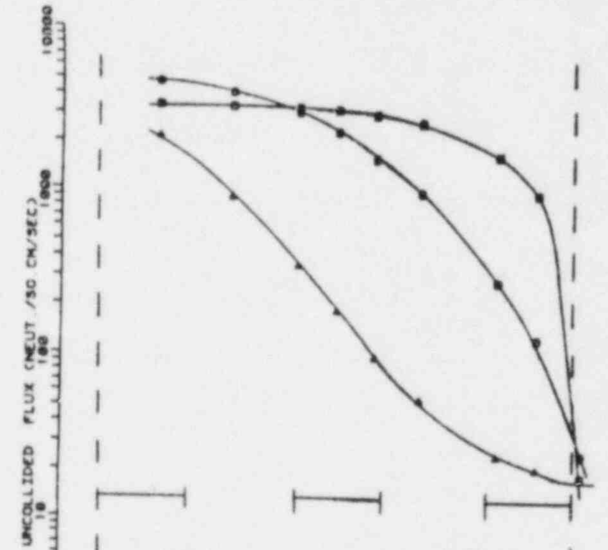


FIGURE 8. UNCOLLIDED NEUTRON FLUX LEVELS RESULTING FROM DOWNCOMER VOIDING OF TMI-2 (PSU CALCULATIONS)

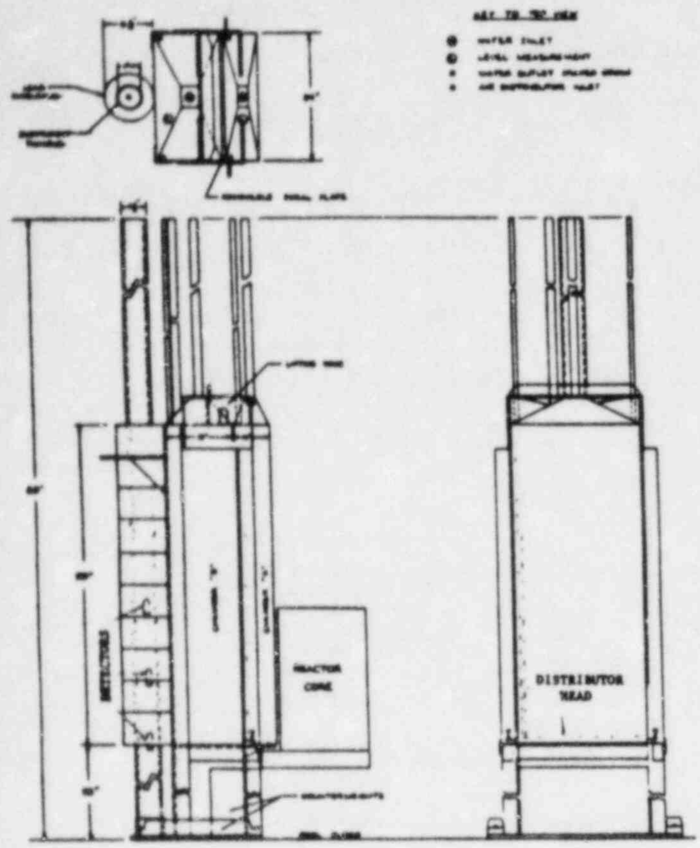


FIGURE 9. LEVEL GAUGE APPARATUS AND PENN STATE TRIGA REACTOR CORE SCALE = 1½" = 1'

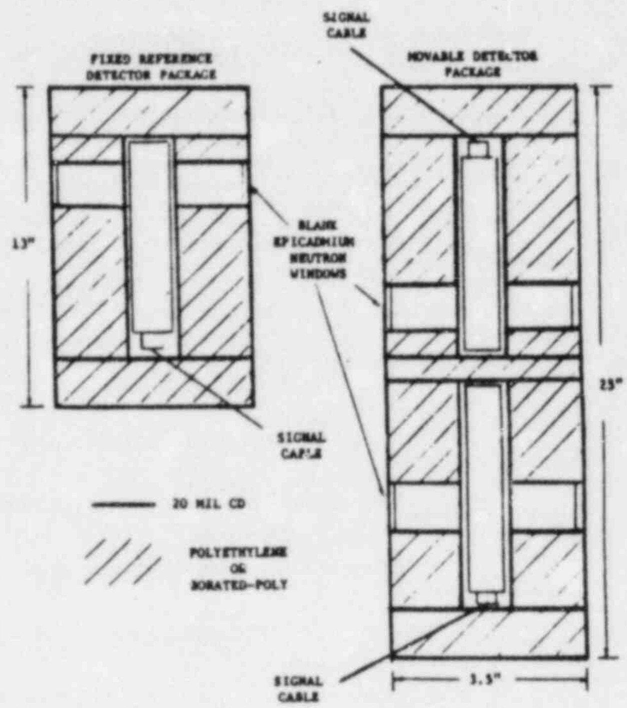


FIGURE 10. BLANK WINDOW LEVEL GAUGE DETECTOR PACKAGING

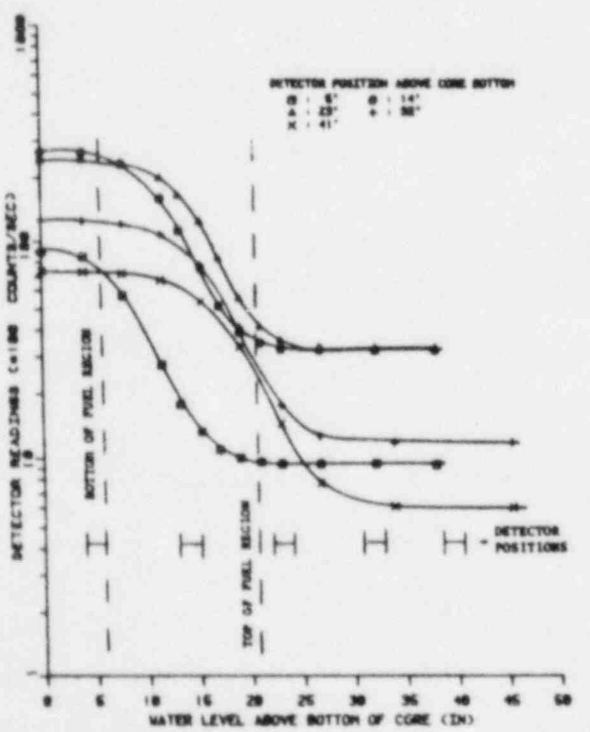


FIG. 11. CD-B-POLY SHIELDED (2" WINDOW) DETECTOR RESPONSE VS WATER LEVEL IN LEVEL GAUGE APPARATUS COUPLED TO THE CORE

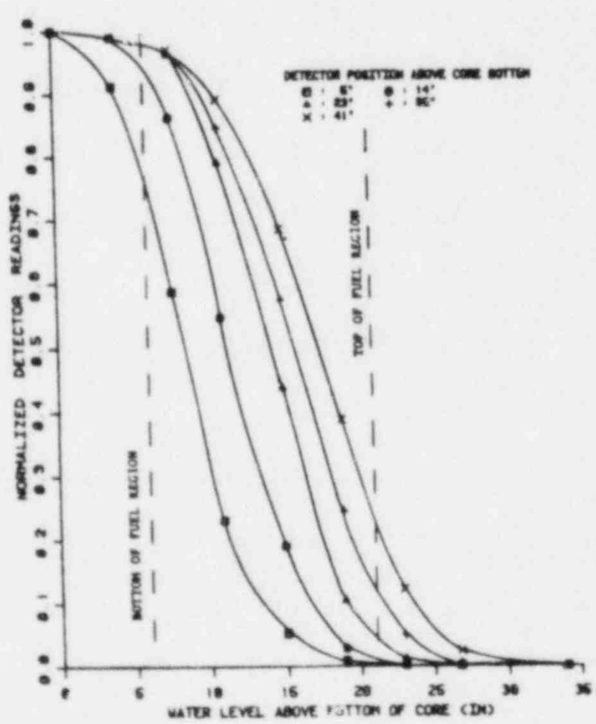


FIG. 12. CD-B-POLY SHIELDED NORMALIZED DETECTOR RESPONSE VS WATER LEVEL IN LEVEL GAUGE APPARATUS COUPLED TO THE CORE

the air-filled annulus that exists between the outside of a pressure vessel wall and the biological shield. The vertical instrument tube allows for the positioning of a vertical string of fission chambers used in the experiments. The three fission chambers used in this work were Reuter Stokes, Model R/S P6-0805-135. These chambers have an active length of five inches. Nitrogen gas can be bubbled into the bottom of Chamber A by means of a distributor head to simulate steam voiding which occurs during the early phases of a LOCA.

A variety of fission chamber packaging arrangements made from various combinations of cadmium, polyethylene, and borated-polyethylene were tested in this apparatus. The tests were conducted in order to optimize the detection of the higher energy neutrons coming more directly from the core while screening out the lower energy neutrons streaming in the air-filled annulus and instrument tube.

Figure 10 shows a diagram of one of the more successful detector packages tested in the level gauge apparatus. Each detector is wrapped in 20 mil of cadmium and then surrounded with borated-polyethylene (5%-boron) except for a 2-inch horizontal epicadmium neutron window region. The whole assembly including the window is then wrapped in 20 mil cadmium. Other experiments indicated that similar results can be obtained by substituting polyethylene for the borated-polyethylene, thereby removing the potential problem of having alpha heating melt the polyethylene.

In the instrument tube, a detector is kept in a fixed position adjacent to the bottom of the core and serves as a reference detector while the movable detector package with its two detectors can be moved to various positions above the reference detector. Figure 11 shows the response curves obtained with this package using the level gauge apparatus. This data can be processed in a number of different ways. For example, in Figure 12, the original data is normalized between zero and one to remove systematic variations in flux levels from point-to-point along the horizontal axis. This plot shows clearly that the higher the detector position, the sooner it sees a change in liquid level in Chamber A. Figure 13 is a replot of the same data showing the relative readings of each detector for each water level condition. It can be seen that as the water level drops below each detector, the detector shows a large response.

In the initial concept on which this project is based, it was proposed that the data from all other detectors should be normalized against the bottom-most detector reading since the output of this detector would be the last one affected by core boiling or level changes. This type of normalization would minimize the interference from system effects such as changes in power level, coolant water boron concentration, and water temperature. To evaluate this concept, the detector readings from the 5-inch level were divided into the detector readings at the 14, 23, 32, and 41-inch levels for the various water level conditions. These results are plotted in Figure 14. These ratio plots show sharp peaks that vary somewhat with detector position. A possibly more useful relationship is shown in Figure 15. This is a plot of the ratios of the responses of adjacent detectors. While these peaks are not as large as those in Figure 14, there is a better correlation of the peak location with detector position and water level.

Tests were conducted with the level gauge apparatus to evaluate the concept of monitoring density changes due to boiling. In these tests, the level gauge apparatus was employed as previously described with the exception

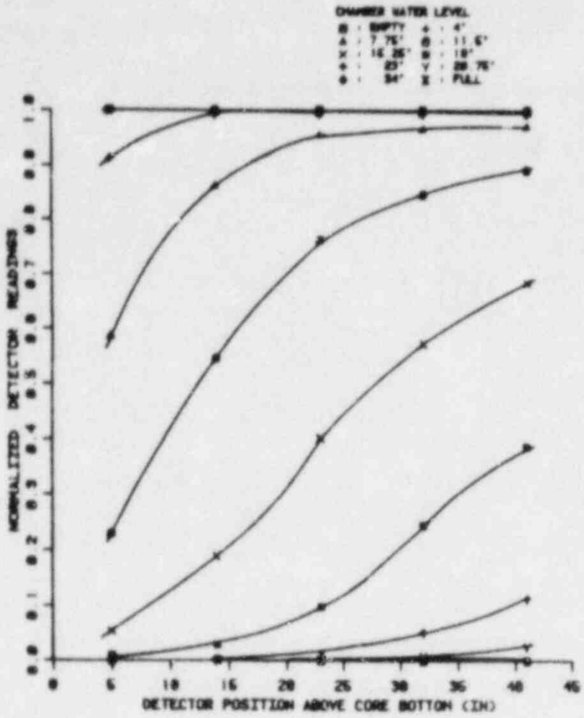


FIG 13. CD-6-POLY SHIELDED NORMALIZED DETECTOR RESPONSE VS DETECTOR POSITION FOR VARIOUS WATER LEVELS IN LEVEL GAUGE APPARATUS

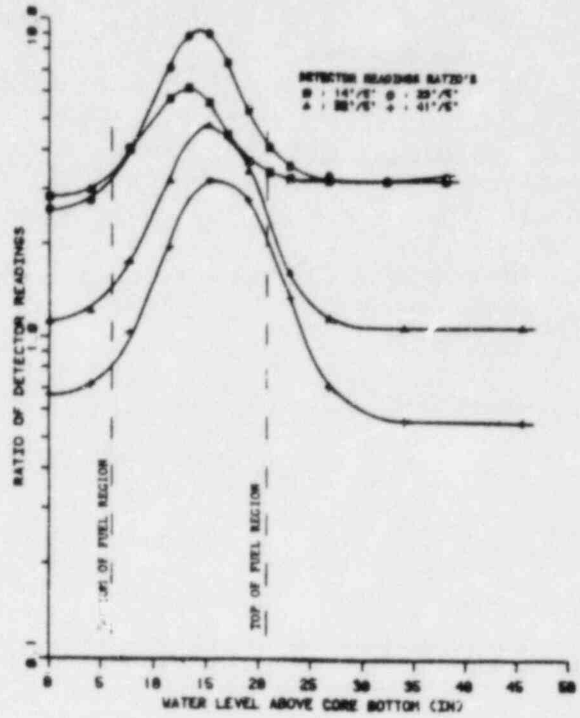


FIG 14. CD-0-POLY SHIELDED DETECTOR RESPONSE DEVIDED BY THE RESPONSE OF THE REFERENCE DETECTOR

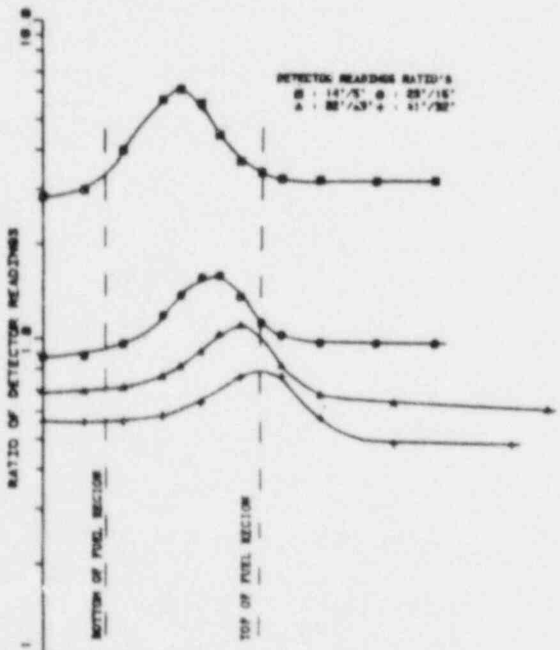


FIG 15. CD-0-POLY SHIELDED DETECTOR RESPONSE DEVIDED BY THE RESPONSE OF THE DETECTOR POSITIONED BELOW IT

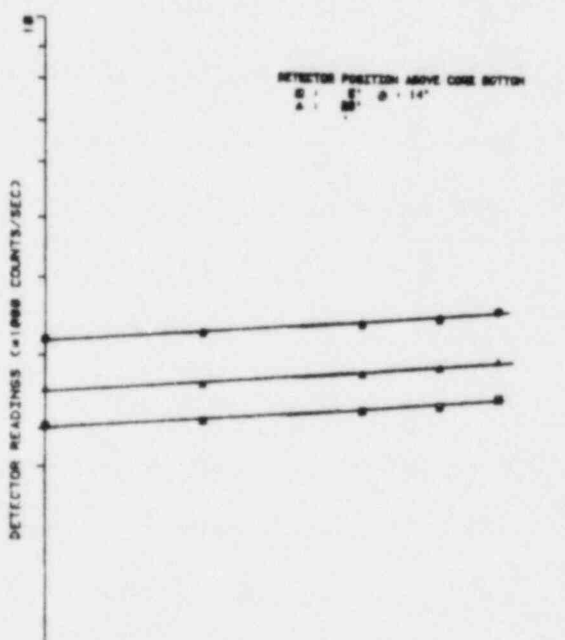


FIG 16. CD-0-POLY SHIELDED DETECTOR RESPONSE TO CHAMBER A VOIDS WITH NITROGEN GAS BUBBLES

that a uniform distribution of nitrogen gas bubbles was produced in the water-filled chamber A by injecting the gas into the bottom chamber by means of the distribution head. Three detectors packaged as previously described were centered on the 5-inch, 14-inch, and 23-inch positions, respectively. Figure 16 shows the results of these measurements with the percent voiding varying from 0 to 4.6%. Each detector sees an exponential increase in epicadmium neutron intensity of between 16 and 17%. Such a behavior is expected since the attenuation of epicadmium neutrons should decrease exponentially with the loss of water between the detector and the core.

A similar experimental increase in the source range detector output of the TMI-2 core was observed during the accident prior to the shutdown of the coolant pumps, except in this case a voiding of about 4.6% produced a 40% increase in detector output (see Figure 2). At TMI-2 boiling was occurring throughout the core and thus a larger response is to be expected.

The ability of this concept to detect in-core boiling prior to the formation of a bubble at the top of the pressure vessel is one of the most important aspects of this concept.

Discrete ordinate calculations were undertaken to aid in explaining the data obtained from the level gauge calculation, the one-dimensional ANISN code, and the two-dimensional DOT-4 code were employed. The results of these calculations compared well with the experimental results obtained using the level gauge apparatus.

VI. LOFT TEST DATA

A member of the project staff participated in the most recent LOFT test which took place on June 16, 1982. This test (L2-5) was a large break and resulted in repeated uncovering of the core. Data from the source range and intermediate range detectors and the fuel cladding thermocouples were provided by EG&G, Idaho. The test simulated a 200% cold leg break with immediate reactor scram and Emergency Core Cooling Injection. Figures 17 and 18 show the fuel clad thermocouple response of thermocouples TE-5H07-058 and TE-5H07-008, at 58" and 8" above the bottom of the core, respectively. It can be seen that there was immediate core uncovering. The core remained uncovered for about 2 minutes before it was totally covered with water. The low pressure injection system was then turned off and a second core uncovering occurred at about 200 seconds, a partial uncovering at about 450, and possibly a partial uncovering at about 1500 seconds before the reactor was stabilized.

Unfortunately the source range detectors were saturated until 700 seconds into the test; thus the response curves of the less sensitive intermediate range detector had to be used in this study. The response of this unshielded intermediate range detector (RE-T-86-3) (see Figure 19) located outside the pressure vessel on the midplane of the core shows a definite correlation with the thermocouple data. As may be seen, the first spike that occurred during the initial transient is clearly visible as is the spike of the second core uncovering. In the third (partial) uncovering, both the lower thermocouple and the intermediate range detector show a double peak structure not seen in the output of the upper thermocouple. It also appears that the response function of the intermediate range detector is considerably more rapid than that of the thermocouples. This may result from the detector sensing the density changes that precede the level change while the thermocouples are still receiving adequate cooling. The fourth uncovering took place only at the top of the core and was not seen by either the intermediate range detector

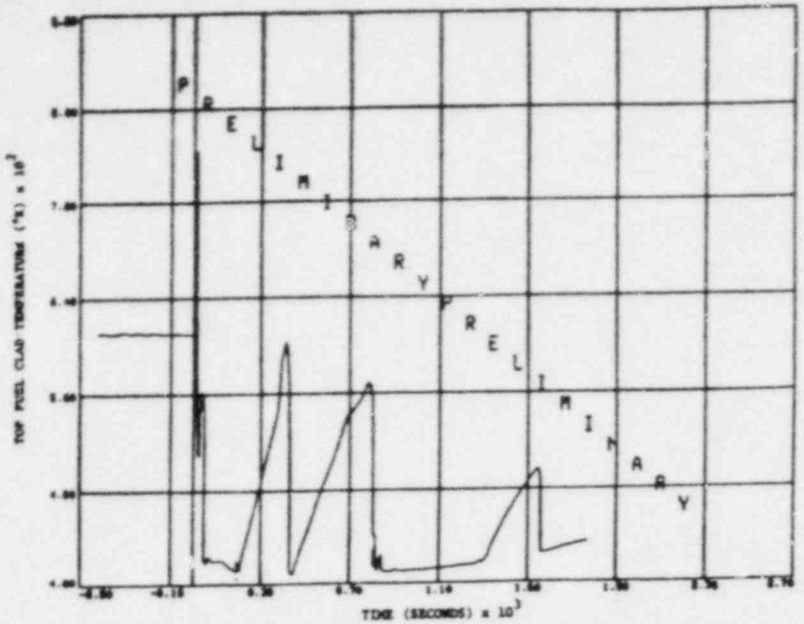


FIGURE 17. TOP FUEL CLAD THERMOCOUPLE (TE-5B07-058) RESPONSE VS. TIME FOR LOFT LOSS OF COOLANT EXPERIMENT L2-5(9)

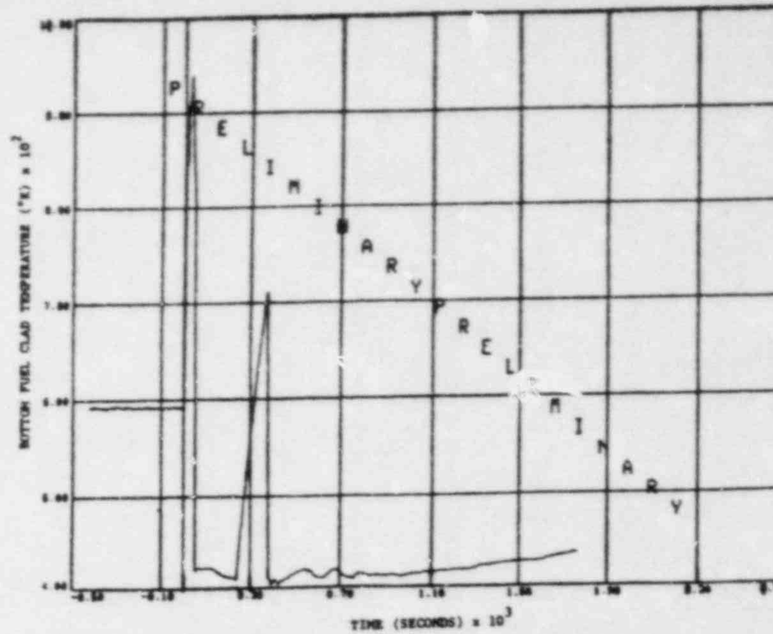


FIGURE 18. BOTTOM FUEL CLAD THERMOCOUPLE (TE-5B07-008) RESPONSE VS. TIME FOR LOFT LOSS OF COOLANT EXPERIMENT L2-5(9)

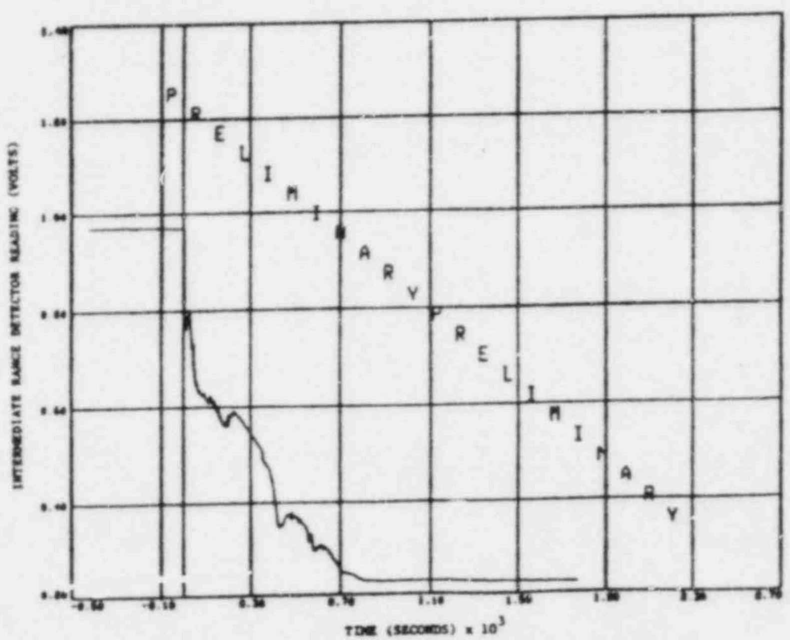


FIGURE 19. INTERMEDIATE RANGE NEUTRON DETECTOR RESPONSE VS. TIME FOR LOFT LOSS OF COOLANT EXPERIMENT L2-5(9)

or the lower thermocouples. It should be noted that by this time, the neutron flux level had dropped to the point that the intermediate range detector had lost most of its sensitivity

VII. CONCLUSIONS FROM THE WORK TO DATE

The experimental and computational work to date indicates that each epithermal neutron detector located at a different vertical position in a pressurized water reactor instrument tube will produce a unique response depending on the type of voiding taking place in the core. When boiling occurs, as in the early stages of LOCA, there will be an exponential rise in detector response, seen by all detectors, as some of the water shielding is lost between the core neutron source and the detectors.

As liquid level falls, there will also be an exponential increase in detector response, again as a result of the loss of neutron shielding caused by the removal of water between the core and a detector. Thus the higher the detector is positioned, the sooner it will respond to the loss of water. In the case involving only the voiding or filling of the downcomer, the maximum detector signal would be seen only at the time that the downcomer is completely voided.

For the case of the simultaneous voiding of the core and downcomer, there will be the initial rise in detector response caused by the loss of the shielding water. Then the detector response curve will fall off due to a reduction in the neutron source intensity. This reduction is due to the loss of (γ,n) neutron source from the water and the reduction of core reactivity as the water level falls in the core region. The position of the peak will depend on the vertical position of the detector with the higher detector peaking first.

The LOFT studies show qualitatively the same behavior observed in the TMI-2 calculations and experiments. While detailed analytical work has yet to be done, good agreement was obtained between in-core thermocouples and external neutron detectors when coolant level changes occurred in the LOFT system.

Based on the above work, the authors consider that a pressure vessel level gauge using externally mounted neutron detectors can provide unique and unambiguous water level information. In addition, such a system would have the unique ability to discern density changes in a reactor system which occur during the early stages of a LOCA. For a more detailed description of this work, see reference 10.

VIII. ACKNOWLEDGMENTS

This work was supported by the U.S. Nuclear Regulatory Commission (NRC) under grant No. NRC-G-04-81-024. The opinions, findings, conclusions, and recommendations expressed herein are those of the authors and do not necessarily reflect the views of the NRC.

IX. REFERENCES

1. Schultz, M. A., W. A. Jester, and I. B. McMaster, Proposal for a Non-invasive Liquid Gauge for Reactor Pressure Vessels, Proced. U.S.N.R.C. meeting on Reactor Vessel Liquid Level Measurements, NUREG/CP-0016, pp. 75-89 (1980).

2. Jester, W. A., A. J. Baratta, et al., A Concept for a Fast Neutron Non invasive, Liquid Level and Density Gauge for Nuclear Power Reactors, Presented at the U.S. NRC 9th Water Reactor Safety Research Meeting (1981).

3. Edwards, J. A., A Non-invasive Liquid Level Gauge Comparison for Nuclear Power Reactor Pressure Vessels. A M.S. paper in Nuc. E., The Pennsylvania State University, 1981.

4. "Analysis of Three Mile Island-Unit 2 Accident," NSAC-80-1, NSAC-1 Revised, March 1981.

5. "Neutron Transport Calculations," a NSAC Draft Report, Undated.

6. "Calculations of the Core Temperature Transient during the Early Phase of Core Uncovering at TMI-2," a NSAC Draft Report, December 1980.

7. Malloy, D. J. and Y. I. Chang. Neutron Analysis of the Three Mile Island Unit-2 Ex-core Detector Response, ANL-81-75 (1981).

8. Jester, W. A., and A. J. Baratta, Jr., Experimental Phase of the Development of a Non-invasive Liquid Level and Density Gauge for Nuclear Power Reactor Vessels, 3-month Progress Report, Contract No. NRC-G-04-81-024, submitted to NRC Office of Nuclear Regulatory Research, Jan. 1982.

9. Data provided by EG&G Idaho, June 1982.

10. Baratta, A. J., Jr., Jester, W. A., et al. Final preliminary Progress Report on the Development of a Non-invasive Liquid Level and Density Gauge for Nuclear Power Reactors, Submitted by Penn State Nuc. Engr. Dept. to NRC, Sept. 1982.

Non-Condensible Gas Fraction Prediction Using Wet and Dry Bulb Temperature Measurements at Elevated Temperature and Pressure.

John K Bowman*

Peter Griffith**

Abstract

Using only wet and dry bulb temperature readings a method of calcuating air (or nitrogen) and water vapor concentrations is developed. It is appropriate for high temperature and pressure. The method works in up or down flow and in stagnant mixtures. Experiments show that compositions can be predicted to $\pm$ 4%.

*Polaroid Corporation, Cambridge, Massachusetts
Formerly Research Assistant, MIT Cambridge, MA.

**Professor of Mechanical Engineering
MIT, Cambridge, Massachusetts

Introduction

During various LOCA's air or nitrogen is likely to be found in the primary coolant system, either as a result of admission through the break, in large breaks, or through the discharge of the accumluators in any sized breaks. The ability to reject heat to the steam generator, the system pressure and therefore, the dryout level in the core and the circulation through the various loops all depend on where this gas is accumulated. At this time we have no means of predicting where this gas is or how it will be transported around the system. In order to run experiments to find this out, we need a method of determining the local gas composition. This work was undertaken to fill this need.

Similarly, no experiments have been run to determine whether or how much stratification will occur in a containment if a LOCA causes steam to be discharged into a containment. Containment heat transfer and local containment temperatures depend on whether stratification exists. If an experiment is run to see how significant stratification is, a method of determining local composition is needed. This device can be used in that experiment too.

In this note the method of calculating non-condensible gas composition using wet and dry bulb temperature readings will be presented. The experiments which show how well this method works will be described and finally some guidelines on system design will be given. The details are fully presented in reference (1).

Theory

The heart of the theory which allows one to use wet and dry bulb temperature readings to calculate non-condensible gas concentrations is the heat balance on the wet bulb thermometer. Figure (1) shows the fritted glass wick used in some of the experiments along with the heat and mass transfers occuring on this wick. Equation (1) is the heat balance. Substituting heat and mass transfer coefficients for the appropriate terms in eqaution (1) and using temperature and concentration driving potentials, one can solve for the vapor concentration at the surface of the wick - equation (2). The problem is how to evaluate the heat and mass transfer coefficients.

With a little reflection, one recognizes that there might be forced circulation over the wick, natural circulation, or simple conduction and diffusion. In addition, the forced or natural circulation could be either laminar or turbulent. All of these possibilities were looked at and it was found that the ratio $[(h_c + h_r) / h_d]$ in equation (2) is very little affected by the mode of transfer, that is laminar or turbulent or forced or natural convection. In addition, the conduction limits that could be imagined never actually occurred in the experiments apparently because the density differences between the wick and the bulb never quite went to zero everywhere in the vicinity of the wick. Yet another problem that was anticipated but did not arise was that the Grassoff Number might change sign. Let's consider this a moment.

The density in the vicinity of the wet bulb thermometer changes due to to two effects, reduced temperature which increases the density and increased water vapor concentration which decreases the density. Some analysis in the literature indicated that a reversal of the natural convection plume was a real possibility. It was never observed; the plume always went down. Apparently, the water vapor concentration at the surface of the wick never became high enough to reverse the density difference.

A trial and error procedure must be used to evaluate the composition at infinity because the properties are such complex functions of the composition, pressure and temperature. In essence, one uses an assumed composition at infinity plus the measured wet and dry bulb temperatures to substitute into equations (2), (4), (6) and Figure (2). When equation (2) balances, the appropriate composition has been determined.

The natural convection heat transfer equation was chosen because it yielded a slightly better correlation with the data than the turbulent heat transfer correlation. Whether equation (3) or equation (4) is used to relate to heat and mass transfer coefficients turns out be of very little consequence however, because the Prandl and Schmitt number ratio is very close to 1. Equation (4) was used for simplicity. The conduction-diffusion limit for zero flow which we were concerned about never appeared in the experiments so we didn't have to consider it. It appears that the density profile in the vicinity of the wick is never exactly flat.

Experiments

The apparatus used to perform the forced convection experiment is illustrated in Figure 3. A fan in the draft tube circulates the steam-nitrogen mixture over the wet and dry bulb thermometers. The comparable natural convection apparatus and associated thermocouple array are illustrated in Figures (4) and (5). The thermocouple array was used to see whether natural convection plume went up or down from the wet bulb. In both experiments it was necessary to insure that uniform temperature and composition existed throughout the vessel.

A comparison of the calculated and measured compositions is shown in Figure (6) using the recommended natural convection heat and mass transfer coefficients. The error is approximately $\pm$ 4%.

The experimental program showed several of the design features that must be adopted if this system is going to work. The cool, moist plume in natural convection always went down so it is necessary to place the dry bulb thermometer to the side so it will not be affected. In forced convection of course, the plume is downwind of the wet bulb so the dry bulb must be placed upwind. Splatter can be a problem depending on the set up; it may be necessary to splash shield over both the thermometers. Several methods of providing water to the wet bulb thermometer are considered.

Finding a suitable wick material was a time-consuming task. In this set of experiments, pressure went from atmospheric to 600 psig and the dry bulb temperature from 400°F to 600°F. Cotton cord, fiberglass and fritted glass were all tried. The cotton degraded quickly at 400°F. The fiberglass degraded slowly enough to allow the collection of the forced convection data. The fritted glass wick worked fine except that for some modes of operation, the wick was not wet 100%. The ratio (Ac/Ad) in equation (2) was therefore set to a value greater than 1. Recommendations for calculating how a design will perform are made in reference (1) along with the information on the axial conduction errors, transient response and observations of the behaviour when the wick is only partially wet.

Conclusions

1. An algorithm suitable for calculating the composition of air-water or nitrogen-water mixtures from wet and dry bulb temperature readings is given.

2. Compositions can be calculated to $\pm$ 4% independent of whether one has natural or forced convection is in up or down flow, the flow laminar or turbulent.

3. Recommendations for the design of such an instrument are given including thermocouple placement, wick material selection, and provision for supplying water.

4. A device for determining the composition of the gas phase suited to experiments in large integral test facilities can be designed with the information presented in reference (1).

REFERENCES

(1) Bowman, John K., "Non-condensible Gas Fraction Predictions Using Wet and Dry Bulb Temperature Measurements," PhD thesis in Mechanical Engineering, M.I.T., May 1982.
(Also to appear as a NUREG report in the fall or winter of 1982 or 1983).

(2) McAdams, W. H., "Heat Transmission," McGraw-Hill, New York, 154, p.173.

Equations

$$q_c + q_r = m\, h_{fg} \qquad (1)$$

$$C_{v,fs} = C_{v,wb} - \frac{h_c + h_r}{h_d} \; \frac{A_c}{A_d} \; \frac{1}{h_{fg}} (T_{db} - T_{wb}) \qquad (2)$$

$$\frac{h_c}{h_d} = \rho\, Cp \left(\frac{Sc}{Pr}\right)^{2/3} = \frac{k}{D} \left(\frac{Pr}{Sc}\right)^{1/3} \qquad (3)$$

$$\frac{h_c}{h_d} = \frac{k}{D} \left(\frac{Pr}{Sc}\right)^{1/2} \qquad (4)$$

$$h_r = \varepsilon_w\, \sigma\, (T_{db}^{\;4} - T_{wb}^{\;4})/(T_{db} - T_{wb}) \qquad (5)$$

FRITTED GLASS WET BULB WICK

LATER VERSION

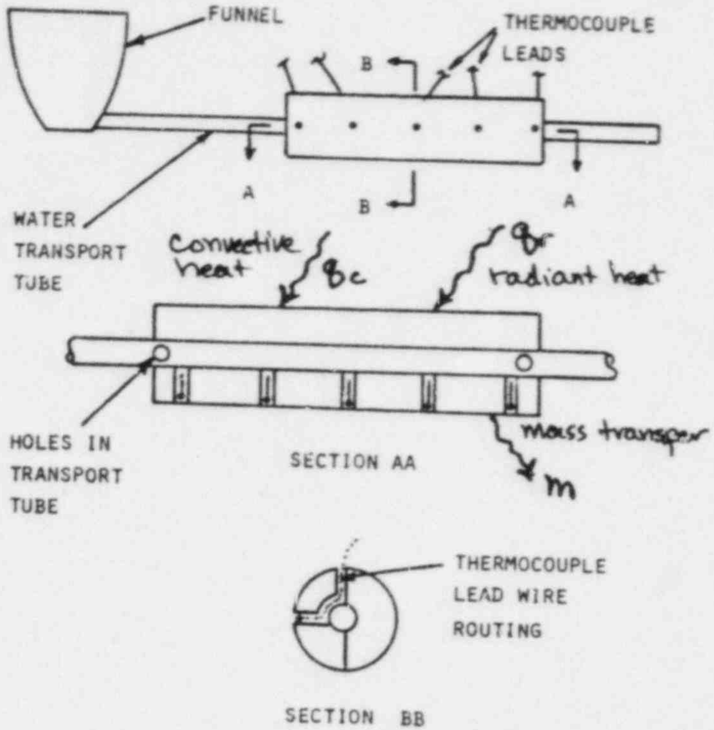


Figure 1

FREE CONVECTION HEAT TRANSFER FROM

HORIZONTAL CYLINDERS

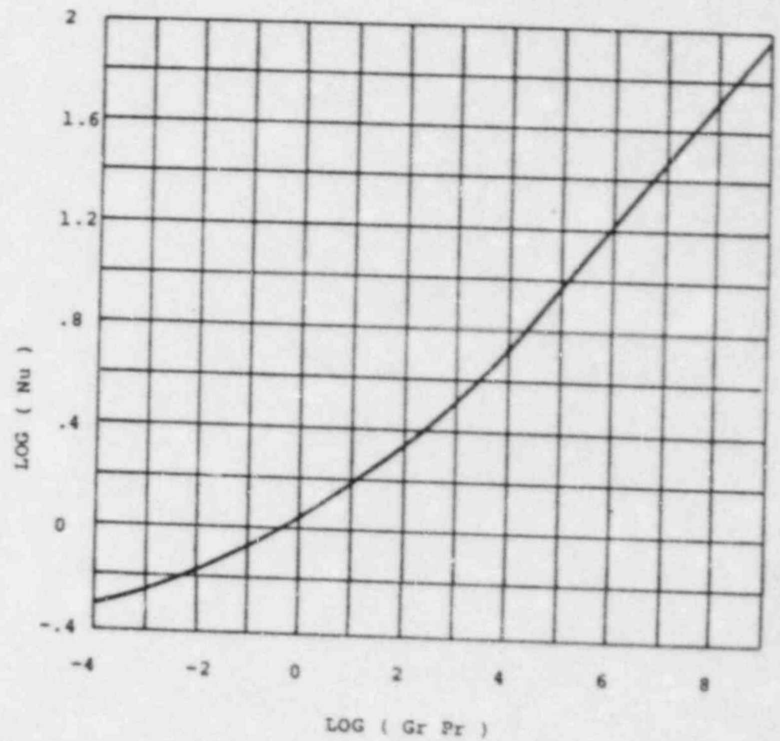


Figure 2

FORCED CONVECTION TEST APPARATUS

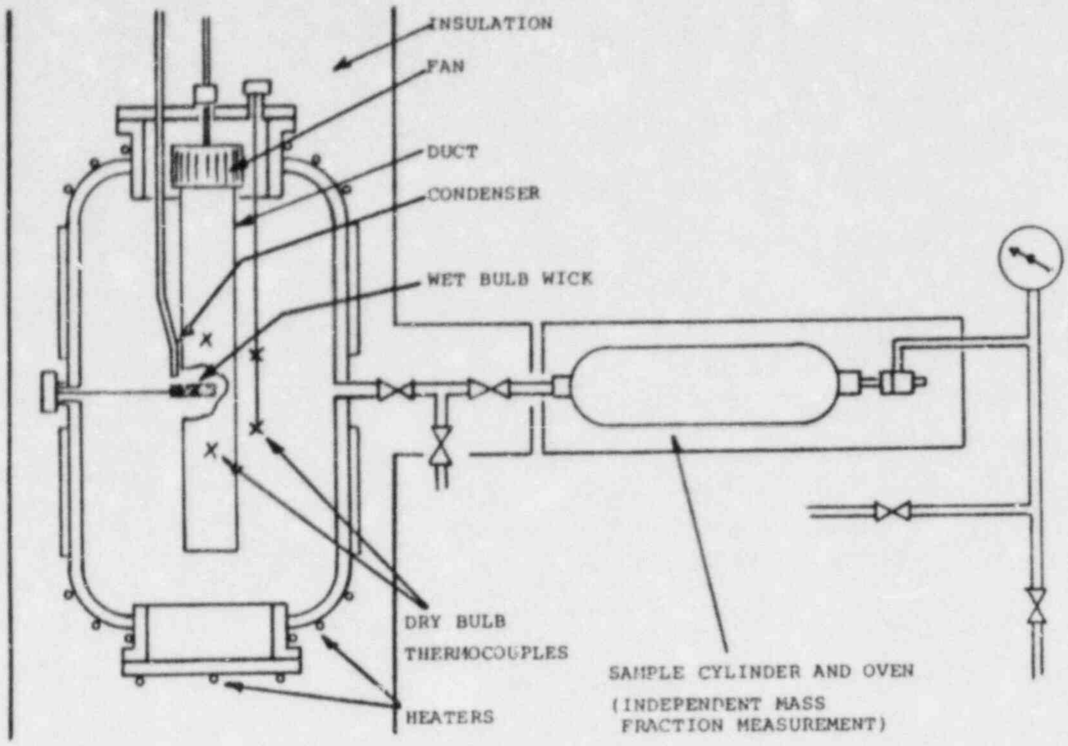


Figure 3

NATURAL CONVECTION TEST APPARATUS

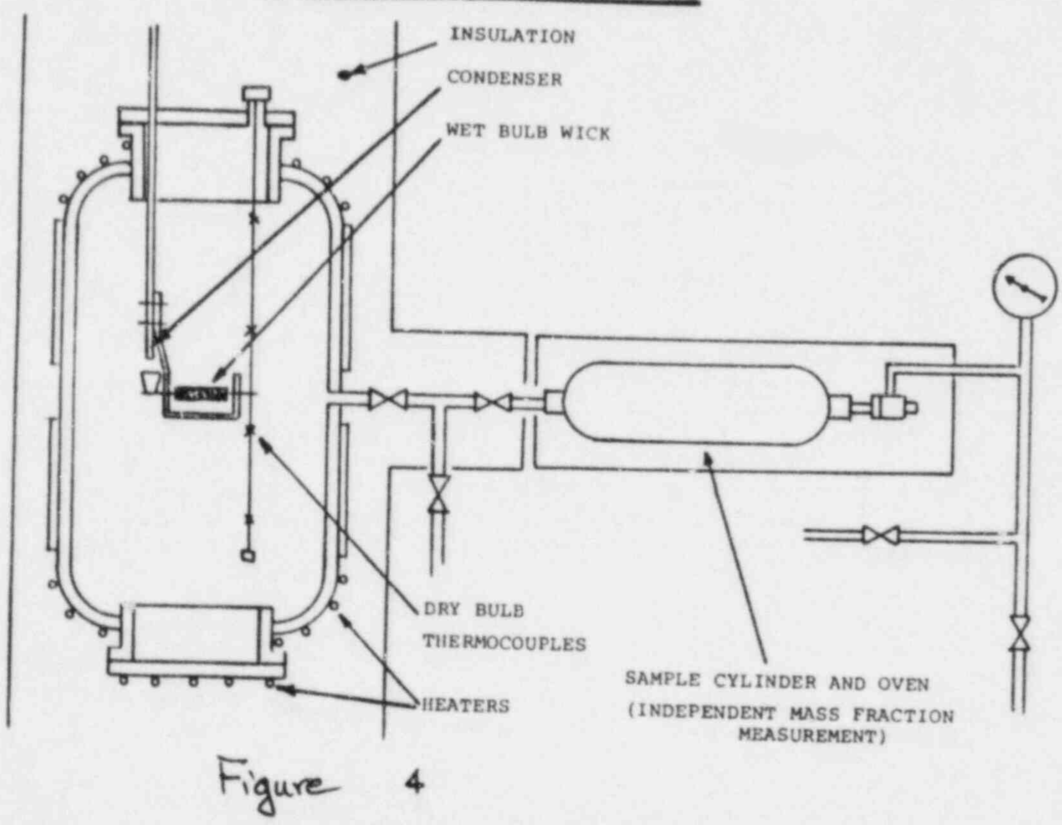


Figure 4

NATURAL CONVECTION APPARATUS

TEST SECTION

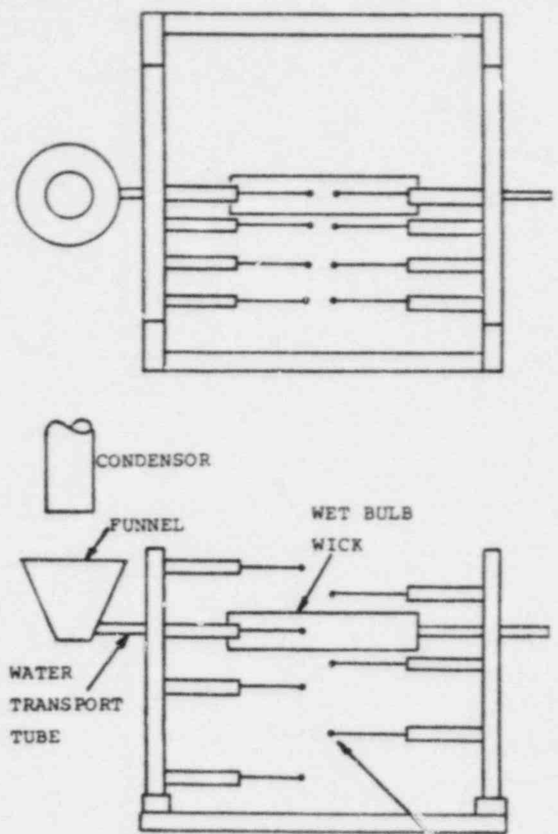


Figure 5

NATURAL CONVECTION TEST RESULTS

CALCULATED VS MEASURED

VAPOR MASS FRACTIONS

WICK PARTIALLY DRY

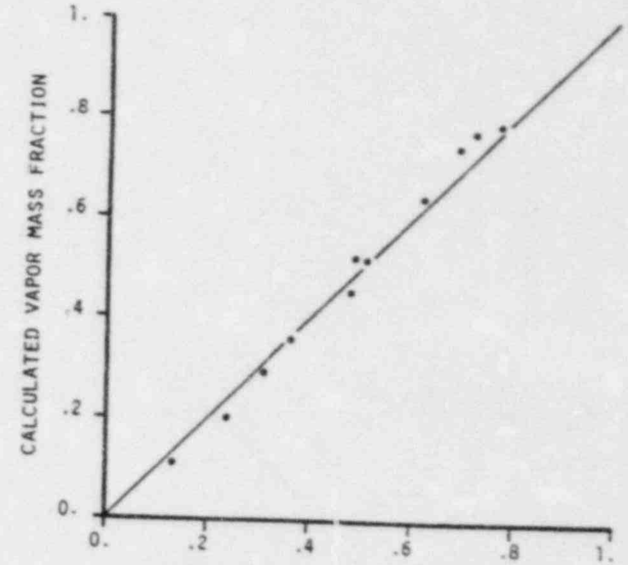


Figure 6

A QUANTITATIVE ASSESSMENT OF THE EFFECT OF CORROSION PRODUCT BUILDUP ON OCCUPATIONAL EXPOSURE

J. R. Divine
Pacific Northwest Laboratory

The program described was developed to provide a method for predicting occupational exposures caused by the deposition of radioactive corrosion products outside the core of the primary system of an operating power reactor. This predictive capability will be useful in forecasting total occupational doses during maintenance, inspection, decontamination, waste treatment, and disposal. In developing a reliable predictive model, a better understanding of the parameters important to corrosion product film formation, corrosion product transport, and corrosion product film removal will be developed. This understanding can lead to new concepts in reactor design to minimize the buildup and transport of radioactive corrosion products or to improve methods of operation. To achieve this goal, three objectives have been established. These objectives are to provide:

1) Criteria for acceptable coolant sampling procedures and sampling equipment that will provide data which will be used in the model development;

2) A quantitative assessment of the effect of corrosion product deposits on occupational exposure; and

3) A model which describes the influence of flow, temperature, coolant chemistry, construction materials, radiation, and other operating parameters on the transport and buildup of corrosion products (crud).

BACKGROUND

Corrosion and corrosion product buildup were a major concern in boilers and other power plants long before the advent of nuclear reactors. As a consequence, periodic chemical cleaning was needed. When commercial nuclear power plants were designed with their requirement for high purity water, many people were of the opinion that cleaning no longer would be needed because of the reduced corrosion with pure water. Indeed typical, uniform corrosion rates are low, too low to be of any concern structurally. Nevertheless, corrosion and corrosion product transport have proven to be a long-term problem because even low corrosion rates when combined with large surface areas can lead to measurable amounts of mobile corrosion products. A major effect of corrosion product mobility in modern nuclear plants is the generation of activated corrosion products which deposit in out-of-core piping and lead to radiation fields. It can be readily calculated that if all the corrosion products stayed where they were formed or if all corrosion products that dissolved stayed in solution, out-of-core radiation levels during shut-down would be minimal.

As a result of corrosion, corrosion product transport, and activation, radiation fields and occupational exposure begin to grow with operation of the reactor. The increase of the radiation fields, the increasing need for maintenance and inspections, and the increasing concern with occupational exposure together with the increasing awareness that someday the reactor plant will need to be decommissioned have led to a need to understand and to predict the magnitude of radioactivity buildup. This basic need has been the driving force for research at least as early as the 1950s in the United States. With state-of-the-art methods for sampling and analysis, with past efforts as a base, and with improved methods for correlating data, the current program is expected to be better equipped to follow and interpret the observed behavior.

PROCEDURE

Three tasks were defined to reach the stated objectives:

Task 1: Reactor Coolant Sampling System and Procedure Development

Task 1 is to review and evaluate current sampling systems and methods and to construct and demonstrate a sampling system which meets or exceeds criteria defined by the task. Sampling methods used in other NRC programs as well as U.S. Department of Energy research programs including the DOE Geothermal sampling program are being specifically reviewed.

The basic criteria will, in part, consider the following observations: Most sampling systems do not sample or analyze at temperatures; the general practice is to cool the sample before collection. Furthermore, because temperature affects solubility and particle size, a reliable sampling system must be able to do part of the analysis, or take controlled samples, at the reactor operating temperature. Cooling the samples prior to the measurement of number and size of particles, for example, requires kinetic data on nucleation and particle growth which are not currently available.

Task 2: Effect of Activated Corrosion Product Buildup on Occupational Exposure

Task 2 is to catalog the areas at power reactors where occupational doses are received during routine and maintenance operations and during inspections. Data on pertinent field or, preferably, contact radiation measurements will be collected. These data, together with those which are developed in Task 1, will be used to provide a relationship between the locations where occupational doses occur and the sources of radiation fields due to corrosion product buildup.

Task 3: Corrosion Product Transport and Deposition Model Development

There have been a number of programs involved in modeling corrosion product transport. Little significant published progress is apparent at this time. In this program, the existing models are being examined in detail and improvements are being made based on currently available data, improved calculational methods, existing thermodynamic data bases, and increased knowledge of deposition and transport mechanisms.

The degree of sophistication of the model being developed is based on the expected availability of future data and its expected use--detailed calculations of corrosion product distribution and composition. Past models are improved by incorporating recent information on the effects of parameters such as temperature, flow rate, and the effect of water chemistry on solubility, deposition rates, release rates of corrosion products and on corrosion rates. For a complete model, which will handle the entire cooling circuit, later inclusion of the effects of temperature, temperature gradients, heat flux, and radiation are necessary.

CURRENT STATUS

In Task 1 the emphasis has been an initial sampler design. Major input to this has been based on past experience, mostly unpublished work. The basis for the design has been the following set of criteria:

1. The sample shall be collected, and liquid/particle separation performed, at reactor temperature and pressure. The sample system should retain any dissolved gases that may affect pH and/or solubility.
2. The sampler shall be designed to minimize introduction of foreign materials or gases into the sample.
3. The sampling lines shall be as inert as possible to the solution being sampled, and shall not introduce corrosion products that might be mistaken for components of the primary coolant. Sample lines should be no longer than necessary.
4. Sampling shall be isokinetic to give representative particle size distribution in the sample.

5. The effect of sample handling on later analysis shall be minimized and quantified.

6. The sampler shall be designed to minimize risk to the operator from the high temperatures, pressures and radionuclides involved. Safety standards shall be adhered to strictly.

7. The sampler shall be constructed to be an acceptable primary system pressure boundary.

In general the choice for inert systems has been titanium. At least two reasons are given for this--it is relatively available and it does not contain zirconium so that fuel corrosion can be monitored.

Zircaloy, our preference for a construction material, is rarely used, mainly because of the cost. Our opinion is that the use of Zircaloy components versus titanium is justified because the Zircaloy is resistant to a wider range of decontamination solutions should it become necessary to clean the system and because of the hard, smooth surface that can be formed on the components. Further, the corrosion rate of the Zircaloy components in the sampler is too low to be of concern and is more than compensated for by the other features.

At the present time, preliminary filter tests are being conducted using a recirculating autoclave system loaded with a source of corrosion products. The filters are of duplex construction with a silver or gold filter downstream of a PTFE filter. The deposits found on the metal filter are larger than the pore size of the PTFE indicating formation and growth from ions or colloids. There is a definite time dependence on the crystalline type and size, but details have not been developed.

The Task 2 work on occupational exposure is being coordinated both with the other tasks of this program and with the related NRC program, "Decontamination Effectiveness and Its Impact on Occupational Exposure." Because the other tasks of this program have not progressed to a point of being able to provide data to Task 2, little effort has been expended on it this year.

The Task 3 effort has been to review existing corrosion product transport models, and to evaluate features for possible inclusion into our own state-of-the-art model. Although there are a large number of models and a great deal of effort has been expended doing modeling, no great success has been achieved.

We are developing a model using applicable concepts of past models and ideas developed from our background in the areas of corrosion product transport, activation of impurities in reactor coolants, and decontamination.

The initial model may be summarized by the following:

1. Iron and cobalt are introduced into the coolant via corrosion, erosion and feedwater impurities. The cobalt of major importance is ionic or colloidal although some particulate cobalt is present.
2. The ionic cobalt absorbs onto the iron oxide which adheres to the fuel surface. Other sorption sites are possible but based on the relative abundance of iron oxides most sites will be on these oxides. The only mechanism by which the cobalt remains in the flux long enough to become activated is by this adhesion to or incorporation in the iron oxide.
3. The activated cobalt-60 may either permanently adhere to the fuel or be released from it by dissolution or wear. The exact mechanism will be studied during the program. The released fraction will either be soluble or insoluble, designated as either ionic or crud respectively; the ionic material includes colloids because the transport of both can be readily represented by the concept of diffusion.
4. The corrosion product on coolant piping system walls is modeled as bilayer with an inner tightly adherent spinel underneath a loosely adherent crud layer. The cobalt-60 is considered to deposit via three mechanisms:
   a. The crud layer acts as a sink for ionic cobalt which is picked up; the latter then diffuses into the crud layer. Some of the layer may convert into the spinel structure.

b. The cobalt containing crud in the coolant mechanically deposits on the surface and adheres loosely.

c. Cobalt in the crud layer diffuses into the spinel or is sorbed onto the spinel surface.

The constants required for the equations describing this model are taken from the literature and will be adjusted as new information or experimental results become available. In particular as the model sophistication increases, an increasingly larger number of regions of the coolant system can be mathematically isolated as nodes. Each node will have its own set of constants corresponding to the conditions of that region.

Validation and refinement of the model will be accomplished by comparison with actual in-reactor radiation dose rate measurements and continued evaluation of the assumptions in the model that affect the buildup of radioactivity.

Our effort, though modest, is a multidisciplinary effort involving chemical engineers, chemists, health physicists and computer specialists all of whom have experience or information on transport processes, chemistry, mathematics, and activity distribution. It is anticipated that this group, using information from the several tasks within the program, will provide the novelty needed for success.

GUIDANCE FOR AIR SAMPLING AT LIGHT WATER REACTORS[a]

Bryce Rich
Paul Ritter
EG&G Idaho, Inc.

This report covers evaluations of radiological air sampling for all Nuclear Regulatory Commission licensees (including nuclear power plants). The evaluations were made under a technical assistance contract. Information gathered from a literature review, worksite visits, and equipment tests will provide an insight into the present status of air sampling and the possibility of improving the programs with new technology and improved methods. The product of this work will be a summary of findings, equipment and technique evaluations, and conclusions and recommendations for improving air sampling at nuclear facilities.

The research planned for this project was designed to provide the basis for the recommendations and will include: (a) a review of literature, (b) visits to typical worksites, (c) equipment testing and evaluation, and (d) internal dosimetry evaluation.

There is an extensive, although not exhaustive, collection of literature dealing with air sampling and allied fields. Several articles contained findings of researchers attempting to determine and reduce sources of inaccuracy in air sampling as related to evaluation of worker exposure.

Of greatest applicability to this study were articles dealing with differences between exposure estimates obtained using personal air sampling and general area air sampling. These indicate that lapel mounted personal air samplers (PAS) were the more reliable means of measuring worker exposure. The research indicated that underestimates of exposure by factors

---

a. Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570.

of up to 100 were possible when using general area air sampling. This work is of renewed interest due to proposed use of new dosimetry philosophies and techniques.

Performance tests of air samplers were planned to determine the capabilities and relative merits of both routine and special purpose samplers. Selection of sampling equipment (and methodology) to incorporate in a sampling system should be based on knowledge of the advantages and disadvantages associated with any particular sample. A complete system may be assembled from parts that, by themselves, could not provide adequate performance.

The research plan also included an evaluation of current air sampling programs. Worksite visits were planned for field characterization of airborne material, and to provide first-hand knowledge of working conditions. Conditions at each worksite define the performance requirements for an adequate air sampling system. Design of the system is directed towards meeting these requirements.

Worksite visits are essential for this study because of the substantial differences in the airborne contamination and working conditions at each facility.

To date the worksite research has focused on the uranium milling industry. The information gathered during uranium mill site trips is presented as an indication of some of the current procedures employed at NRC facilities. The mill air sampling programs were not standardized to any extent. Most mills used general area monitoring methods, although a number of different types of pumps were used to perform the sampling, including stationary PASs.

In almost all cases, total airborne uranium concentration was used for calculation of MPC•hs, as an indication that internal exposure limits had not been exceeded. These calculations are not exact estimates of internal exposure; a number of factors determining actual internal exposure are overlooked. Improvements in sampling techniques may help some uncertainty, especially with regard to measuring respirable fraction. Our measurements and previous research indicate low respirable fractions. Size selective sampling may give better information for determining worker exposure. Results of solubility and chemical analysis are not yet available, but will provide further information regarding worker exposure.

A less extensive series of reactor site visits is planned; reactor safety programs are familiar to members of our group because of continued association with a variety of EG&G Idaho support programs. The Monticello power reactor has been visited as a part of this project.

The primary role of air sampling at power reactors is to provide a means for planning avoidance and control of exposure.

Worker internal exposure is usually determined with whole body counting and/or excreta analyses. Some reactor safety operations also use air sampl''g information for estimating exposure in MPC•hs as an indication of degree of license compliance.

A detailed evaluation of commercially available PASs was completed specifically to determine if they would be acceptable for extensive, perhaps fulltime use as monitors of workers inhalation exposure. PAS monitoring is inherently superior to general area monitoring for this purpose; however, PASs have not been used as extensively in the NRC-licensed industries as general area air sampling systems. To some extent, the resistance to using PASs is due to the imposition placed on the workers and their subsequent tendency to avoid wearing the samplers. Noise level, weight, and size are key factors for worker acceptance and have been measured for this evaluation. We found a wide range of performance among the pumps tested. The best pumps would be suitable for use in an extensive PAS-based monitoring program.

This project also includes plans for controlled testing of equipment in simulated working situations. An experiment was conducted in collaboration with the Inhalation Toxicology Research Institute to determine the properties of aerosols generated during pipe cutting, simulating a decontamination and decommissioning operation. The aerosol was generated in typical work conditions, and several types of samplers were operated in the cutting room.

A particularly notable finding was the factor of 3 difference in measured concentration between the right and left lapel-mounted PAS filters. This may be due to the large concentration gradients near the work piece, resulting from the violent cutting technique used. More testing is planned, using controlled aerosol generation. Performance tests of sampling system collection efficiency as a function of particle size are an important concern, especially if size selection devices are used.

Air sampling systems can be designed to meet two major objectives:

- Monitor the containment of contamination in the worksite. This is a check of the function of physical contamination barriers as well as the effectiveness of job planning and standard procedures in reducing the dispersal of contamination. The need for respiratory protection is indicated by this method as well.

- Provide information for worker exposure estimates. Air sampling is used to help estimate exposure or potential exposure of workers. This information may be used for a number of purposes, including exposure assignment for regulatory compliance or exposure assessment in emergency situations for determining potential health effects.

General area monitoring in an adequate number of locations is an effective means of checking for containment. Sensitivity is usually high and unusual measurements can be associated with a particular area. PAS monitoring is virtually required for measurement of individual intake, because it draws the most representative sample of air breathed by the worker. The relative importance of these two monitoring methods depends on the dose assignment scheme of the internal dosimetry system.

Design of an air sampling system depends on the choice of internal dosimetry method. This will dictate whether the air sampling system should monitor the individual or the area. Worksite conditions are taken into consideration to optimize the program (area or individual oriented) for the existing conditions.

Current regulations emphasize use of in vivo and excreta analyses for estimation of internal exposure. Air sampling is used primarily for control of exposure (using MPC•hs as a guideline) and as supplementary information in support of exposure estimates. Other systems (especially International Commission on Radiological Protection 26 and 30) set exposure limits based on calculated intake. Air sampling, particularly PAS, has a primary role in monitoring programs designed for these systems. Bioassay is to be used if additional information is needed.

Guidance for Air Sampling at Light Water Reactors

B.L. Rich

EG&G Idaho, Inc.

Contract Objectives

I. Survey current air sampling equipment techniques, and worksite conditions
- Literature search
- Site visits

II. Test available sampling/monitoring equipment
- Personal air samplers (PAS)
- General air samplers
- Special purpose air samplers

III. Evaluate effectiveness of current sampling programs and recommend preferred techniques

82 10 317

Research Plan

I. Conduct literature review
- State-of-the-art in sampling methods and technology
- Worksite exposure evaluation/aerosol science applications

II. Survey worksite conditions
- Aerosol characterization
- Worksite physical conditions
- Industrial processes and work routines

82 10 318

Research Plan (continued)

III. Equipment testing
- Durability, reliability
- Human factors (PAS)
- Sampling efficiency, representativeness

IV. Dosimetry techniques evaluation
- Organ dose vs. calculated intake
- Need for bioassay and WBC

82 10 327

Literature Search Highlights

- Breathing zone vs. general area monitoring can differ by a factor up to 100
- PAS best approximation of breathing zone sampling
- PAS proven an effective detection device in field use

52 10 320

Mills/Sampling Equipment and Techniques

- Concentration for MPC·hrs calculation determined with spot sampling
- PAS, high volume, low volume samplers used
- Programs depend on license requirements and were not standardized

52 10 330

Mills/Internal Dose Assessment

- Airborne activity and working time monitored. Worker MPC·hrs recorded for compliance
- Calculation of critical organ dose not an objective
- Urinalysis measured against administrative limit
- Whole body counting

52 10 329

Results of Uranium Mill Site Visits

Physical properties of airborne contamination:

- Small respirable fraction (~15%)
- Liquid aerosols in solvent extraction area
- Chemical studies and solubility–pending

52 10 328

Equipment Evaluation

PAS evaluation

- Human factors–worker acceptance
- Useful operating features–simplifies use
- Mechanical performance

Conclusion–some models of PAS have excellent performance
Technology is still developing
General and special purpose air sampler evaluation–pending

52 10 325

INEL/ITRI Experiment

Measurements of aerosols produced by pipe cutting

- Large aerosol concentration gradients
- Factor of 3 difference between right and left PAS on worker

52 10 321

Controlled Aerosol Generation and Testing

- Diffusion, dispersal experiments
- Particle size effects and sampling efficiency

52 10 326

Methods for Estimating Exposure

- Critical organ limits
  - Exposure estimates based on whole body counting or bioassay
  - Air sampling provides supplementary information
- Calculated total intake limits
  - Intake estimates based on air monitoring
  - Bioassay provides separate limit or supplemental information
  - Total intake system requires more accurate individual exposure estimates

52 10 324

Choice of Samplers and Techniques Based On:

- Sensitivity requirements
- Personal vs. area monitoring
- Preventive vs. diagnostic monitoring for exposure (CAMs vs. PAS)

82 10 219

Summary of the Personal Air Sampling Pump Evaluation

	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1. [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2. [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
3. [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
4. [illegible]			[illegible]				[illegible]	[illegible]		
5. [illegible]			[illegible]			[illegible]				
6. [illegible]										
7. [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
8. [illegible]								[illegible]	[illegible]	
9. [illegible]		[illegible]								
10. [illegible]	[illegible]	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]		[illegible]
11. [illegible]	[illegible]	[illegible]					[illegible]			[illegible]
12. [illegible]				[illegible]		[illegible]		[illegible]		

[illegible]

Sampling System Design Philosophy

Standardize internal dosimetry--two choices

- Critical organ dose
- Calculated total intake

82 10 322

Conclusions

- Essential to decide on internal dosimetry method
- Dosimetry method determines need for personal vs. general air sampling
- Worksite characteristics also dictate methods

82 10 331

Measurement of Neutron Dose And Spectra at Light Water Reactors

K. L. Soldat, G. W. R. Endres, F. M. Cummings
Pacific Northwest Laboratory

Over the last 4 years dose equivalent and neutron energy spectrum measurements were made at operating commercial power reactors under a Nuclear Regulatory Commission (NRC) program to evaluate personnel dosimetry at the plants. Six reactors, 5 PWRs and 1 BWR, were visited in order to characterize neutron energy spectra and dose equivalent rates. At the PWRs, measurements were conducted inside containment, while at the BWR, measurements were conducted at pipe penetrations outside containment. All the reactors were at full power during the measurements. Two of the PWRs and the BWR were revisited to make additional measurements. The data presented in this report were collected from these three sites.

Additional measurements were conducted at the PNL accelerator facility using monoenergetic neutrons with energies of 70, 97, 110, 161, 264, 358, and 448 keV, as well as energies between 4 and 5 MeV. Measurements were also performed at the NBS research reactor using monoenergetic neutrons from the filtered beams with the following energies: thermal, 2, 24 and 144 keV.

INSTRUMENTS AND DOSIMETERS

The instruments used to measure dose equivalent and/or neutron energy spectra included the multisphere system, tissue equivalent proportional counter (TEPC), helium-3 spectrometer and two types of portable remmeters.

The multisphere system used in this study incorporated a lithium-6 iodide (europium-doped) scintillation crystal, a cadmium sleeve and 5 polyethylene spheres having radii of 3", 5", 8", 10" and 12". The Sanna response functions were used in the unfolding code, LOUHI 78, to infer the neutron energy spectra over 26 energy groups. Group 26 (the highest energy group) was "tied" in such a way as to normalize the dose equivalent to a californium-252 source.

The TEPC is a self-calibrating proportional counter which directly measures absorbed dose in tissue equivalent gas. Its response was confirmed using a californium-252 source, both bare and D_2O-moderated. In order to discriminate the neutron events from gamma events in the counter, all events below 15 keV/μ were discarded.

The helium-3 spectrometer is a proportional counter that detects neutrons through the (n, p) reaction in the helium-3 gas. The signal is proportional to the energy of the proton and by inference, the energy of the neutron. Unfortunately, the width of the thermal peak is such that the lowest neutron energy above thermal which can be detected under field conditions is 30 keV.

Two types of remmeters were evaluated during this study, the cylindrical SNOOPY and the 9" spherical rem-ball. Both had been calibrated using a plutonium beryllium source.

The types of personnel neutron dosimeters which were used in this study included 1) NTA-film, 2) several types of TLD-albedo dosimeters, 3) a polycarbonate track-etch dosimeter with and without boron loaded (n, α) radiators which enhance track formation and 4) a combination TLD/CR-39/polycarbonate dosimeter. The dosimeters were all irradiated on a water phantom, five dosimeters from each type being irradiated together in order to characterize the precision of dosimeter measurements.

ENERGY THRESHOLD DETERMINATION

The dosimeters were evaluated for their energy threshold using monoenergetic neutron irradiations at PNL's accelerator facility and NBS' research reactor. Irradiations were also made using a bare and D_2O moderated Cf-252 source. Table 1 summarizes the results of this study.

Table 1. Energy Threshold Evaluation for Neutron Dosimeters

Dosimeter	Energy Threshold
TLD albedo	None
NTA film	1200 keV
Polycarbonate - without (n, α) radiators	5000 keV
- with (n, α) radiators	None
CR-39 track etch	100 keV

Several types of TLD albedo dosimeters were used in this study. Each type incorporated slightly different phosphors and dose evaluation techniques. However, it appeared that no matter what phosphor was used or what analysis technique employed, the dosimeters exhibited almost identical energy responses. That is, each dosimeter had a response that was proportional to the inverse neutron energy.

A single NTA film neutron dosimeter was used during the reactor exposures but was not part of the energy threshold evaluation study on this particular project. NBS has reported (Schwartz, et al., 1982) an energy threshold for NTA film inside reactor containment to be 1.2 MeV.

Two polycarbonate track etch dosimeters were evaluated 1) a polycarbonate film which did not use (n, α) radiators and was part of a combination dosimeter and 2) a polycarbonate dosimeter which did use (n, α) radiators.

A threshold of 1.5 MeV has been reported (Griffith 1980) for a polycarbonate track etch dosimeter used without radiators, but the dosimeter evaluated in this study failed to respond even at the 4-5 MeV irradiations.

There were polycarbonate dosimeters using two types of boron loaded radiators, those using boron enriched in ^{10}B and those using natural boron. The enriched radiators caused the dosimeter to saturate at every location. Two of the dosimeters using natural boron radiators (out of forty) saturated at low doses. The others exhibited a flat energy response, but also high degree of variability.

The polycarbonate dosimeter which used (n, α) radiators did not exhibit any threshold and was sensitive to all neutron energies included in this study.

A single type of CR-39 film dosimeter was evaluated in this study. The CR-39 was part of a TLD-albedo/CR-39/polycarbonate combination dosimeter. The data from accelerator irradiations suggests a threshold of around 100 keV. Additionally, CR-39 suffers from problems in manufacture which produces different response characteristics in CR-39 plastics manufactured in different batches.

REACTOR CONTAINMENT MEASUREMENTS

The nuclear power plants at which irradiations were performed are designated as Site E (BWR), Site G (PWR), and Site I (PWR). All irradiations were performed in locations where routine entry is made and while the reactors were at 100 percent power. The dosimeters were irradiated inside two units at the BWR plant and both locations were at sample line pipe penetrations through the biological shield. Since dose equivalent rates were on the order of a millirem per hour (mrem/hr) at both locations, long irradiation times were required. Dosimeters were irradiated at four locations inside containment of each of the PWR plants. The neutron fields at these locations in each of the three reactors were characterized during earlier measurements.

Neutron Energy. The neutron energies inside reactor containment were determined using the multisphere system, the He-3 counter and the neutron dosimeters. A summary of the data is shown in Table 2. The multisphere data indicated relatively low average neutron energies inside containment with an apparent dependency on the amount of moderation present.

The helium-3 spectrometer was used only at Site I and exhibited sharp cut-off on the neutron energy spectrum at 300 keV. Instead of arriving at some average, then, it is just noted that the average was below 200 keV.

The energies based on NTA film, CR-39 and the polycarbonate dosimeter radiators are inferred from the threshold information and the fact that they failed to respond inside containment.

The energies derived from TLD data were determined by first calculating response functions from the accelerator irradiations and then using the response at the reactor irradiations to solve for energy. This process was not done for the polycarbonate dosimeters with (n, α) radiators although they did respond inside containment.

Neutron Dose Equivalent Responses. Previous studies (Endres et al. 1981) have shown the TEPC to be the best reference instrument for determining neutron dose equivalent inside reactor containment. Generally the TEPC and multisphere systems agreed closely on the neutron dose equivalent rates inside

Table 2. Average Neutron Energy (keV) Inside Reactor Containment

	M/S	^3He	NTA	CR-39	POLY (without radiators)	TLD-Albedo
SITE E (BWR)	155	---	<1200	<100	<5000	17-140
SITE G (PWR)	50-65	---	<1200	<100	<5000	8-45
SITE I (PWR)	30-60	<200	<1200	<100	<5000	5-31

containment. The SNOOPY and 9" remmeter responded high as shown in Table 3 which gives the range of dose equivalent responses for the instruments and dosimeters using the TEPC as the reference measurement.

As mentioned previously, the NTA film, CR-39, and poly-carbonate without (n, α) radiators showed no response. The TLD-albedo dosimeters responded high for the most part, depending on the calibration source, the moderation present at the measurement location and the technique used to correct the response (if any). Although the polycarbonate dosimeters with (n, α) radiators did show a response inside containment, the variability was high and it saturated at several locations which indicates the need for caution in the use and analysis of the polycarbonate film.

Table 3. Dose Equivalent Responses Using the TEPC as Reference

INSTRUMENTS

- Multisphere = 1
- SNOOPY = 1.3 - 5.4
- 9" Rem-ball = 1.5 - 5.3

DOSIMETERS

- NTA film = no response
- CR-39 = no response
- Polycarbonate (no radiators) = no response
- Polycarbonate (with radiators) = 0.5 - 5
- TLD-Albedos = 1 - 105

CONCLUSIONS

The foregoing discussion has led to one primary conclusion, namely that dose equivalent in the field needs to be accurately measured and that personnel dosimeters need to be corrected for spectral differences encountered from plant to plant.

In choosing one dosimeter over another, both sensitivity and precision must be addressed. It has been determined that dosimeters employing NTA film lack adequate sensitivity for use inside containment of nuclear plants (Endres et al. 1981; Schwartz et al. 1982). From this study, it is apparent that CR-39 and polycarbonate track etch films used without radiators are also inadequate. The rest of the dosimeters tested displayed adequate sensitivity. The two general types of dosimeters which comprise that group are: 1) TLDs and 2) the poly-carbonate track etch which was used in conjunction with (n, α) radiators.

The precision of the TLD dosimeter depends on the calibration technique and response correction technique. TLD dosimeters calibrated to D_2O-moderated californium-252 or corrected based on 9" to 3" sphere response rates seemed to function best inside reactor containment.

Polycarbonate track etch film with (n, α) radiators was found to be sensitive enough for use inside reactor containment; however, the standard deviation for the results was roughly twice the value of the most precise TLD albedo dosimeters. Also, several of the polycarbonate dosimeters saturated, rendering any evaluation impossible. Future improvements in this dosimeter may help to render this dosimeter practical for general use in reactor containment.

REFERENCES

Brackenbush, et al. 1979. Brackenbush, L. W., G. W. R. Endres and L. G. Faust. "Measuring Neutron Dose and Quality Factors With Tissue Equivalent Proportional Counter," in Proceedings of a Symposium on Advances in Radiation Protection Monitoring, June 26-30, 1978, Stockholm, Sweden.

Cummings, et al. 1982. Cummings, F. M., L. W. Brackenbush and G. W. R. Endres. "Neutron Dosimetry at Commercial Nuclear Sites--Final Report of Subtask B: Dosimeter Response," USNRC Report NUREG/CR-2956, October 1982.

Endres et al. 1981. Endres, G. W. R., J. M. Aldrich, L. W. Brackenbush and R. V. Griffith. "Neutron Dosimetry at Commercial Nuclear Plants--Final Report of Subtask A: Reactor Containment Measurements," USNRC Report NUREG/CR-1769, May 1981.

Griffith et al. 1979. Griffith, R. V., D. E. Hankins, R. B. Gammage, L. Tommasino and R. V. Wheeler. "Recent Developments in Personnel Neutron Dosimeters - A Review," Health Phys. 36, 235 (1979).

Griffith 1980. Griffith, R. V. Hazards Control Progress Report No. 58, April Through September 1979. UCRL-50007-79-2, Lawrence Livermore Laboratory, University of California, Livermore, California.

Ryan 1982. Ryan, R. M. "Evaluation of Personnel Neutron Dosimetry at Operating Nuclear Power Plants," USNRC Report NUREG/CR-2454, June 1982.

Schwartz et al. 1982. Schwartz, R. B., G. W. R. Endres and F. M. Cummings. "Neutron Dosimeter Performance and Associated Calibrations at Nuclear Power Plants," USNRC Report NUREG/CR-2233, April 1982.

JOB SKILLS ANALYSIS OF NUCLEAR POWER PLANT HEALTH PHYSICS TECHNICIANS

Charles B. Meinhold and Frank J. Marotta

Safety and Environmental Protection Division
Brookhaven National Laboratory
Upton, New York 11973

The traditional approach to "training" radiation protection technicians was to hire intelligent young people perhaps with an associate degree in the hard sciences, or nuclear navy veterans. Their training consisted primarily of on-the-job training in the form of direct interactions with their supervisors. When group instruction was given, it normally consisted of classroom instructions with education rather than training as the goal without a clear understanding that there is an important difference between education and training.

Training, as we understand it, is linked to the instruction and practice that are required to develop job related skills or modes of behavior. Education, which may include training, implies achievement of a greater degree of understanding.

Although the traditional approach has produced hundreds of skilled radiation protection technicians, the training tended to be subjective both in scope and depth. The interests and strengths of the supervisor tended to be the determining factors in the program. Such training can result in significant gaps in a trainee's job performance. Today, however, we recognize the need for assuring that all radiation protection technicians are competent to perform their assigned tasks. This takes clear precedence over their "education." Each technician must be able to perform <u>all</u> the tasks included in his specific job classification.

In 1979 the NCRP asked its Scientific Committee 46 on Operational Radiation Safety to develop a report on "Training for the Radiation Worker." At an early stage in the preparation of this report, the Committee recognized the need to adopt a multi-step training process with its roots in job and task analysis. Again, training rather than education was the issue at hand.

This precise approach to training evolved from the military's Instructional Systems Development (ISD) and it has since been adapted by Analysis and Technology, Inc., for power reactor operations for the Institute of Nuclear Power Operations (INPO).

The questions which need to be answered for this project are:

1. Exactly what tasks comprise radiation protection technicians' jobs?
2. Is there a sufficient range of difficulty in these tasks to justify different grades of technicians?
3. Given No. 2 above, what are the criteria to be used in making such a cut?
4. What is the most effective training technique to be used for each specific task?

It was this background, coupled with a desire to work closely with INPO, that led us to adopt the job analysis approach. Indeed, it is only in working closely with INPO's Training and Education Section that the effort funded by NRC for this project can succeed. We have been invited to use INPO's computer programs to interpret job and task analyses. This alone will save many years of effort.

We intend to survey a representative number of radiation protection technicians. The results of the survey will aid in determining exactly what tasks are performed by these technicians. This completed list of tasks can be considered a job analysis.

The next step, the task analysis, involves direct interviews with practicing radiation protection technicians. The interviewer dissects each individual task into all of its simplest components.

The following is a somewhat arbitrary breakdown of the many steps involved in job and task analysis as developed by INPO which we expect to use.

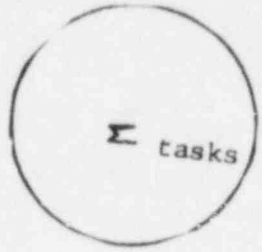
Step	Description
1	Conduct research to identify existing problems, causes, solutions, lack of skills, etc. Identify constraints. Obtain manpower staffing surveys.
2	Create a questionnaire which will permit an objective evaluation of existing information. Review the questionnaire using subject matter experts and job incumbents. Pilot-test the questionnaire. Revise and retest it, if necessary. Review and approve.
3	Identify the target population (HP Technician).
4	Administer the questionnaire to the target population.
5	Compile and analyze data (job analysis).
6	Train interviewers to perform task analysis.
7	Develop performance objectives, conditions, engineering systems. Conduct interview of incumbents.
8	Select team for content validation and editorial review.
9	Conduct in-service workshops: "Train-the-trainer."
10	Develop curriculum (plant specific).
11	Evaluate and revise the training program on a continuing basis.

Although a process such as this may take many staff years to complete, it assures us that: 1) the job is exactly defined and, therefore, 2) the worker can be trained to do exactly what is necessary. A program such as this simply means that workers know and understand their job.

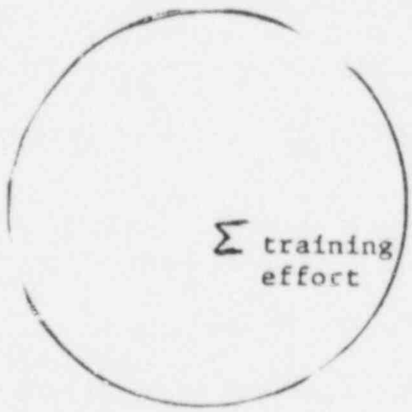
Its major advantage is that the training is tailored to the need and thereby greatly reduces the possibility of unnecessary training or (worse still) poor and inadequate training.

E.L. (Red) Thomas of Duke Power, formerly on loan to INPO, has suggested the following illustration of this idea.

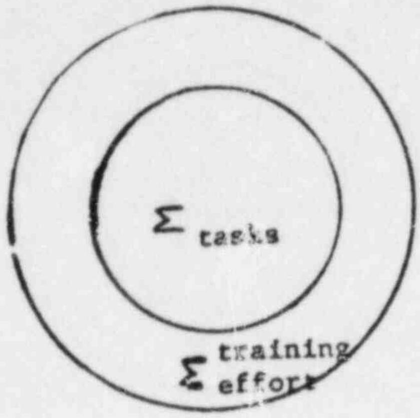
First we depict the total number of tasks which comprise the radiation protection technicians job:



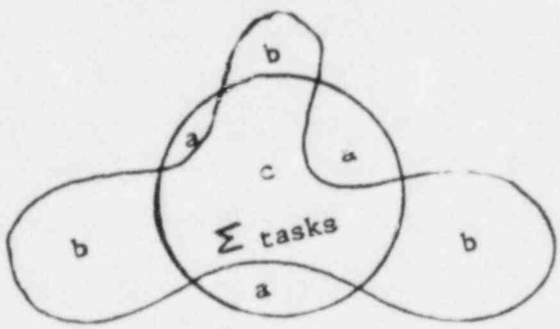
Second we depict the "ideal" training program:



The relationship between the two should be as follows for the trainer to feel comfortable:



The unfortunate reality is perhaps best depicted as follows:



"A" areas are those required tasks for which there has been insufficient training.

"B" areas are those training experiences which do not contribute to the performance of the tasks.

Area "C" is the situation in which the training and tasks are coherent. Utopia, of course, is realized when the Σ tasks circle and the Σ training effort circle are congruent.

Such progressive approachs to technicain training are essential if we are to maximize the productivity of the radiation protection staff while insuring that we keep radiation exposure as low as reasonably achieveable.

PNL-SA-10743

SPECTRA AND DOSIMETRY FOR HIGH AND LOW ENERGY PHOTONS AT LIGHT WATER REACTORS*

P. L. Roberson, G. W. R. Endres, R. A. Fox
D. L. Haggard, K. L. Holbrook and L. A. Rathbun

Radiological Sciences Department
Pacific Northwest Laboratory
Richland, Washington 99352

Field measurements are being performed at commercial nuclear power plants to aid an evaluation of the impact of revisions of 10 CFR 20 on licensees and to aid an evaluation of the adequacy of regulations for monitoring dose due to high-energy photons (>3 MeV). Proposed modifications to 10 CFR 20 incorporate improved exposure-to-dose-equivalent conversion factors (C_x factors) for low-energy photons. These factors differ from the currently-specified factor of unity by as much as 50%. The improved C_x factors are shown in Table 1. They were developed for the four-element, 30-cm diameter ICRU sphere by Dimbylow and Francis (1979).

The use of improved C_x factors may impact methods used to report dose received by radiation workers. Methods include work area surveys and personnel dosimetry services. Air ionization chambers (e.g. area survey meters) typically underestimate dose received in fields consisting of low-energy photons. Personnel dosimeters designed to respond like tissue and worn on the trunk of the body would respond accurately to low-energy photons. Dosimeters designed for an energy response similar to exposure in air or those with inaccurate energy response functions may require redesign or modifying factors developed through field surveys which determine radiation spectra at work areas.

The production of low-energy photons is not limited to those due directly to decay of radioactive atoms. Attenuation of the primary photon field by the use of shielding material generates a contribution at lower energies primarily through Compton interactions (Fenyves 1969, pp 89-90). The maximum flux of scattered photons is expected between 50 keV and 150 keV, just above the sharp rise of the photoelectric cross section for the atoms of the shielding material. The relative flux of primary and scattered photons

* Work performed for the U.S. Nuclear Regulatory Commission under a Related Services Agreement with the U.S. Department of Energy Contract DE-AC06-76RLO 1830.

varies with the amount, type and geometry of the source and shielding material. Low-energy photons in operating plants are also produced by the decay of short-lived nuclei, such as ^{133}Xe (81 keV) and ^{57}Co (122 and 136 keV).

The depth in tissue of the maximum dose due to high-energy photons is greater than the monitoring tissue depths of 0.3 cm and 1.0 cm (Johns 1978, pp. 327-330). Estimated depth-dose curves for parallel incident fields of 1.25 MeV (^{60}Co) and 6 MeV are shown in Figure 1. For 6-MeV photons the dose at the 0.3-cm and 1-cm depths is approximately 40% and 75%, respectively of the maximum dose (which occurs at a depth of about 3 cm). Dose received by internal organs may be signficantly greater than the personnel dose reported. The greatest errors are made using personnel dosimeters designed to monitor at the 0.3-cm depth or those with a compensated design for improved response at the lower photon energies (20-200 keV), (Cummings 1981). Regulations for monitoring dose due to high-energy photons may require modifications to ensure accurate reports of personnel dose received.

High-energy photons are produced in several ways. The decay of ^{16}N results in photons with energies of 6.1 and 7.1 MeV and beta particles with energies of 4 and 10 MeV. Beta particles produce high-energy photons by radiative energy losses in matter (bremsstrahlung). Photons are also produced in neutron absorption reactions (capture gamma-ray processes). Most notable are reactions with silicon to produce 3.3- and 4.9-MeV photons and with iron to produce 5.9- and 7.6-MeV photons (Adyasevich 1956). These production mechanisms are present in operating reactors only.

The nuclear power plant sites are being monitored in the auxiliary areas, in containment and in waste storage areas. Operating and non-operating pressurized-water reactors (PWR) and boiling-water reactors (BWR), and old and new plants are being visited. Exposure rates in measurement locations range from 0.1 mR/h to greater than 10 R/h. Measurements to date have been performed for three operating PWR's, two nonoperating PWR's and one operating BWR.

Spectral, dose and exposure measurements are being performed at the plant sites. Spectral measurements are performed using germanium (Ge) and/or lithium-drifted germanium [Ge(Li)] photon spectrometers. Figure 2 contains a block diagram of the equipment. A pulse-height distribution is collected at the measurement site using a multichannel analyzer. The data is stored on magnetic tape. After the measurement trip, the data is transferred into computer memory for analysis. The data is corrected for photon scattering in the detector crystal and for detector efficiency to yield accurate spectra for both discrete and continuous components. Effective C_x factors are estimated using the spectral information. The approximate maximum rate at which uncollimated photon spectrometers can be used is 10 mR/h. This maximum can be increased by a factor of 5 to 10 by using a collimator assembly.

Direct measurements of effective C_x factors are being performed using an extrapolation chamber made of tissue-equivalent plastic to measure dose and an ionization chamber to measure exposure. The dose at depths of 0.007 g/cm^2, 0.3 g/cm^2 and 1 g/cm^2 in a 30 cm x 30 cm x 15 cm block of tissue-equivalent plastic (NUREG/CR-1057, pp. 16-19) is determined for the distributed sources at the measurement site. Measurements at all three depths are not performed for each measurement location. A compromise is made between number of measurements at one location and number of locations measured. The extrapolation chamber has a free-sliding central plug controlled by a piston which is driven by a stepping motor (See Figure 3). The shaft of the drive mechanism has a micrometer readout used to precisely determine incremental changes in the chamber volume. The stepping motor control, the electrometer and the voltage control can be positioned 7 to 8 meters from the chamber. The extrapolation chamber and the ion chamber are sequentially mounted on the same base support to assure reproducible geometry for the comparison of dose to exposure. Corrections for dose due to beta particles are derived from data taken with thermoluminescent dosimeters (TLD). The extrapolation chamber is also used to measure depth-dose curves in areas containing high-energy photons. The minimum dose rate required for a sufficiently precise measurement is approximately 100 mrad/h.

Thermoluminescent dosimeters (TLD) imbedded in a plastic (methylmethacrylate) phantom are used to obtain depth-dose information. Eight depths are monitored from the surface to 7 cm along the central axis of a 20 cm x 20 cm x 15 cm phantom. Because measurements can be performed at nearly any dose rate, this technique helps correlate the spectral with the extrapolation-chamber measurements. It also provides relative beta/photon information, relative depth-dose information for the monitoring of high-energy photons, and approximate effective C_x factors.

Data taken with photon spectrometers for two PWR's during refueling shutdowns are shown in Figures 4 and 5. Both figures show discrete line spectra of Cesium and/or Cobalt isotopes and a significant continuum with a maximum energy near 120 keV. The continuum was attributed to photon scattering within shielding material. The flux to exposure conversion reduced the effect on the effective C_x factor of the lower-energy continuum by a factor of approximately six. Assuming parallel incidence, the effective C_x factors derived from these distributions are approximately 1.1.

Conversion factors measured directly using the extrapolation chamber and the ionization chamber equipment are given in Table 2. The set of measurements for shutdown PWR's, resulting in factors ranging from 0.82 to 1.05, indicated that medium photon energies dominated. Because of the need to perform such measurements in high dose-rate areas and because such areas in a shutdown plant are typically associated with accumulation of long-lived radionuclides (e.g. Co-60 and Cs-137), the fact that medium photon energies dominated was not surprising. The measured values less than unity were ascribed to distributed sources. The measurement performed in the 10.8 R/h field (C_x of 1.00 $\pm$ 0.05) was for a localized source and is consistent with C_x factors expected for ^{60}Co or ^{137}Cs sources assuming parallel incidence.

Measurements performed in PWR's during shutdown using the TLD-loaded phantom provided depth-dose information consistent with ^{137}Cs and ^{60}Co photon energies, with less than 10% surface dose due to beta particles.

Pulse-height distributions for an operating PWR are shown in Figures 6 and 7. Figure 6 shows discrete spectra from short-lived Xe isotopes in addition to the discrete and continuum spectra observed in the shutdown reactors. The lower-energy components are enhanced, with the extreme example of this effect shown in Figure 7, where the 81-keV photon from ^{133}Xe dominates the spectrum. The C_x factor for 81 keV assuming parallel incidence is large (>1.4), while the C_x factor assuming uniform incidence is not significantly enhanced compared to 1 MeV photons (Dimbylow 1979). Thus, radioactive Xe permeating the air (as in the personnel hatch, Figure 7) results in no enhancement of the C_x factor. The Xe isotopes in the demineralizer room of the auxiliary building (Figure 6) are probably primarily present in the water. They contribute to the enhanced C_x factor of 1.2.

The effects of plant age on the spectral composition is seen by comparing Figure 4 with Figure 6. The data were taken in the demineralizer rooms at a twin-reactor site whose operating ages are 5y (Figure 4) and 1.6y (Figure 6). The contribution of ^{60}Co photons relative to the low-energy continuum is greatly enhanced in the older reactor. As long half-life radioactive deposits ("crud") build up with age, the relative contribution of low-energy photons to dose received by workers declines.

Photon spectrometer distributions demonstrating the presence of high-energy photons in operating plants are shown in Figures 8 and 9. The data in Figure 9 was taken in a PWR overlooking the reactor cavity (2 mR/h). The position expected for the 6.1 MeV photon of ^{16}N is shown. No definite peak was observed; however, a significant contribution from photons near 6 MeV was observed. Since the detector efficiency at 6.1 MeV is 1.5% of the efficiency at 120 keV, the density of photons near 6 MeV is approximately 10% of those near 120 keV. Because of the much greater energy of the 6 MeV photons, they will have a significant effect on dose received.

Figure 9 was taken near a moisture separator tank in the turbine building of an operating BWR (14 mR/h). The presence of ^{16}N is indicated by the full-energy peak at 6.1 MeV and the single- and double-escape peaks at 5.6 and 5.1 MeV, respectively. The escape peaks represent the loss from the Ge crystal of one or two electron-positron annihilation photons. When corrected for efficiency, the spectrum is dominated by the ^{16}N photons. Not all location with large contributions from high-energy photons are dose-monitoring problem areas. In particular, lightly-shielded areas containing ^{16}N produce maximum dose rates at or near the surface because of contributions from the ^{16}N beta particles (4 and 10 MeV). Additional data collection and analysis is required to identify problem areas.

References

Adyasevich, B. P., L. V. Groshev and A. M. Demidov. 1956. "Investigation of Gamma-Rays Emitted by Nuclei of Titanium, Iron and Silicon on Capturing Thermal Neutrons." Soviet Journal of Atomic Energy, 1, 183-192.

Cummings, F. M., R. C. Yoder, G. W. R. Endres and R. L. Kathren. 1981. "Study 2-High Energy Photon Response." Included in Hanford Personnel Dosimetry Supporting Studies FY-1980. PNL-3536, Pacific Northwest Laboratory, Richland, Washington.

Dimbylow, P. J. and T. M. Francis. 1979. "A Calculation of the Photon Depth-Dose Distributions in the ICRU Sphere for a Broad Parallel Beam, a Point Source and an Isotropic Field." National Radiological Protection Board, NRPB-R92, Harwell, England.

Fenyves, E. and O. Haiman. 1969. The Physical Principles of Nuclear Radiation Measurement. New York Academic Press.

Johns, H. E. and J. R. Cunningham. 1978. The Physics of Radiology, 3rd ed. Springfield, Illinois, Charles C. Thomas.

NUREG/CR-1057. 1979. "Confirmation of Conversion Factors Relating Exposure and Dose-Equivalent Index Presented in ANSI N13.11." Pacific Northwest Laboratory, Richland, Washington.

TABLE 1. Conversion Factors for Computing Dose Equivalent from Exposure. The factors are derived from Monte-Carlo calculations for the four-element ICRU sphere (Dimbylow 1979).

PHOTON ENERGY (keV)	CONVERSION FACTOR (rem R^{-1})* TO DOSE EQUIVALENT IN THE ICRU SPHERE AT A DEPTH OF		
	1.0 cm ("DEEP")	0.3 cm	0.007 cm ("SHALLOW")
15	0.28	0.67	0.90
20	0.58	0.79	0.94
30	1.00	1.07	1:11
40	1.28	1.29	1.34
50	1.46	1.46	1.50
60	1.47	1.47	1.52
70	1.45	1.45	1.50
80	1.43	1.43	1.48
90	1.41	1.41	1.45
100	1.39	1.39	1.43
110	1.37	1.37	1.40
120	1.35	1.35	1.36
130	1.33	1.33	1.34
140	1.32	1.32	1.32
150	1.30	1.30	1.30
662	1.03	1.03	1.03

*1 rem = 10^{-2} Sv; 1 R = 2.58×10^{-4} C kg^{-1}

TABLE 2. Extrapolation Chamber and Ion Chamber Measurements

SHUTDOWN PWR

SITE	LOCATION	EXPOSURE RATE, mR/h	C_X FACTOR	
			0.007 cm	1 cm
K	DRAIN VALVE	100	1.05 ± 0.08	0.93 ± 0.15
B	PIPING NEAR STEAM GENERATOR	195	0.89 ± 0.07	0.82 ± 0.04
B	LET DOWN HEAT EXCHANGER ROOM	10,800		1.00 ± 0.05
B	REMOVED NOZZLES	900		0.89 ± 0.05

OPERATING BWR

SITE	LOCATION	EXPOSURE RATE, mR/h	0.007 cm	1 cm
M	CLEAN-UP PHASE SEPARATION TANK	7,350		1.03 ± 0.04

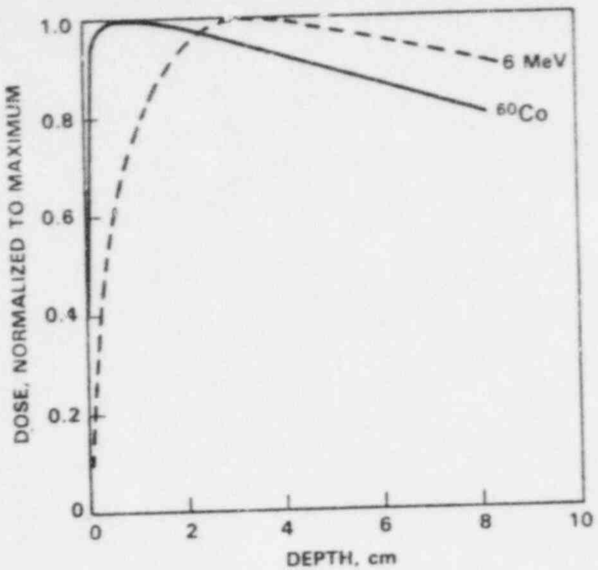


FIGURE 1. Depth-Dose Dependence on Photon Energy. The 6-MeV curve was estimated based on parallel incidence of the photons.

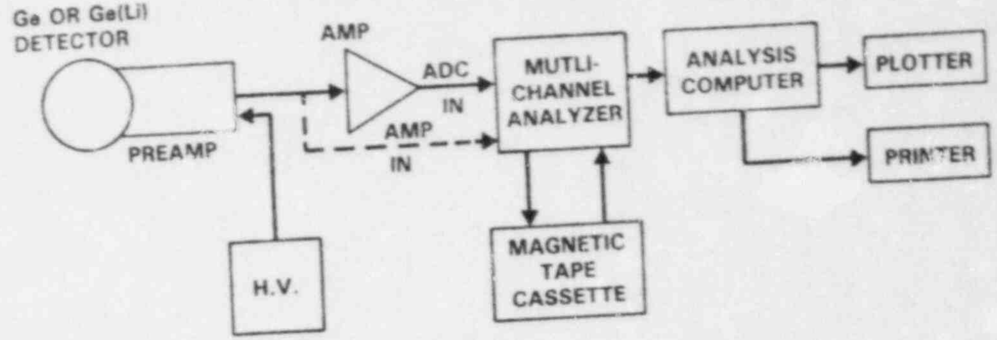


FIGURE 2. Block Diagram for the Photon Spectrometer

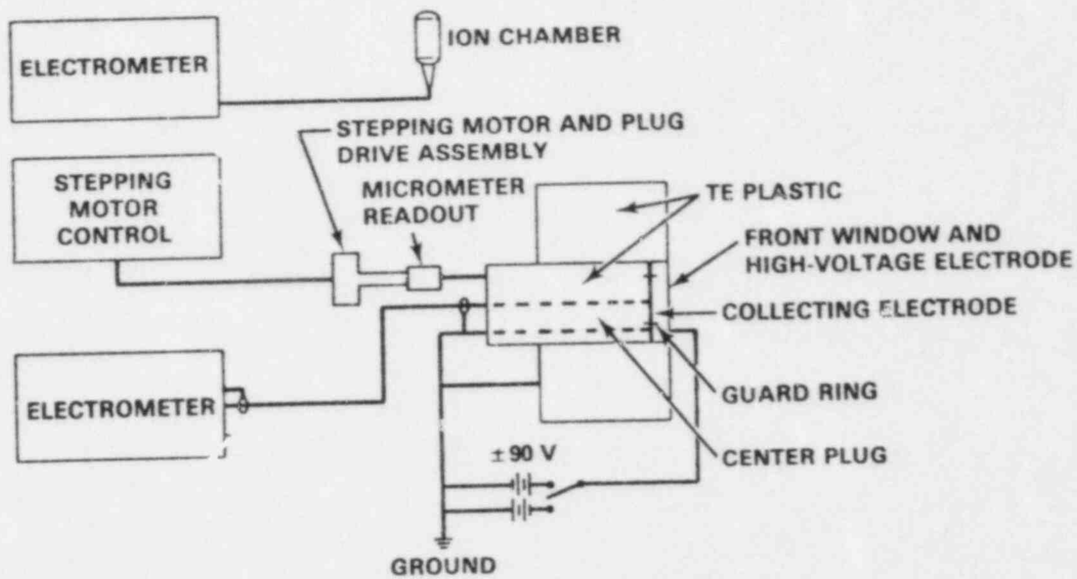


FIGURE 3. Block Diagram of the Tissue-Equivalent (TE) Extrapolation Chamber and Ion Chamber System

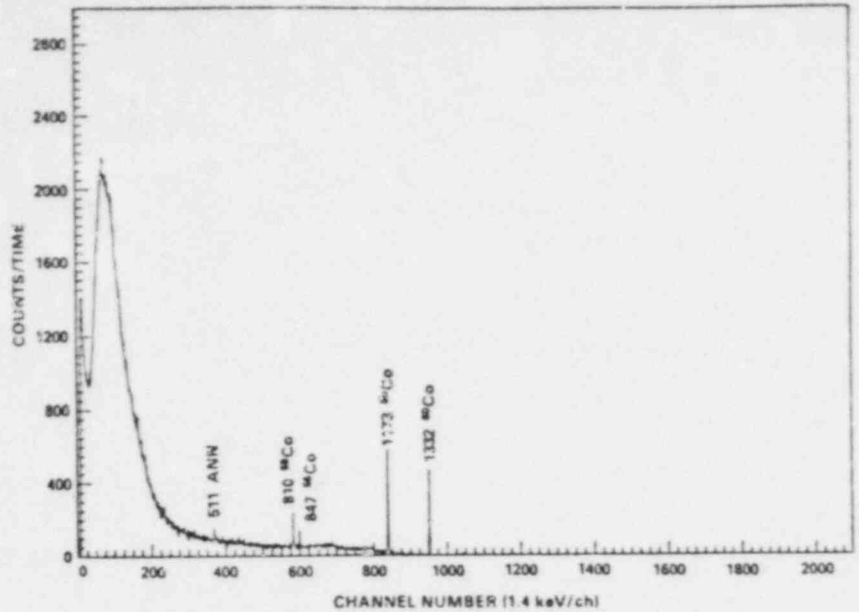


FIGURE 4. Pulse-Height Distribution for a Shutdown PWR (Site K, Unit 1, 5y Commercial Operation). Auxiliary bldg., Demineralizer Room, 0.3 mR/h. The 1 MeV/120 keV relative detection efficiency is 11%.

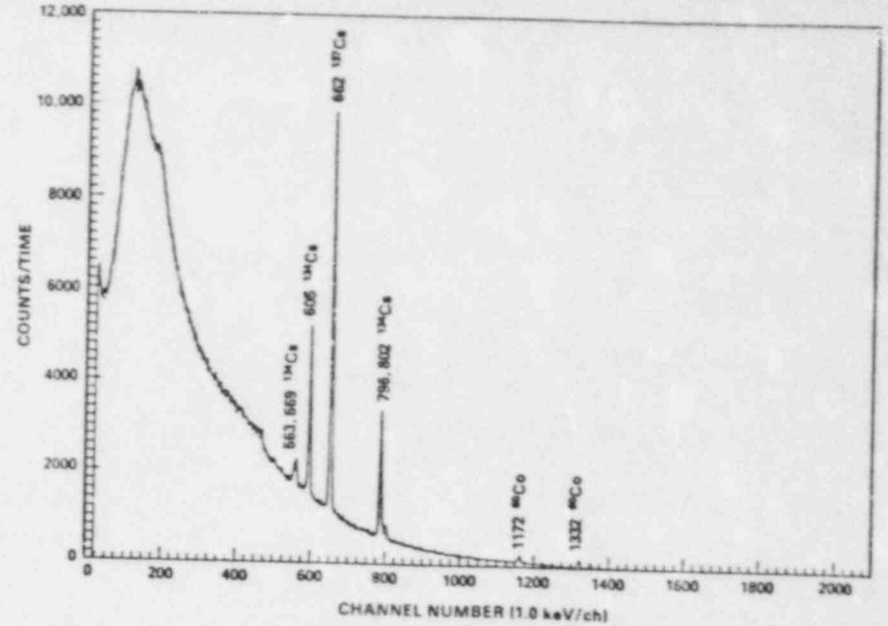


FIGURE 5. Pulse-Height Distribution for a Shutdown PWR (Site B, 7y commercial operation). Near Containment Elevator, 4 mR/h. The 1 MeV/120 keV relative detection efficiency is 9%.

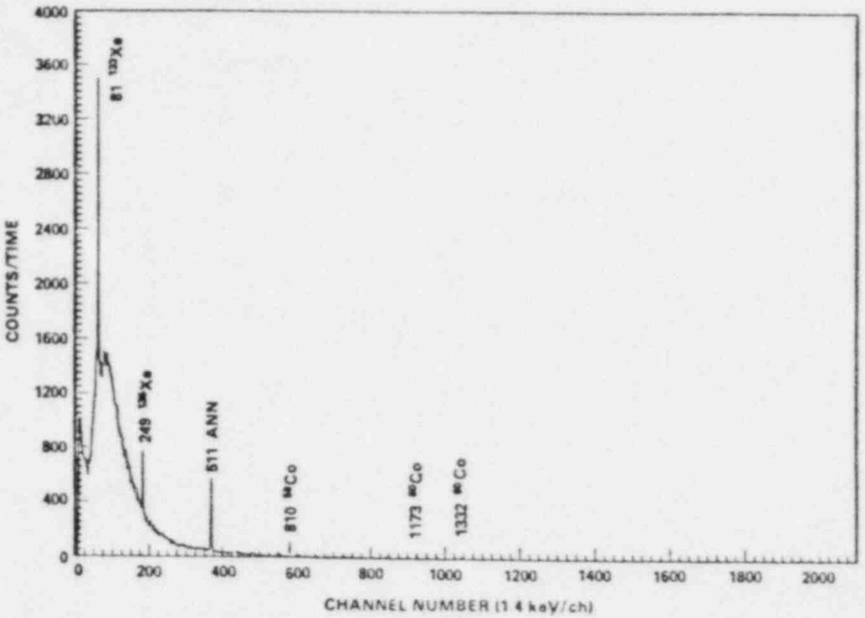


FIGURE 6. Pulse-Height Distribution, Operating PWR (Site K, Unit 2, 1.6y commercial operation). Auxiliary Bldg., Demineralizer Room, 0.2 mR/h. The 1 MeV/120 keV relative detection efficiency is 11%.

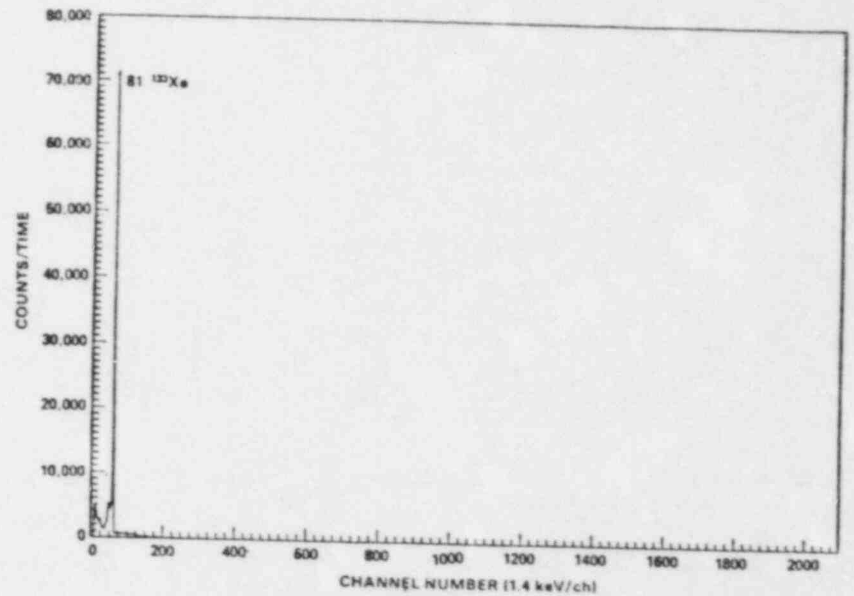


FIGURE 7. Pulse-Height Distribution, Operating PWR (Site K, Unit 2, 1.6y commercial operation). Personnel Hatch, 0.4 mR/h. The 1 MeV/120 keV relative detection efficiency is 11%.

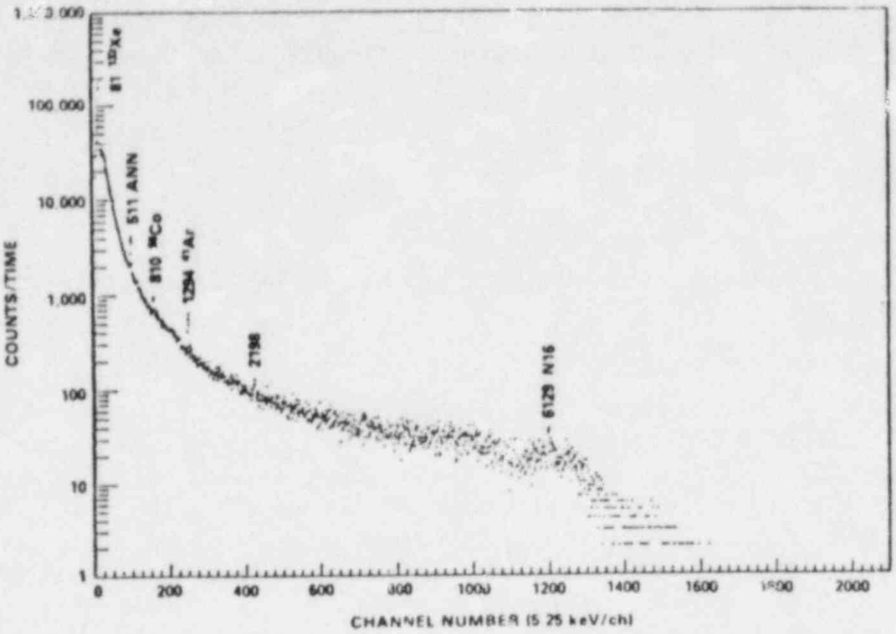


FIGURE 8. Pulse-Height Distribution, Operating PWR (Site K, Unit 2, 1.6y commercial operation). Containment, Overlooking Reactor Cavity, 2 mR/h. The 6 MeV/120 keV relative detection efficiency is 1.5%.

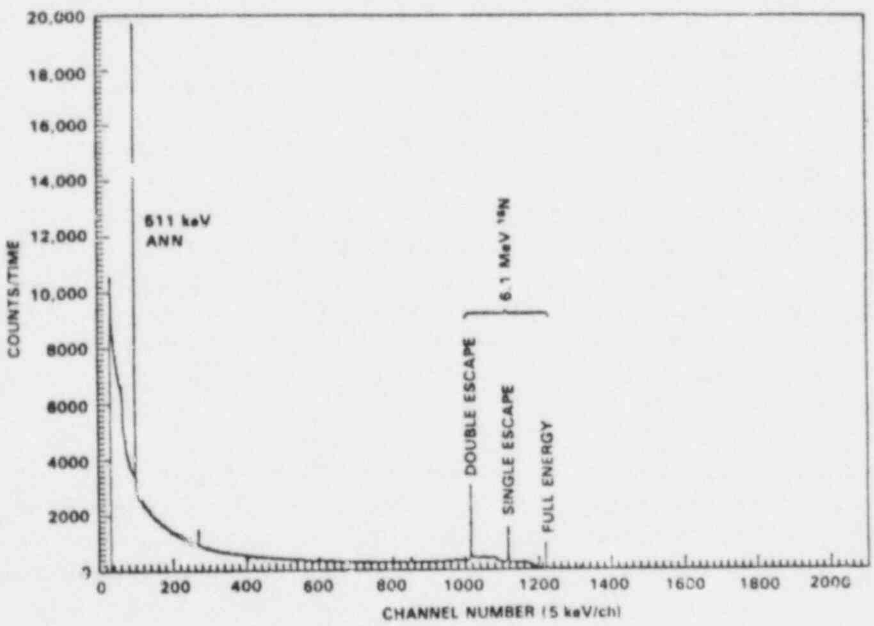


FIGURE 9. Pulse-Height Distribution, Operating BWR (Site M, 8y commercial operation). Turbine Bldg., 14 mR/h. The 6 MeV/120 keV relative detection efficiency is 1%.

NEW DATA AND PERSPECTIVES ON RADIATION RISK COMPARISONS

by

Mary Anne Zanetos, Battelle Columbus Laboratories

In recent years, there has been extensive discussion concerning the adequacy of the maximum permissible dose (MPD) of 5 rem/yr. in affording sufficient protection for radiation workers (ICRP, 1977). Much of the discussion has centered around the issue of whether exposure to the MPD (or alternatively, to average exposures within the nuclear industry) on a regular basis throughout a working lifetime constitutes a risk comparable to that faced by workers in other industries. Literally hundreds of epidemiological studies have examined the morbidity/mortality experience of large groups of workers chronically exposed to hazardous substances. Emerging from these studies are a plethora of quantitative estimates of risk associated with specific exposures or industrial operations. These should, theoretically, provide the basis for a comparative assessment of the risks faced by radiation workers versus their counterparts in other industries.

Unfortunately, both theoretical and methodological issues have complicated such comparative assessments. First and foremost is the characteristic absence of historical, quantitative exposure data for nonradiation workers. Even where such data exist, few companies have maintained sufficiently detailed individual work histories to enable calculation of cumulative exposures. Dose-response trends can often be evaluated qualitatively or semi-quantitatively (e.g. comparing disease rates among workers with different lengths of service) however, the important point is that there is no real equivalent of the "risk per rem" estimates which exist for radiation workers (BEIR, 1980). Second, continual improvements in working conditions have greatly reduced exposures in many industries. Consequently risk estimates based on employees who received their exposures under very different circumstances may be expected to overestimate risks faced by current workers. Third, since data on smoking and other non-occupational risk factors necessary for refining occupational risk estimates are typically absent, any increase above expected disease rates tends to be interpreted (rightly or wrongly) as an effect of exposure.

Other problems concern the noncomparability of the units in which risk is expressed. Risk estimates for ionizing radiation are generally expressed in terms of an increment in cancer deaths per person-rem (absolute risk model), as a percentage increase per rem over the spontaneous cancer rate (relative risk model), or average loss of life expectancy derived from either of these models (BEIR, 1972; BEIR, 1980). On the other hand, risk estimates for nonradiation workers are nearly always expressed either as standardized mortality ratios (SMR's) or proportionate mortality ratios (PMR's). The latter measures express the risk of death relative to some standard population, but do not convey any information on the absolute risk of death or the increase in death rates associated with exposure. There are, of course, underlying relationships between death rates, SMR's, PMR's, and life expectancy but all of the valid methods for converting these into a common metric require access to very detailed mortality data which are normally available only from the original investigator. In view of these problems, it is not surprising that few truly comparative studies have been undertaken. Those which have been published (Cohen and Lee, 1979; Cohen, 1981), while commendable, have utilized suboptimal or outdated mortality studies, improper statistical methodology, or unsupported assumptions concerning the size or composition of the U.S. workforce. The resulting risk estimates are thus somewhat suspect. While the study described below is modest in scale and the results preliminary, we feel it offers important methodological refinements which increase our confidence that the risk estimates derived are indeed valid and fairly representative of recent experience in the industries examined.

Through a consulting agreement with the Department of Biostatistics of the University of Pittsburgh, Battelle's Columbus Laboratories obtained access to mortality data from numerous occupational mortality studies conducted there. The specific industrial cohorts included in the present assessment were selected on the basis of the size of group, length of follow-up, availability of exposure data, representativeness of the industry, absence of any overwhelming mortality hazard, and overall quality of the study from which the mortality data were derived. We also wanted to select relatively common exposures and to represent diverse types of exposures. Pertinent data on the groups selected are shown in Table 1.

Estimates of life expectancy at various ages were computed for the standard population (U.S. white males, 1960) and for each of the four industrial cohorts using the current life table method. Basic life expectancies were first computed based on the total mortality experience of each group. To evaluate the impact of cancer more specifically, the mortality and survival experience of the various groups were then compared with the hypothetical experience of the same populations which would exist if all cancer deaths were eliminated. Chiang (1968) provides computational methods for such competing risk situations. At each age, the difference between estimated life expectancy with all causes of death present and that with cancer deaths statistically eliminated represents the average number of years of life lost in the hypothetical population due to all forms of cancer. The relative effect of employment in each of the four industries on cancer mortality can be estimated by comparing the age-specific loss of life expectancy due to cancer in each cohort with the corresponding value for the standard population. Results for the four industrial cohorts are presented in Table 2; risk estimates for radiation workers derived by other authors are included for comparison.

The data presented in Table 2 suggest several conclusions. First, the overall mortality experience of three of the four industrial cohorts was favorable to that of their counterparts in the general population. This is not unexpected since working populations are always selected to some extent for good health. Although no directly comparable data were available, the all-cause SMR of 80 observed in the Hanford workers (Gilbert and Marks, 1979) suggests this may apply to radiation workers as well. Second, the impact of cancer in terms of loss of life expectancy (LLE) in all four industrial groups is comparable to that projected for radiation workers receiving doses of 0.5 to 5.0 rem/yr. for a working lifetime. LLE due to cancer for three of the four cohorts falls into the range projected for radiation exposures of 5.0 rem/yr. whereas radiation workers receiving doses of 0.5 rem/yr. or less are projected to suffer less loss of life expectancy than any of the four industrial groups examined. Third, while cancer SMR's of the four groups (see Table 1) show no marked increases in terms of death rates, the magnitude of LLE estimates suggests that the cases which do occur in these workers occur at earlier ages than in the general population.

References

ICRP (International Commission on Radiological Protection). 1977. Recommendations of the International Commission on Radiological Protection. ICRP Publ. 26. Permagon Press, New York.

BEIR (Biological Effects of Ionizing Radiation). 1972. BEIR Committee Report. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation. National Academy of Sciences, Washington, D.C.

BEIR (Biological Effects of Ionizing Radiation). 1980. BEIR Committee Report. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation. National Academy of Sciences, Washington, D.C.

Cohen, B. L. and I. S. Lee. 1979. A Catalog of Risks. Health Physics 36:707.

Cohen, B. L. 1981. Perspective on Occupational Mortality Risks. Health Physics 40:703.

Chiang, C. L. 1968. Introduction to Stochastic Processes in Biostatistics. John Wiley and Sons, Publ. New York.

Gotchy, R. L. 1978. Estimation of Life Shortening Resulting from Radiogenic Cancer per rem of Absorbed Dose. Health Physics 35:563.

Bunger, B. M., J. R. Cook, and M. K. Barrick. 1981. Life Table Methodology for Evaluating Radiation Risk: An Application Based on Occupational Exposures. Health Physics 40:439.

Table 1

Summary of Retrospective Cohort Data Used in Life Expectancy Calculation

	Cohort	Definition	Follow-up Period	Occupational Exposures: Primary	Occupational Exposures: Secondary	Major Findings
I.	Workers in a nickel refinery and nickel alloy manufacturing plant (N=1855)	All males employed 1+ years prior to 1948	1948-1977	Nickel	Chromium, iron, copper, grinding dust, solvents, acid mists, carbon monoxide	Overall cancer SMR[c]=98.8 Excess sinonasal cancer in refinery workers only. Moderately elevated SMR's overall for lung, stomach, and prostatic cancer.
II.	Workers in a plastics producing plant (N=2490)	All male workers on hourly payroll employed 1+ years between 1/1/49 and 12/31/66	1950-1976	None	Formaldehyde, vinyl chloride polymers, styrene, acrylonitrile, phenol, butanol, vinyl acetate, ethyl alcohol	Overall cancer SMR[c]=107.4 Excess digestive and genito-urinary cancers.
III.	Workers in fibrous glass manufacturing plants (11 plants, N=14884)	All males employed 1+ years[a] in production-maintenance jobs between 1/1/45[b] and 12/31/63	1946-1977	Glass Fibers	Silica, asphalt, phenol, formaldehyde, ammonia, carbon monoxide, solvents, metal fumes	Overall cancer SMR[c] = 96.4 Moderately elevated SMR's for malignant and non-malignant respiratory disease
IV.	Workers in mineral wool manufacturing plants (6 plants, N=1846)	All males employed 1+ years in production-maintenance jobs between 1/1/45 and 12/31/63	1946-1977	Mineral Wool Fibers	Asbestos, asphalt, phenol, formaldehyde, ammonia, carbon monoxide, solvents, metal fumes	Overall cancer SMR[c]=123.8 Moderately elevated SMR's for malignant and non-malignant respiratory disease

a. For two plants minimum employment reduced to 6+ months
b. For one plant earliest employment date set back to 1940
c. Expected number of deaths based on U.S. white male mortality experience

TABLE 2. ESTIMATES OF LOST LIFE EXPECTANCY AMONG WORKERS EXPOSED TO CHEMICAL HAZARDS OR LOW LEVEL RADIATION.

Occupational Group	Exposure Conditions	Reduction in Life Expectancy[a]	Basis of Risk Estimate	Comments	Reference
Nickel workers - Huntington, W. Va. N=1855	Males employed 1 or more years before 1948 followed through 1977	-832 days includes all causes of death	Based on differences in life expectancy at age 25 compared to 1960 U.S. population.	Increased life expectancy relative to general U.S. population. Probably due to "healthy worker effect".	Battelle 1982
Chemical workers 1 plant, N=2490	Male workers employed in plastics plant at least one year between 1949 and 1967. Followed through 1977.	-708 days including all causes of death	Same, except based on life expectancy at age 20	Increased life expectancy	"
Fibrous glass workers at 11 U.S. plants, N=14884	Males employed 1 or more years in production or maintenance jobs between 1945 and 1964.	-365 days including all causes of death	Same, except based on life expectancy at age 20.	Increased life expectancy relative to general population.	"
Mineral wool workers at 6 U.S. plants, N=1846	Males employed 1 or more years in production or maintenance jobs between 1945 and 1964.	106 days including all causes of death	"	Only one of the four cohorts which showed decreased life expectancy relative to U.S. males.	"
Nickel workers	Same group as nickel workers described above	117 days- due to cancer alone.	Life expectancy based on deaths due to all causes except cancer in U.S. males vs. each of the four cohorts, respectively.	Cancer deaths eliminated statistically using method of Chiang (1968).	"
Chemical workers	Same group as chemical workers described above	397 days - due to cancer alone	"	"	"
Fibrous glass workers	Same group of fibrous glass workers as described above.	95 days - due to cancer alone	"		
Mineral wool workers	Same group of mineral wool workers as described above	314 days - due to cancer alone	"	"	"
U.S. radiation workers, both sexes	Dose of 5 rads/year over ages 20-65	320 days - due to cancer	Based on risk estimators from 1980 BEIR report	Based on extrapolated dose-response data rather than epidemiological studies of radiation workers per se. Includes only risks due to cancer.	Cohen, 1981

U.S. radiation workers, both sexes	Dose of 5 rems/year over ages 18-65	147-274 days - due to cancer alone	Life expectancy calculations based on occupational exposures at ages 20-70 which lead to an estimate of life shortening of 0.63 to 1.16 days per rem and a total cumulative dose of 5 (65-18) =235 rems. Underlying risk estimates from WASH-1400.	Risk estimates extrapolated from studies of diverse groups instead of epidemiological studies of radiation workers. Includes only risks due to cancer.	Gotchy, 1978
U.S. radiation workers, both sexes	**Dose of 5 rems/year over ages 18-65**	**289 days (relative risk/ life plateau)**	**Life expectancy calculations based on various dose-response models from the 1972 BEIR report.**	**Same as above**	**Bunger et al., 1981**
		253 days (relative risk/ 30 year plateau)			
		177 days (absolute risk/ life plateau)			
		167 days (absolute risk/ 30 years plateau)			
"	**Dose of 0.5 rems/year over ages 18-65**	**37 days (relative risk/ life plateau)**	"	"	"
		29 days (relative risk/ 30 year plateau)			
		18 days (absolute risk/ life plateau)			
		15 days (absolute risk/ 30 year plateau)			
"	"	**15-27 days**	**Life expectancy calculations based on occupational exposures at ages 20-70 which lead to an estimate of life shortening of 0.63 -1.16 days per rem and total cumulative dose of 0.5 (65-18) = 23.5 rems. Underlying risk estimates from WASH-1400.**	"	**Gotchy, 1978**
"	"	**12 days**	**Based on risk estimates from 1980 BEIR report.**	"	**Cohen, 1981**

(a) Negative sign indicates increased life expectancy.

LA-UR-82-2931

RESPIRATORY PROTECTION AT NUCLEAR POWER PLANTS*

by

Bruce D. Reinert

Los Alamos National Laboratory

I. BACKGROUND

The Industrial Hygiene Group at Los Alamos National Laboratory has been conducting respiratory protection research and development since 1969. This work has been supported by the Nuclear Regulatory Commission (NRC), Department of Energy (DOE), National Institute for Occupational Safety and Health (NIOSH), Bureau of Mines (BOM), Air Force, Army, Navy and Occupational Safety and Health Administration (OSHA). All participating agencies have benefited mutually because of their similar interests in protecting the health of workers. NRC and DOE are primarily responsible for the establishment of this program at Los Alamos and have been the largest supporters of the program. The NRC has supported some of the most important and beneficial respirator research conducted at the Laboratory. The major areas of research and development supported by NRC have included development of respirator fit testing methods and equipment, determination of protection factors for various classes of respirators, evaluation of the performance of respiratory protective equipment, technical assistance on special respiratory protection problems, and development of visual aids and guides for respirator training.

All NRC licensed nuclear power plants benefit from these activities because a comprehensive respiratory protection program is a vital part of any occupational radiation protection program. There are many operations,

*Work supported by the Nuclear Regulatory Commission and performed at the Los Alamos National Laboratory operated under the auspices of the US Department of Energy, Contract No. W-7405-ENG-36.

particularly maintenance, in a nuclear power plant where engineering controls are not feasible for controlling airborne radioactive materials that pose a respiratory hazard. In such cases only respiratory protection equipment is available to reduce occupational radiation exposure. Information available from the Los Alamos research efforts have kept the NRC and its licensees aware of the latest advances in respiratory protection, as well as being aware of the performance of various respiratory protective equipment. This allows the licensee to limit radiation exposure to their employees, and obtain the most benefit for their money in occupational radiation protection. A short review of the highlights of this program at Los Alamos, past and present, is presented here.

II. FIT TESTING AND PROTECTION FACTORS

In the past, respirator facepiece fit was determined qualitatively by pressure checks, use of irritating aerosols, or by detection of the odor of a challenge vapor. Each of these qualitative tests relies on the wearer of the respirator to detect the challenge aerosol/vapor if it leaks past the respirator. Individual variation in response to these aerosols/vapors results in varying abilities to detect them. Because of the variability of response to these aerosols/vapors only low protection factors are allowed for this type of fit test. These low protection factors limit the use of air-purifying devices and would require the use of positive pressure atmosphere-supplying devices for the majority of work that would be done at power reactors.

Los Alamos has developed respirator fit test methods and equipment that allow respirator users to quantitatively determine what type of respirator facepiece best fits an individual. Whereas the qualitative fit test determines whether a respirator leaks, the quantitative fit test determines the degree of leakage. Quantitative fit testing involves measuring the concentration of a test atmosphere both outside and inside the respirator. A protection factor is calculated by the ratio of the test atmosphere concentration outside the respirator to the concentration inside the respirator. Assuring a good facepiece fit by quantitative means allows one to assume a higher level of protection than is acceptable if other fitting methods are used. As part of this program, Los Alamos has developed quantitative fit test equipment that is now commercially available as well as the test method used to minimize variability of results while adequately simulating workplace conditions.

Los Alamos developed the use of protection factors for various classes of respiratory protective equipment. Protection factor is a number assigned to a class of respirator representing the minimum degree of protection that the respirator is thought to provide for the majority of users. These protection factors were developed to aid NRC licensees in selecting the best type of respirator for a particular degree of hazard. Subsequent consensus standards have adopted this concept as an aid to respiratory protective equipment selection.

III. EVALUATION OF RESPIRATORY PROTECTIVE EQUIPMENT

Respiratory protective device evaluation has been ongoing at Los Alamos since 1969. Many categories of devices have been evaluated; however, NRC-sponsored research has been particularly significant in the area of atmosphere-supplying equipment. In fact, this research has provided the most comprehensive studies ever conducted on evaluating the performance of the various types of atmosphere-supplying devices. Testing of atmosphere-supplying respirators has included complete evaluations of airline respirators, self-contained breathing apparatus (SCBA), and emergency escape SCBA. These evaluations have covered weight of the device, size, in mask air pressures, ease of donning, airflow, wearing comfort, alarms, and evaluation of respiratory protection performance. This type of test protocol is significantly more comprehensive than that used by the regulatory agencies that approve respiratory protective equipment. In using such a comprehensive protocol, Los Alamos provides NRC and its licensees significantly more information than is available from the regulatory agencies that approve respiratory protective equipment. This additional information includes evaluation of not only the respiratory protection characteristics of the device but also wearing and use characteristics of the device that may be critical to the choice of the type of device for use in nuclear reactors. Special emphasis is placed on the application of these types of protective devices to the hazards encountered at nuclear facilities during these evaluations.

IV. TECHNICAL ASSISTANCE AT THREE MILE ISLAND

Technical assistance to NRC has been a continuing object of this program. Special hazards associated with nuclear facilities have required unique solutions in the area of respiratory protective equipment. Los Alamos has provided a ready source of expertise to NRC on such problems at Three Mile Island. Los Alamos provided on-site assistance to NRC on assuring the proper respiratory protective equipment was used during recovery operation.

This assistance covered consulting on most aspects of the respirator program including respirator selection, fitting, training, and maintenance. Particular emphasis was placed on assistance in selecting the proper types of respiratory protection to get the job done while still providing adequate protection. Since radioiodine was a problem during the recovery phase Los Alamos was asked to recommend what types of canisters and/or cartridges could be used with air-purifying respirators. Atmosphere-supplying respirators were not always practical for this application because of the difficulty of supplying airhoses to all work areas and the limited work time available with SCBA. An evaluation was conducted at Los Alamos of using air-purifying respirators for protection against radioiodine. This evaluation and final recommendations were based on review of available devices and also laboratory work to establish the performance of commercially available sorbent canisters against elemental iodine, hydroiodous acid, and methyl iodine.

A selection of atmosphere-supplying devices to support other recovery operations was made. This included recommendation of a combination airline/air-purifying device and selection of a closed-circuit, self-contained breathing apparatus to provide longer stay times in hot areas. The selection of the closed-circuit SCBA involved laboratory testing at Los Alamos to assure adequate protection of the users.

Technical assistance at Three Mile Island provided valuable information to the NRC, its licensee, and Los Alamos on the unique aspects of respiratory protection during emergency situations. It has required a new look at the requirements for respiratory protection in these situations and also demonstrated the need for a significant amount of preplanning if recovery operations are to be conducted with a minimum of radiological exposure problems. A project to develop a manual for respiratory protection in radiological emergencies was developed as a result of the lessons learned at Three Mile Island.

V. TRAINING

NRC has used the expertise available at Los Alamos to assist them in developing training programs for respiratory protection. Throughout the 13 years that Los Alamos has been providing respirator support to the NRC it has sponsored several symposiums and training courses on respiratory protection. These programs provided information to NRC and NRC licensee personnel on the details of establishing a respirator program, regulatory requirements, and in the case of the symposiums offered a forum to discuss mutual respiratory protection problems. Los Alamos has produced several video tapes that are available to licensees to help them in establishing and maintaining their respiratory protection programs. This has included the following training tapes.

1. Acceptable Practices for the Use of Air-Purifying Respirators
2. Acceptable Practices for the Use of Atmosphere-Supplying Respirators
3. Acceptable Practices for Fitting Respirator Users
4. Acceptable Practices for Cleaning, Inspection, Maintenance, and Storage of Respirators

We have also assisted NRC in the development of Regulatory Guide 8.15, Acceptable Programs for Respiratory Protection, and NUREG-0041, Manual of Respiratory Protection Against Airborne Radioactive Materials, which detail the requirements of a respiratory program for power reactors and other NRC facilities.

VI. CURRENT ACTIVITIES

During FY 1982, the program at Los Alamos was directed to

a) provide the NRC with information necessary to supplement NRC Regulatory Guides covering the use of respirators for protection against inhalation of airborne radioactive materials;

b) develop an NRC manual of respiratory protection practices for radiological emergencies such as Three Mile Island. This manual will be published as a NUREG report detailing practical information to guide NRC and its licensees in implementing a respirator program during these vital work situations. The report will cover equipment requirements under postulated emergency conditions, contaminants of concern, personnel, skills necessary to implement and maintain the program and administrative requirements. The manual will be organized to provide specific information for the various type of licensee operations such as power reactors;

c) advise NRC of new developments in respiratory protection so these developments can be promptly integrated into programs at NRC facilities. This is an ongoing program that provides NRC with information on new respiratory protective devices, new test methods, changes in test equipment, and impending consensus standards that may be of interest to NRC and its licensees;

d) provide technical assistance and laboratory support in respiratory protection for guidance necessary for standards development, compliance cases, NRC licensee program reviews or other actions within the responsibility of the NRC. A report of a survey of licensee respiratory protection programs was completed during FY 1982. This survey was conducted at selected power reactors, uranium mills, and research reactors. An evaluation was made of the respiratory protection programs at each facility and NRC was given recommendations for improvement of the overall respiratory protection programs for licensees based on what was seen at the facilities visited by Los Alamos. Also, under this assistance program, Los Alamos evaluated three powered air-purifying respirators (PAPR), provided information on measurement and control of airflows in supplied-air respirators, and evaluated special PAPR for use at Three Mile Island;

e) provide criteria for test procedures and instrumentation for evaluating performance and defining protection factors of respiratory protective equipment. Many operations at NRC licensee facilities have respiratory protection requirements that cannot easily be met by currently available NIOSH approved respirators. In such cases, special unapproved devices may be all that is available. To assure that these types of devices are adequately evaluated, NRC has requested that Los Alamos establish a system to evaluate such special respiratory protective equipment. A charter is also being prepared to establish a committee of respirator experts who will review the Los Alamos test results, formulate conclusions, and make recommendations to NRC on whether the tested device should or should not be accepted for use by the licensee. Another project recently began under this program which involves comparing the use of monodisperse and polydisperse aerosols for quantitative respirator fitting and quality assurance testing of respirator filters. This project proposes to determine if a single type of aerosol can be used for both purposes. Use of a single aerosol would greatly reduce the cost of equipment needed to support a respiratory protection program; and

f) develop criteria and test methods for certifying air-purifying respirators against elemental, organic vapor, and gaseous forms of radioiodine. Since radioiodine is a hazard of particular interest to power reactors, NRC requested that

Los Alamos develop some criteria and test methods for radioiodine cartridges so that NIOSH could certify commercial cartridges and canisters for use in atmospheres containing radioiodine. This required identification of environmental conditions of use, development of testing apparatus and procedures, experimental studies with comme cial sorbents and cartridges to identify parameters affecting this performance, development of acceptable performance and approval criteria, and transfer of testing technology to NIOSH to establish a respirator sorbent approval schedule for iodine. Transfer of the testing technology to the NIOSH Testing and Certification Branch was completed during FY 1982.

Future work planned for NRC includes developing test procedures to determine the field performance of respiratory protective equipment. These test procedures will be designed to, as realistically as possible, determine how well a respirator works when challenged with hazardous materials in the workplace. The project will attempt to determine how the field performance of respirators compares with the perfo mance as determined under controlled laboratory conditions. Future phases of this project will involve actual field testing and data evaluation.

Additionally, the work on setting up a system for evaluating the performance of special respiratory protective equipment will be continued. The test protocols and charter of the review committee will be reviewed with NRC. The review committee will be established and evaluation of special respiratory protective devices will be conducted.

PUBLICATIONS

1. "Evaluation and Performance of Open-Circuit Breathing Apparatus," NUREG/CR-1235 (January 1980).

2. "Evaluation and Performance of Escape-Type Self-Contained Breathing Apparatus," NUREG/CR-1586 (July 1980).

3. "Evaluation and Performance of Closed-Circuit Breathing Apparatus," NUREG-CR-2652 (April 1982).

DECONTAMINATION EFFECTIVENESS AND ITS IMPACT ON OCCUPATIONAL EXPOSURE REDUCTION*

L. F. Munson and W. D. Reece

Radiological Sciences Department
Pacific Northwest Laboratory**
Richland, WA 99352

This program, now in its first year, will determine the available methods and incentives for various decontamination options. Methods of primary system decontamination, major component decontamination and fuel surfaces decontamination will be included.

This paper describes a computer code being developed at the Pacific Northwest Laboratory (PNL) to quantitively estimate the radiation exposures from decontamination tasks and the radiation exposure savings from these efforts. We intend to call the code 'PERCS' for 'personnel exposure from right cylindrical sources.'

The program will calculate the dose from any number of designated, distributed cylindrical sources for any number of dose points, allowing any configuration of cylindrical or rectangular slab shields. For the geometry encountered in decontamination work, PERCS offers many improvements over current point kernel shielding codes such as ISOSHLD or Monte Carlo codes such as MORSE. Some of the principal advantages are as follows:

1) Completely arbitrary geometry is allowed in pipe location, pipe sizes, source distribution (both spatially and isotopically), composition of material inside pipe, composition of pipe wall, and placement of auxiliary shields.

---

* Work for this project was performed for the U.S. Nuclear Regulatory Commission under Department of Energy Contract DE-AC06-76RLO 1830 - Related Services.
** Operated by Battelle Memorial Institute.

2) The shielding of the distributed source by the pipe itself and its contents is calculated.

3) All other pipes and shields on a line-of-sight between the differential source point and the dose points are included in the shielding and buildup calculation.

4) The principal advantage of this code is speed. For example, for one distributed source and one dose point, this code is typically a factor of five to ten faster than other codes. Furthermore, because the primary use of this code will be calculation of dose in containment where the shielding geometry will not change with time, as each dose point is calculated, up to ten different source distributions can be integrated at the same time - to allow for changing isotopic concentrations as the plant ages or to accommodate various decontamination options. Because the geometric relations do not have to be recalculated, the run time of the code with ten source distributions is only 5% to 15% greater than the time to calculate a single source distribution.

For every dose point, PERCS chooses each source pipe in turn and calculates all the parameters, such as upper and lower limits of integration, needed to initiate the numerical integration of the point kernel. The basic double integration through angle and length of pipe is done by spline quadrature routines or eight-panel Newton-Cotes quadrature routines, depending on the location of the dose point relative to the source pipe. As each differential source element is chosen, all other pipes and auxiliary shields are checked to determine if they lie between the source element and dose point. A tally is kept of the contribution to optical thickness and the relative location of each shield.

After these geometric calculations are done, the concentration of each isotope in the differential area is computed and each gamma energy and yield for that isotope is used, along with the calculated buildup factor, to evaluate the point kernel. Also at this time, up to ten different source concentrations can be calculated as mentioned above.

The quadrature routines are then used to calculate the integral with respect to all angles around the pipe and these integrals are used to compute the integral along the length of the pipe.

After this, the next source pipe is chosen and the calculations outlined above are repeated. A running sum is kept of the dose from each pipe to give the total dose for that point. The next dose point is selected and all calculations are done again until all dose points have been calculated.

One check which was done to assure that the code is operating correctly was to run a simple geometry case which could be calculated by a general purpose code like ISOSHLD. For a pipe 10 meters long with a radius of 10 cm, filled with water and having walls of iron 1 cm thick the dose calculated by ISOSHLD and PERCS is given in Table 1.

TABLE 1. Dose From 1 Curie of Cobalt-60 Distributed On the Inside Surface of a Pipe (mrem/hr)

	ISOSHLD Buildup = 1.0	PERCS Buildup = 1.0	ISOSHLD	PERCS
Dose point at pipe end and:				
36 cm from centerline	108.0	121.4	201.9	240.9
61 cm from centerline	61.77	68.92	112.0	141.5
Dose point at pipe midplane:				
36 cm from centerline	210.4	237.7	395.1	470.4
61 cm from centerline	121.7	145.8	220.0	293.6

As can be seen, when the buildup factor is set to one so that only the geometry of the source is considered, PERCS and ISOSHLD agree to within 10 percent. When buildup factors are selected by the code, ISOSHLD, doses are considerably lower than PERCS because ISOSHLD neglects the buildup effect from the water in the pipe.

For a second verification, a particular case that allows hand calculation was used. If the dose point sets exactly on the centerline of the source pipe and the pipe is empty, the point kernel can be integrated analytically. For a pipe 50 cm long, with a radius of 10 cm, the results of the analytical expression and PERCS is given in Table 2. PERCS and the analytical method show good agreement.

TABLE 2. Dose From 1 Curie of Cobalt-60 on the Inside Surface of a Pipe (mrem/hr)

Dose Point Location	PERCS	Analytical
50 cm from pipe	1.58	1.53
20 cm from pipe	5.65	5.47
10 cm from pipe	13.1	12.7
pipe end	232	238
5 cm inside pipe	469	448
10 cm inside pipe	475	462
25 cm inside pipe	461	469

The code is currently being used to estimate dose rates from crud deposits on the Surry steam generator.

The code, however, is not yet in finished form. The following refinements are being done:

- add graphics input and graphical display of output
- ensure all portions are "user friendly"
- prepare user manual
- verify with additional real data (input reactor configuration and real radionuclide distribution and compare calculated and measured dose rates).

Not only is the code expected to be extremely valuable in assessing dose for decontamination operations and dose savings from decontamination, but it may have other applications as well. We expect this code to become a standard tool for exposure estimation and extremely valuable in all types of ALARA evaluations.

While it was designed for reactors, it should be equally applicable to other nuclear facilities, such as reprocessing operations where right cylindrical tanks are an important source. With good data from operating reactors and/or good predictive models of isotope distribution, the code could be an important design tool.

SAND-82-2210C

Status Report on Nuclear Power Plant Design Concepts for Sabotage Protection

David M. Ericson, Jr.*
G. Bruce Varnado *

Background and Introduction

Concern about the security of commercial nuclear power plants began to surface in the early 1970's. Between 1973 and 1976, Sandia National Laboratories conducted a number of studies for the Nuclear Regulatory Commission that examined plant vulnerability to sabotage.[1,2,3] These studies identified three types of measures which provide sabotage protection: (1) physical protection or the security system, (2) plant design, and (3) damage control.

The study of plant design concepts for sabotage protection evolved from these early programs. The objectives of this effort were to estimate the potential value of various configurations of plant design and damage control measures for providing protection against sabotage, and to establish the impact of such measures on facility costs, operations, and safety.

To accomplish these objectives, a multi-task program was established which proceeded in the following way. The initial step was to select and characterize a baseline plant representative of LWR design practice. Practical design alternatives with a potential for increasing protection were then identified. Concurrently, sabotage events which may be amendable to damage control were identified. Results from these two efforts were combined to provide plant configurations that are alternatives to the baseline. A physical protection system consistent with current regulations was integrated with these alternatives to create a set of preliminary reference designs. For each of these designs, a limited analysis of safeguards effectiveness and impacts was performed. The initial effort, Phase 1, took place between 1978 and 1980. It was directed toward new designs; not retrofits to existing facilities.

Phase 1 Program[4]

The first step in the study was the selection of the Standardized Nuclear Unit Power Plant System (SNUPPS)[5] as the baseline plant (Figure 1). SNUPPS is a highly compartmentalized and standardized pressurized water reactor (PWR) design, several units of which are under construction. In addition, the innovative modeling techniques being used in the design process and the project management scheme provided a source for technical data[6] required in the analysis. SNUPPS was used only to define the system design, plant arrangement, and equipment locations for a

Sandia National Laboratories

baseline plant.* The physical protection system characteristics were developed by the authors based upon our understanding of NRC requirements and do not necessarily represent the approach to be taken by the SNUPPS licensees.

To prevent an unacceptable release of radioactive material from a nuclear plant, whether from accident or deliberate act, one must insure reactor trip in response to upset conditions, maintain coolant system integrity and inventory and remove decay heat. As we compiled the list of design alternatives to enhance safeguards, options were sought which would provide improved sabotage resistance for at least one of these functions. The goal was to identify practicable alternatives, document the concepts in a consistent fashion so they could be compared and then develop them to the point where they could be evaluated. For each alternative, we attempted to answer the following questions:

1. Is it technically feasible?

2. Can it be done with existing technology?

3. Is the change independent or does it involve multiple systems?

4. What are the impacts on operations and maintenance?

5. Are there any collateral benefits?

In the process potential design measures were categorized into four broad groups:

1. Hardening critical systems or locations,

2. Plant layout modifications,

3. System design changes, and

4. Addition of systems.

These four categories include measures ranging from those requiring little or no change in plant systems or layout through those which may require the addition of complete new operational systems. In the initial screening some 29 design measures were cataloged, examined and evaluated. Based upon that initial evaluation several options were selected for more detailed design and analysis. Those selected included:

---

*As a result of the Phase 1 effort, a boiling water reactor (BWR) design was examined in Phase 2 to insure that the conclusions reached in Phase 1 are generally applicable to light water reactors (LWRs).

1. Hardened Enclosures for Makeup Water Tanks.

2. Separation of Containment Penetrations for Redundant Trains of Safety Equipment.

3. Separation and Hardening of Redundant Trains of Safety Equipment.

4. Hardened Decay Heat Removal System.

5. Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary.

6. Design Changes to Facilitate Damage Control.

Options 2 and 3 were combined in the design process, and Options 5 and 6 were incorporated in all of the reference designs. As a result, there were essentially three alternate plant configurations examined:

1. Baseline plus hardened tank enclosures,

2. Physically separated and protected redundant trains, and

3. Baseline plus a hardened decay heat removal system.

The latter two concepts are compared with the baseline plant on Figures 2 and 3.

The design for physical separation (Figure 2) derives from the baseline plant and basically involves dividing the existing auxiliary building into three separate buildings and eliminating the control and diesel generator buildings. The redundant safety equipment is separated into two buildings, A and B, while the remaining non-safety equipment and control room are placed in a new, small auxiliary building. The Class 1E switchgear, diesel generators, batteries and associated electrical equipment are placed in the safety buildings. An auxiliary feedwater storage tank and a refueling water storage tank are located in each building. This results in storage of more water than might otherwise be the case, but cross connections are eliminated and the independence of the two buildings is enhanced. A key feature of this design is the provision for personnel access. Main access to the auxiliary building is via an access control building at ground level. Buildings A and B may be accessed individually from the auxiliary building, there is no direct cross access, A to B or B to A.

There are various ways to implement a hardened decay heat removal system. The design chosen for development here (Figure 3) uses electric power supplied by a dedicated diesel generator co-located with the rest of the system in a hardened building. Heat is removed from the reactor by supplying emergency feedwater to the secondary sides of the steam generators; the steam generated is then discharged to the atmosphere. Natural circulation provides primary system flow, but a charging pump is included for coolant inventory control. A separate pipe tunnel connects the hardened decay heat removal building to containment. The system is a single, 100% system without redundancy or single failure capability. The design period of unattended operation is 10 hours.

Methods for the evaluation of safeguards effectiveness or value are still evolving, and therefore, there is no single model or methodology which can be used to evaluate the effectiveness of plant design or physical protection systems against all threats to security. Similarly, there is no procedure which even attempts to model in a single, coherent package the impacts on plant design or operations associated with various alternatives. As a result, the evaluation involved a combination of quantitative or semi-quantitative models and subjective engineering judgment. The safeguards effectiveness of each alternative was estimated for external and internal threats. The external threat includes a determined, violent, external assault or an attack by stealth or the deceptive actions of several persons. The inside threat consists of an insider in any position.[7] The impacts were estimated first for the baseline plant by examining the numbers and types of personnel who must normally visit particular locations (equipment) and the frequency of those visits. Then, the study established whether or not the alternative designs cause significant perturbations to these operational procedures in terms of required manpower and frequency of visits. Capital costs for each design were also considered.

As noted earlier, the objectives were to estimate the potential value of various plant designs and to establish the impact of such measures on cost, operations and safety. These objectives were accomplished through a combination of quantitative and subjective analyses. It was observed that hardening makeup water tanks has the least impact (low cost, no effect on manpower) but at the same time the lowest value (no change for insider, only minor upgrade against external threat). Additional isolation of low pressure systems has some value in that a potential vulnerability to insider actions could be eliminated; that is, a loss of coolant outside containment from certain areas is eliminated. Physically separating the redundant trains is considered to have medium value and impact. There is an increase in protection against external threats when access doors are upgraded, but there is an associated impact upon staff access for inspection and maintenance. The complete redesign does involve some incremental increase in cost and the added equipment could lead to added manning. If this option is combined with administrative controls and work rules (facilitated by this design), then the option could have a higher value because of added protection against insider actions. Unfortunately, that higher value could be accompanied by additional impacts in terms of access for operations and maintenance. There could also be some negative staff reaction to the controls. The hardened decay heat removal system has been given a high value-medium impact ranking. Although the alternative does not alter protection for existing vital areas, the option adds a valuable, well-protected redundancy for essentially all transient events. There is a modest incremental cost, and depending upon how it is implemented there could be some increase in required manpower. There is a potential for increased protection against the insider. The isolation in a separate building, coupled with the added redundancy, certainly facilitates administrative controls, and because it is a separate building, it may be possible to exercise administrative controls without major, adverse impacts.

The range of alternatives considered in Phase 1 and the results of the analyses led to the following conclusions.

1. Structural design changes for PWR plants in and of themselves do not appear to provide significant additional protection against either the external or internal sabotage threat.

2. Design changes can, however, facilitate the implementation of more effective physical protection systems. For example:

   a. Design changes that restrict vital area access to a few well defined routes, if appropriately combined with administrative controls and work rules can increase the protection against the insider.

   b. Design changes that restrict outsider access to a few routes coupled with appropriate physical protection will increase protection against the external threat.

However, it must be observed that design changes that significantly revise plant layouts so as to limit access to vital areas and reduce outside access are practical only for new plants. It was also concluded that:

3. Damage control using installed plant systems in alternate (non-standard) ways has some potential for countering the effects of sabotage (or accidents). This requires further study.

4. Damage control by running repair and/or jury rigging does not appear to be a viable counter to sabotage because of the associated operational impacts and the potential for adversary interference with the damage control effort.

Phase 2 Program

As indicated above, the study was directed initially toward new plant designs. However, when the Phase 1 effort was concluded in early 1980, a number of factors had changed and these subsequently had a direct effect upon the direction and conduct of Phase 2. The TMI-2 incident occurred about midway in the Phase 1 effort, there had been no new plant orders placed and in fact some extensions and cancellations had occurred. The Phase 1 effort had emphasized the PWR designs more than originally intended. And finally, there was an increased concern at all levels about insider threats to the security of commercial nuclear power plants. As a result of these various factors, and based upon the recommendations from Phase 1, a Phase 2 Program was established with four tasks:

- Damage Control for Sabotage Mitigation

- Insider Protection

- Retrofitable Design Changes

- Design Changes for BWR

In considering damage control for sabotage mitigation we set out to extend/revise the matrix of options proposed in Phase 1. The emphasis was on use of existing systems "as is," but in alternate ways. The use of existing systems but with some added components was also considered. The insider protection effort involved establishing a ranking of systems to be protected (this has applicability to safety as will be discussed), reexamining equipment status monitoring, exploring possible design changes, looking at the operations/security interaction, and estimating impacts. The retrofitable design changes task also required some systems ranking as well as identification of the potential sabotage acts which could put particular items of equipment out of service. After identifying problem areas additional effort was directed toward design changes or fixes and the evaluation of their effects or impacts. The final task was to examine a typical BWR to insure that the conclusions from Phase 1 were applicable to LWRs in general.

Systems Ranking. Because it was required in several Phase 2 tasks, a method for system or equipment ranking was developed. In this method the Generic Sabotage Fault Tree (GSFT) is used to define the plant fault logic down to the system level. (If this were a safety study, the GSFT could be replaced with a standard fault tree, because the basic concern is similar: prevent a release of radioactive material which exceeds allowable guidelines). The threat to security is the same as that described earlier under the Phase 1 program. The next step is to develop simplified system interface diagrams to each system of concern The support systems required for components in each branchof the system are identified and noted on a simplified flow network. For example, an Auxiliary Feedwater System, Figure 4, appears as shown in Figure 5. Specific support systems required to provide makeup water from a source, R15, to a steam generator are identified. On one path we need the B trains of dc power, heating ventilating and air conditioning and engineered safety features actuation system.

Simplified fault trees are developed for the required systems (Figure 6). In these trees time is included explicitly. This allows us to account directly for the time at (or by) which a system must function in order to be effective. For convenience all direct system faults (e.g., pump disabled, pipe breached, valve closed) are lumped together because our concern here is not with the specific disabling acts - but rather with what systems must be available to prevent the release. The support system trees are developed in a similar manner. Once the trees have been developed, complement set solutions are obtained for the time independent and time dependent cases. A complement set is a minimum set of components and systems which could, if protected against sabotage, provide the capability to place the plant in a safe shutdown condition. The fault tree solutions are then used to identify strategies for establishing and maintaining a safe shutdown condition and to rank the systems according to the time they are required.

The types of strategies which evolve from the analysis are illustrated in Tables 1 and 2. In the PWR studied, all strategies require that large LOCAs be prevented, especially the shutdown cooling suction line LOCA because that would be a LOCA outside of containment. In one strategy, one must then have high pressure safety injection and auxiliary feedwater available to mitigate small LOCAs and transients. Similarly, in the BWR studied, all strategies require that RHR suction line LOCA's be prevented and that containment isolation be possible. One strategy in this case is to insure that high pressure core spray is always available and all conditions can be mitigated.

In ranking systems, those required in a short time (say in less than two hours) are Rank 1 systems. The remaining systems are ranked sequentially based upon the time at which they are required. If the ranking is being used to examine damage control options, then one can say that for Rank 1 systems damage control isn't practical because time is too short. Other systems are candidates for damage control. Some systems are always Rank 1 in the two reactor plants studied (Table 3). For the PWR, one must always have the scram system (RPS), dc power, RHR isolation and the auxiliary feedwater system. For the BWR one must have the RPS, the control rod drive system (to provide scram), dc power, shutdown cooling line isolation and containment isolation.

In summary, the ranking methodology illustrates protection strategies and ranks systems by considering time sequenced system requirements. Ranking results provide insights for damage control by indicating where it is not a credible approach and where it has potential use. Strategies and rankings are plant specific and will vary to some extent with the assumptions on adversary capabilities. Although the purpose here was to establish the relative importance of protecting individual systems against radiological sabotage, we believe the methodology can also be used to define systems which must be protected against accidents (or conversely, systems which must be functional to counter the effects of accidents.

<u>Damage Control</u>. There are two basic objectives in damage control:

1) Restore or maintain a functional capability required for safe shutdown, or

2) Extend the time available to restore by other means those functional capabilities that have been lost.

As noted earlier, damage control was considered in Phase 1, and it was concluded that conventional damage control (running repair or jury rigging) was not practical. Therefore it is not considered in Phase 2. Here we considered damage control as measures that involved the use of existing systems in normal or alternate modes of operation. In this context all required equipment is in place although system-level design changes may be required to facilitate some damage control measures. The types of system-level design changes considered included:

Fluid system cross connections

Electrical bus ties or load transfer capabilities

Local manual operating capability

Upgrading pumping capacity

Additional onsite ac electric power source

In all, some 27 damage control measures were examined and evaluated. These measures will not all be enumerated here, but several examples will illustrate the variety considered. At some BWRs, the high pressure coolant injection system could be modified to provide suppression pool feed and bleed cooling in the event residual heat removal, containment cooling and essential service water systems were lost. This is illustrated in Figure 7. In some instances, other existing high pressure coolant makeup systems can be substituted if the reactor core isolation cooling system is not available. Turbine driven pumps and associated auxiliaries can be modified to operate without electrical power thus countering potential loss of ac and dc power sources. In either a PWR or BWR, provisions could be included to reenergize non-class 1E loads from the class 1E system if the non-class 1E power system is lost. This would make available a much wider range of methods to control inventory or remove decay heat. This same capability could be provided through the addition of an alternate onsite source of non-class IE power, such as a gas turbine generator set. To make up for the potential loss of dc power, another measure would involve modifying the diesel generators for startup and loading without electrical power. In a PWR plant if the auxiliary feedwater system were unavailable, cross connections could be included to allow the high pressure safety injection system to be substituted for the AFWS; this is illustrated in Figure 8. The 27 damage control measures were evaluated - subjectively - in the following areas:

Technical feasibility

Effectiveness

System or plant modifications required

Operational impact

Regulatory Concerns

All of these are important, but the last item is particularly germane for those measures which involve fluid cross connections between normally separate systems or non-class 1E loads to the 1E buses.

Based upon this evaluation process, it was concluded that damage control is not a stand-alone safeguards measure, but it can make a useful contribution to sabotage protection as an element in an integrated system. Furthermore, implementation of system-level design changes and damage control should be considered on an individual plant basis. It was also observed that many of the design features necessary to implement damage control measures are not commonly found in current nuclear plants. Finally, it should be remembered that any systems or equipment intended for use in damage control must itself be protected against sabotage. It is also apparent that damage control measures can have an impact upon systems ranking. To establish the potential benefit of selected system-level design changes and an effective damage control capability, they should be included in the simplified fault trees used for system ranking. The solutions of such a revised tree will likely reveal additional strategies for maintaining the plant in safe shutdown and some revision to system rankings.

<u>Insider Protection.</u> There is increasing concern - triggered in part by some recent incidents - about the potential role of plant personnel in sabotage activities, and how such activities can be prevented. In this task,[10] a variety of techniques was explored for protecting a plant against unauthorized acts by an insider. Direct physical protection measures were explored including a number of area type safeguards as well as component level safeguards. The role of damage control measures was also examined, as well as the use of plant design changes. This latter category includes component and system level design and plant layout changes. In addition, the study also considered the implications of systems ranking, sabotage detection and security force response for protection against an insider. All of the physical protection measures involve combinations of procedures, personnel and hardware, especially intended to deter, delay, assess, and respond to sabotage threats.

Turning first to the direct physical protection concepts, there are several area type protection measures that could be employed. Team zoning limits all access to plant vital areas to teams of two or more persons. Thus it can be used in all plant areas without changing the physical layout. It does have an operational impact in terms of staffing and it can be defeated by overt action of a team member. Area zoning divides the plant into zones and personnel work only in specified zones. This allows access by a single person but it is really effective only for Type II vital areas and may require plant modifications and additional personnel. Operational zoning is an area measure which restricts an individual's access to vital areas based upon where he has been in past assignments, the relationship of the area he wants to enter to the one just vacated and verification of the operational status of equipment just serviced. This is a dynamic control, in that the areas an individual may access are continually changing. It does allow access by one person and won't require plant modifications. Unfortunately, it really protects only Type II vital areas, requires operational tests after maintenance or visit, and requires significant upgrades of the access computer system. Obviously, area and team zoning can be combined to protect Types I and II areas, but such a combination still may require plant modifications and additional people. Operational and team zoning may also be combined. Such an approach would protect Types I and II vital areas while allowing single person access to the latter. However, it is subject to defeat by overt action by one person and requires significant upgrade of the access computer. Time zoning would restrict the activity in a vital area to a particular time. Unfortunately, this would impact much of the maintenance activity which is predominantly unscheduled. Function zoning would restrict access to particular functional groups, and indeed many plant activities are controlled this way now. However, nearly every functional group requires access at one time or another to every vital area. Component level protection could be provided by operational control elements which are designed to detect illicit activities and delay such acts until appropriate response can be made. Such techniques provide protection across the plant to some minimum number of equipment items (the complement set discussed earlier). There are some significant drawbacks, however; many detectors may be necessary, their installation will affect normal operations and maintenance, and the sensors themselves must be maintained and protected.

Damage control measures and plant design measures were also examined for their interaction with insider protection techniques. No additional comment beyond that above is necessary except for some component level design changes which are discussed later.

The application of the system ranking methodology was also examined, but again no further comment is required beyond that provided above in the discussion of system rankings.

Three means of detection were considered. The operations control elements discussed above can be designed to provide detection wherever needed, but they do have a high maintenance impact and many individual elements could be required. Consideration was also given to using safety related display instrumentation as a way to detect unauthorized acts. The advantage with this equipment is that it is installed and plant personnel are familiar with it. The disadvantages include that fact that it is not effective for equipment which is in a standby mode. Also, it provides detection after the fact and could require a class 1E and non-class 1E interface. Finally, periodic operational surveillance was considered. It has the advantage that it fits in with existing procedures and routines. The problem is that it is not necessarily timely and the surveillance is variable depending the individuals involved. Security force response requires prior detection and coordination with operations personnel. Appropriate response does provide a way to interrupt a sabotage attempt, but it is strongly dependent upon timely detection.

When all of the above factors are considered, it is clear that no single approach provides adequate protection against the insider. Effective safeguards will require an integrated approach. Such an integrated system requires dynamic access controls for vital areas, incorporating a mix of operational zoning and operations control elements. This implies an upgraded safeguards computer system to handle the changing access conditions. This integrated approach would include some damage control measures as well as selected system and component level design changes to support damage control and the dynamic access controls. Finally, security force response would be tailored to be compatible with access controls.

Retrofitable Design Changes. Component vulnerability and component-level design changes are directly related to the other facets of safeguards analyses. Solutions of the fault trees identify vital equipment, vital and plant areas, and event sequences. Component vulnerability evaluations identify the specific sabotage actions which can disable particular components. In this evaluation the generic sabotage fault tree modules were used to identify potential sabotage targets. Then an engineering evaluation was performed to determine exactly how individual components could be damaged. Nineteen types of components were reviewed. Potential vulnerabilities were ranked in five areas: resources and time required to accomplish the act; the necessary operating status of the equipment, the need for sabotage to other components and the certainty of the results (that is, does the saboteur really know the outcome). Potential design changes to reduce or eliminate the vulnerability were ranked as to effectiveness, technical feasibility, first cost of implementation, and the impact upon maintenance. The details of this analyses are considered classified information. However, the conclusions are simply stated. Selected vulnerabilities can be reduced and in some instances eliminated, but any particular component cannot be made invulnerable. Component-level design changes are just one part of an integrated system for protection against an insider. Some design changes are retrofitable, but others require replacement of existing equipment with redesigned counterparts. Finally, most of the potential design changes are not useful against an adversary with explosives.

BWR Plant Analysis. This task characterizes a BWR plant, evaluates that plant using the same techniques used in Phase 1 and then defines appropriate conceptual design changes. Because there was considerable data available from an earlier study and because it represents the current state of BWR design, the design selected as a baseline is the BWR 6/MK 3 (the GESSAR nuclear island) with the balance of plant from the BRAUNSAR standard design. The basic site layout is shown in Figure 9. This design is quite compartmentalized as shown on Figure 10. The study concludes that compartmentalization here is comparable to that in SNUPPS, with safety related pumps in individual rooms,but accessed from a common corridor. The probability of interrupting an adversary in this BWR design is comparable to that in the PWR baseline. The BWR-6 design plus physical protection affords reasonable protection against the outsider. Based upon these similarities with the earlier PWR analyses, there was no need to examine design changes. It is clear, though, that protection against the insider will require additional controls.

Summary. This program has examined plant design changes for their impact upon sabotage protection. Given the compartmentalization present in current generation plants similar to SNUPPS and BWR 6 and the level of physical protection assumed in this study, even drastic changes in design don't significantly affect the protection against an external threat. In Phase 2 a systems ranking methodology was developed which can be used in both safety and sabotage studies. Damage control measures can be used in sabotage and accident mitigation. However, it is clear from these studies that to be effective, damage control must use installed equipment, and significant portions of that must be available in a very short time following an incident. Insider protection is a difficult task, requiring the integration of a number of techniques to be effective. Component level vulnerabilities can be reduced by careful design, but they can not be eliminated. The work done here in damage control, systems ranking and component vulnerability may have direct application to safety issues.

References

1. Safety and Security of Nuclear Power Reactors to Acts of Sabotage, SAND75-0504 (Albuquerque, NM, Sandia Laboratories, March 1976).

2. Protection of Nuclear Power Plants Against Sabotage, SAND77-0116C (Albuquerque, NM, Sandia Laboratories, October 1977).

3. Summary Report of Workshop on Sabotage Protection in Nuclear Power Plant Design, SAND76-0637 (Albuquerque, NM, Sandia Laboratories, February 1977).

4. David M. Ericson, Jr., and G. Bruce Varnado, Nuclear Power Plant Design Concepts for Sabotage Protection, NUREG/CR-1345, SAND80-0477 (Albuquerque, NM, Sandia National Laboratories, January 1981).

5. Standardized Nuclear Unit Power Plant System (SNUPPS) Preliminary Safety Analysis Report, (Rockville, MD, Kansas City Power and Light Co., Kansas Gas and Electric Co., Northern States Power Co., and Union Electric Company, January 1976).

6. Nicholas A. Petrick, "SNUPPS - The Multiple Utility Standardization Project," Nuclear Engineering International, November 1975, pp. 935-941.

7. "Requirement for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Industrial Sabotage," Section 73.55 in "Energy," Chapter 10, Code of Federal Regulations (Washington, DC, GSA, Office of the Federal Register, January 1, 1979).

8. Peter R. Lobner, et al., Ranking of Light Water Reactor Systems for Sabotage Protection, SAND82-7053 (Albuquerque, NM, Science Applications, Inc., for Sandia National Laboratories, July 1982).

9. Peter R. Lobner, Nuclear Power Plant Damage Control Measures and Design Changes for Sabotage Protection, NUREG/CR-2585, SAND82-7011 (Albuquerque, NM, Science Applications, Inc., for Sandia National Laboratories, May 1982).

10. Lewis A. Goldman and Peter R. Lobner, A Review of Selected Methods for Protecting Against Sabotage by an Insider, NUREG/CR-2643, SAND82-7036 (Albuquerque, NM, Science Applications, Inc. for Sandia National Laboratories, August 1982).

11. Peter R. Lobner, et al., Component Vulnerabilities and Component Level Design Changes for Protection Against Sabotage by an Insider (U), NUREG/CR-2693, SAND82-7040, Confidential-NSI (Albuquerque, NM, Science Applications, Inc., for Sandia National Laboratories, August 1982). This report is classified because of detailed vulnerabilities discussed. Any requests for information pertaining to the report should be directed to the Safeguards Branch, Division of Facility Operations, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555.

12. David M. Ericson, Jr., and Charlene P. Harlan, Evaluation of Design Concepts for Sabotage Protection at a Boiling Water Reactor (BWR) Nuclear Power Plant, NUREG/CR-XXXX, SAND82-XXXX (Albuquerque, NM, Sandia National Laboratories, To Be Published).

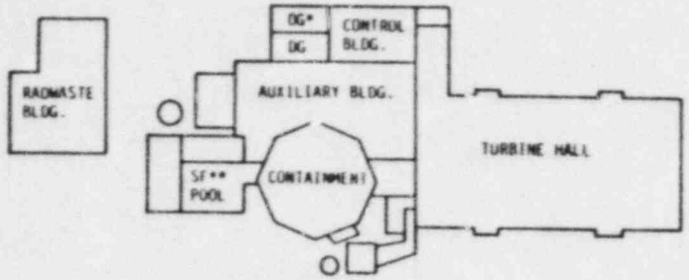


Figure 1. Standardized Nuclear Unit Power Plant System (SNUPPS), Baseline Plant Layout.

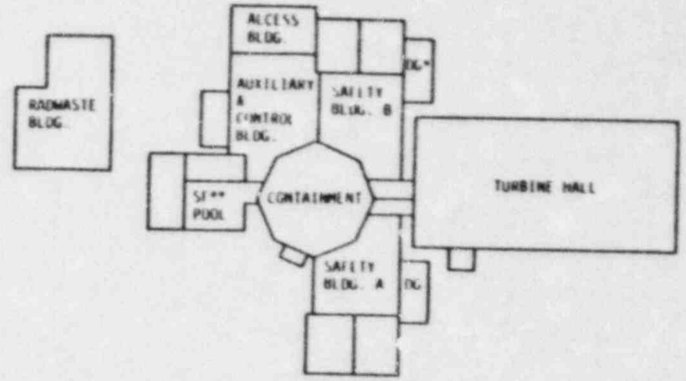


Figure 2. Completely Separated Trains of Safety Equipment, Plant Layout.

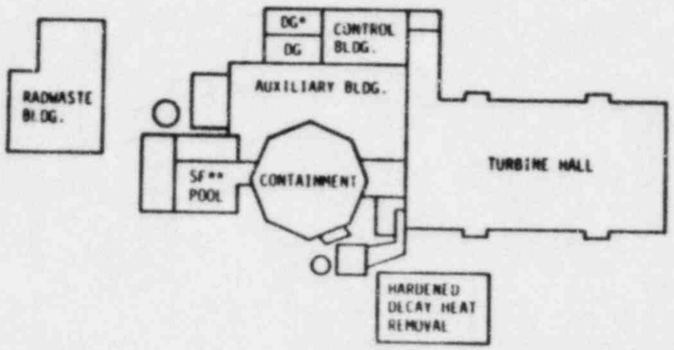


Figure 3. Additional System for Decay Heat Removal, Plant Layout.

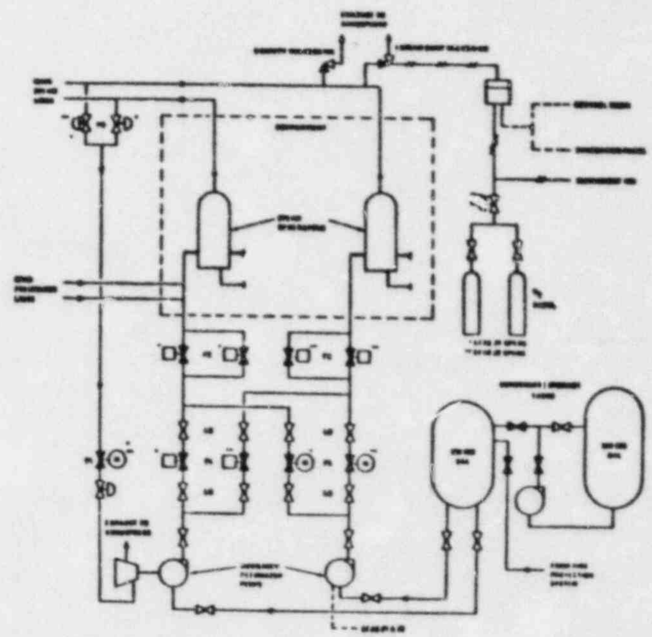
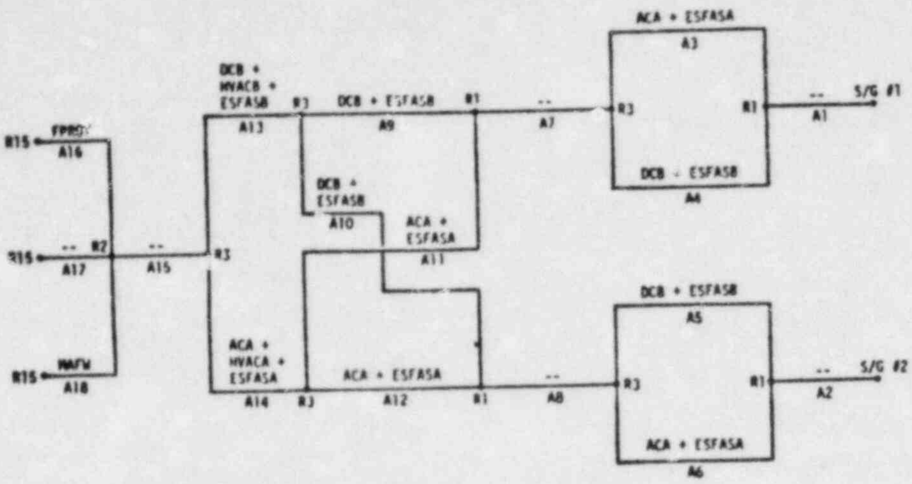


Figure 4. Typical PWR Auxiliary Feedwater System.



Note: Terms are standard Safety Analysis Report abbreviations plus a train designation, i.e., DCB = DC Power, B Train.

Figure 5. Simplified/System Interface Diagram.

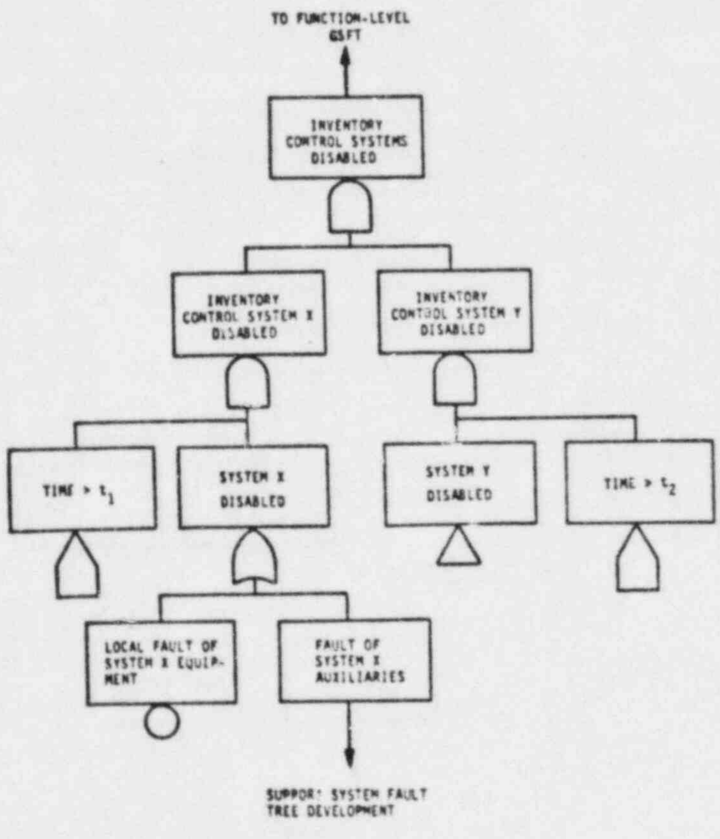


Figure 6. Simplified System Fault Tree.

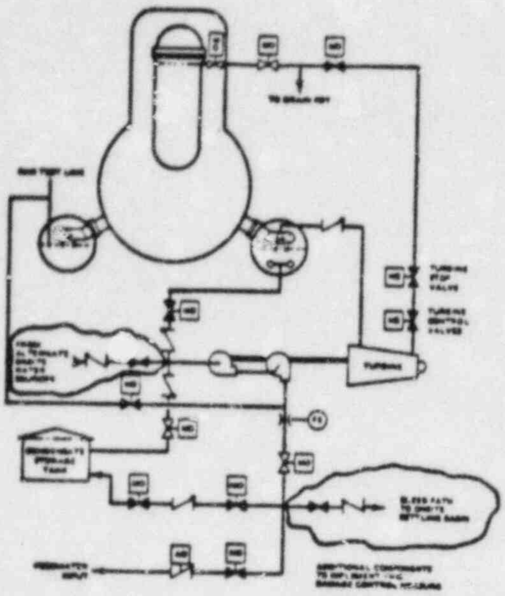


Figure 7. HPCI System Modifications for Suppression Pool Cooling.

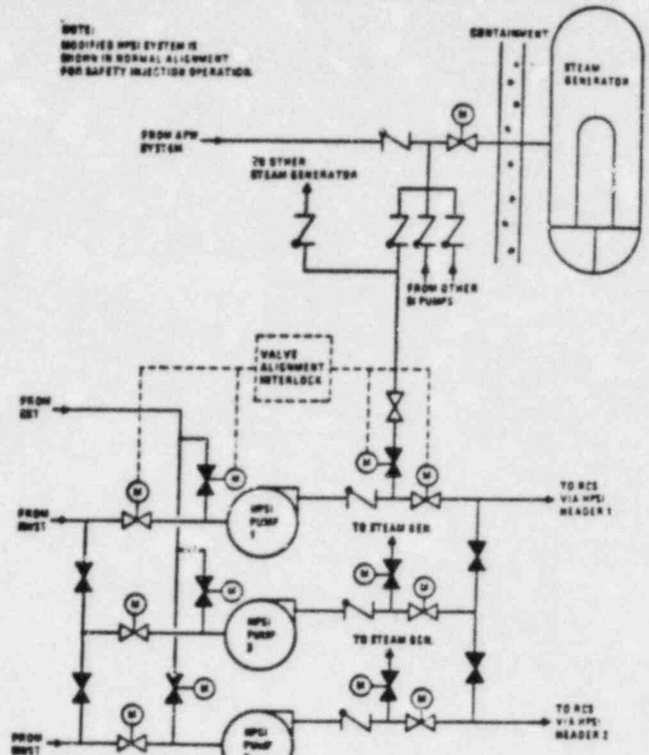


Figure 8. HPSI Systems Modifications for Backup AFW Capability.

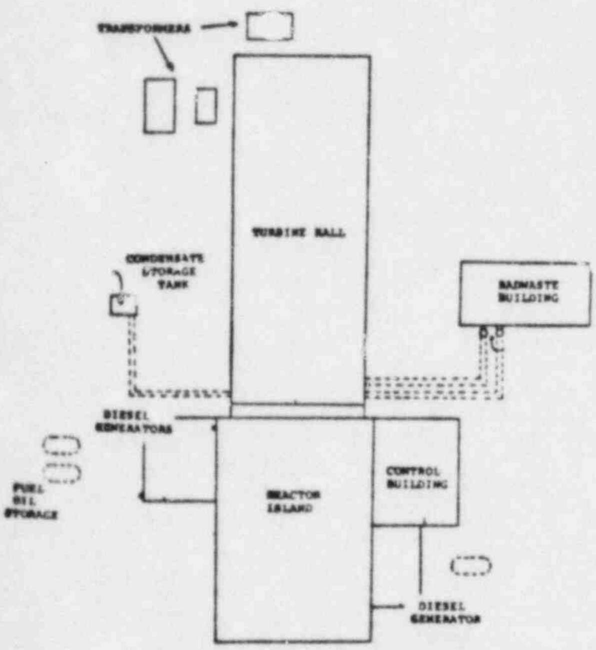


Figure 9. BWR 6 Baseline Plant Layout.

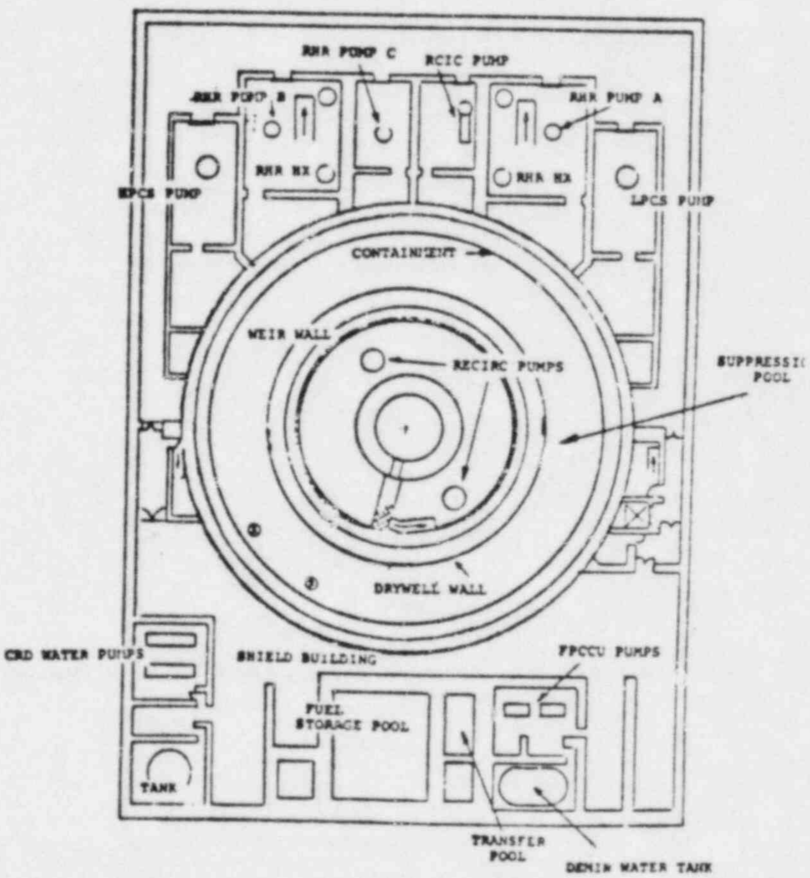


Figure 10. Safety Pump Compartmentalization in BWR 6 Baseline Plant.

PWR System Strategies*

Strategy	Mitigating System		
	Large LOCA	Small LOCA	Transients
1	(Prevented)	HPSI + AFW + Containment	HPSI + AFW
2	(Prevented)	HPSI + AFW	HPSI + AFW
3	(Prevented)	LOCA Isolation Capability	AFW + CVCS
4	(Prevented)	(Prevented)	AFW + CVCS

*All require prevention of shutdown cooling suction line LOCA

Table 1. PWR System Protection Strategies.

BWR System Strategies*

Strategy	Mitigating System		
	Large LOCA	Small LOCA	Transients
1	HPCS	HPCS	HPCS
2	LPCS	LPCS + ADS	LPCS + ADS
3	LPCI	LPCI + ADS	LPCI + ADS
4	LPCS	RCIC	RCIC
5	LPCI	RCIC	RCIC
6	(Prevented)	RCIC	RCIC

*All require containment isolation capability and prevention of RHR suction line LOCA

Table 2. BWR System Protection Strategies.

RANK 1 SYSTEMS IN ALL SYSTEM STRATEGIES

PWR

- RPS
- DC power
- RHR/RCS suction line isolation valves
- Auxiliary feedwater system

BWR

- RPS
- Control rod drive hydraulic system
- DC power
- Shutdown cooling suction line isolation valves
- Containment local

Table 3. Rank 1 Systems in All Protection Strategies.

LA-UR-82-2831

REACTOR SABOTAGE VULNERABILITY AND VITAL EQUIPMENT IDENTIFICATION

by

J. M. Boudreau and R. A. Haarman
Los Alamos National Laboratory

Abstract

Two ongoing programs at Los Alamos, the Vital Area Analysis Program and the Reactor Sabotage Vulnerability Program, are discussed. The Laboratory has been providing the Nuclear Regulatory Commission with technical support in identifying the vital areas at nuclear power plants through the use of sabotage fault trees. This procedure is being expanded to provide support for the Reactor Sabotage Vulnerability Assessment Program. A re-examination of some of the original system modeling assumptions, including a survey of the applicable research, is underway. A description of the survey work and the computerized data bases being used is provided. This program is expected to result in refinements in the existing procedures.

Introduction

This paper will discuss briefly the work performed by the Los Alamos National Laboratory for the Vital Area Analysis (VAA) Program. It also will outline the newly initiated Reactor Sabotage Vulnerability Program. Both of these programs are being performed under contract for the Nuclear Regulatory Commission (NRC).

Since 1979 Los Alamos has been providing the NRC with technical support for determining the locations of vital areas, as defined in 10 CFR 73, for all power reactors in the United States. The NRC now is considering expanding the vital area analysis procedure to provide support for the Reactor Sabotage Vulnerability Assessment Program. A re-examination of certain assumptions currently used by Los Alamos or proposed by the NRC relating to reactor sabotage is required to extend the previous work.

Vital Area Analysis Program

Since the Vital Area Analysis (VAA) Program's inception in 1979, Los Alamos has visited almost all of the operating reactors and approximately 10 plants undergoing their operating license review as part of the Laboratory's Vital Area Analysis Program.[1] The results of the program are used as a resource by the NRC licensing staff to identify vital equipment and areas at the plants that require protection and to verify the licensee-identified vital areas.

The method used to perform the analysis focuses on the fault-tree approach to systematically identify the sabotage scenarios and equipment locations in the plant.[2] The vital area fault-tree methodology was developed by Sandia National Laboratories, Albuquerque (SNLA), in the early 1970s for the NRC's Office of Nuclear Regulatory Research (RES).[3] Starting in 1979, the method was applied to specific plants by Los Alamos for the Office of Nuclear Reactor Regulation (NRR) and most recently for the Office of Nuclear Material Safety and Safeguards (NMSS). This technique has proved to be an excellent tool for performing detailed and systematic analyses of complex plants.

The vital area fault-tree methodology uses the SETS computer code to solve the massive fault trees to provide the results in a usable format. SNLA is continuing its efforts in modifying the code to provide time-saving techniques for computer usage, and cooperation between SNLA and Los Alamos is required to provide interaction between the developer of the code and its user. The formation of the fault tree is central to the whole program. The accurate representation of the plant is essential for credible results. Solving the fault tree requires sophisticated numerical manipulation, and computers are well suited to the process.

Los Alamos uses a multistep procedure in the VAA program that is intended to efficiently gather the necessary data for input to the fault tree. This technique consists of an FSAR review, a site visit, data reduction, formation of a fault tree, and a computer solution. Los Alamos engineers spend time at each plant to gather the site-specific information needed to develop the tree. The initial fault-tree formation uses a combination of generic subtrees to represent the plant. However, experience has shown that all plants differ widely in site-specific data, hence the fault-tree development tends to be an iterative process that concludes with a unique fault tree for each plant. The plant personnel who provide the most useful information are members of the operating, training, licensing, and maintenance staff. A typical site visit is 1 week long and starts with discussions with plant-systems-oriented personnel to establish the initiating events and the system mitigating capabilities. During the discussions, the operating procedures are reviewed to determine system and operator responses. Once the appropriate systems are identified, the Piping and Instrumentation Diagrams, Electrical Single Line drawings, and associated control system drawings are examined and physical locations are noted. Los Alamos engineers make verification inspections of selected equipment locations throughout the plant and maintenance personnel are consulted for various appraisals of component vulnerabilities. The information then is brought back to Los Alamos where the engineers develop the complete trees for eventual computer input. The results are compared with the information received at the plants, and often the plant personnel are consulted again to provide a double check on the input data before submitting the results to the NRC. The entire process takes approximately 6--10 weeks to complete.

Sabotage fault trees differ from safety fault trees in one important area--single failure criteria are not considered for sabotage-related scenarios because the saboteur is not restricted to damaging a single piece of equipment. This has led to the inclusion of multiple-failure scenarios in the sabotage fault trees, which provides a different set of assumptions than might be found on a safety tree. Because most light-water reactor safety work has been done assuming single-failure criteria and system interactions in the sabotage mode are not as well understood, there has been a tendency to use conservative assumptions in the sabotage trees. A good example of this was in the case of whether to permit a plant to use the feed-and-bleed mode of recovery in the event it has lost its feedwater capability. In 1978, calculations were performed using the Los Alamos TRAC (Transient Reactor Analysis Code) for a B&W plant to determine whether the plant should be given credit for using feed and bleed as an alternate procedure to auxiliary feedwater in a safeguards situation.[4] This run was made because the vital area designation impact was significant and it involved multiple failures that had not been considered before in the safety area. Not until after the TMI incident, where a similar scenario was involved, was the feed-and-bleed scenario more fully developed. Generally, the sabotage tree will not include credit for recovery modes that have not been reviewed and approved by the NRC. Here again the flexibility of the tree and computers make changes fairly simple; the analyst is able to focus on the localized problem and use the computer to perform the impact analysis in a straightforward approach.

The most difficult part of the sabotage tree to develop is in the area of determining the system or combination of systems that is required to mitigate various saboteur-initiated incidents. The difficulty is a result of the lack of information in the safety area when multiple failures are considered. It should be stressed that this lack of information does not cause the vital area analysis results to be wrong in the sense that areas that contain vital equipment are not identified, but rather it is entirely possible that more safeguards requirements are put in areas of the plant where they are not required. The case of "better too many than not enough" may appear to satisfy the notion of security. However, when plant operations are considered, these safeguards requirements may affect safety adversely.

It is intended that the reactor sabotage vulnerability and vital equipment identification programs will concentrate on providing the most recent research work applicable to the fault tree formation and thereby eliminate unnecessary conservatism.

Reactor Sabotage Vulnerability Program

As mentioned earlier, the NRC's reactor sabotage vulnerability assessment program is based on the VAA procedure. To extend the work previously performed by Los Alamos, a re-examination of some of the original assumptions about the way certain systems are modeled is needed. To meet this end, the NRC recently has funded additional work at the Laboratory. The objectives of this work are (1) to identify and characterize the existing information regarding the original assumptions, (2) to determine additional research requirements, and (3) to identify the specific aspects of the existing vital area analysis and reactor sabotage vulnerability assessment procedures that should be refined.

To meet these objectives, Los Alamos first will survey and analyze the research and engineering studies that can assist in identifying the vulnerability of reactors to sabotage of the following types of equipment.

a. Individual safety-related cables in cable trays
b. Complete cable trays
c. Systems during shutdown or refueling conditions
d. Sensor systems, instrumentation, and nonsafety related control systems
e. Spatially-extended systems and components (that is, piping, electrical distribution, and HVAC systems)
f. Air systems
g. Electrical equipment by grounding or lifting of grounds

In addition, Los Alamos will identify and analyze any research that:

a. relates best-estimate analyses of plant responses to system failures to the corresponding FSAR analysis;
b. discusses effective inclusion of random events, such as anticipated transients, in fault-tree methodologies;
c. addresses possible system failures after which stable hot shutdown cannot be maintained indefinitely; and
d. considers the use of nonsafety-related equipment, unanalyzed procedures, or operator ingenuity to recover from system failures.

If issues are identified for which no research or insufficient research is being conducted to support a defensible conclusion, this situation will be reported to the NRC as early as possible. It is expected that the survey will highlight needed changes in the assumptions that will affect the results of the VAA or the reactor sabotage vulnerability assessment. These issues will be prioritized according to their anticipated effect. The required refinements to the existing procedures, including the development of modifications to the fault trees, then will be made one by one.

Before Los Alamos concludes that a particular issue has no effect on the results of the analysis, the assumption will be tested. Using the information gained from the survey, Los Alamos will model the system or component and its failure effects in a fault tree and will make a demonstration run. The results of the modified fault-tree analysis then can be compared with the original results. If the results agree, the issue will be removed from further consideration.

The survey phase of this work is expected to be completed in the spring of 1983 with the follow-on work possibly extending into 1985. The survey work was begun in August of this year. One of the major resources we have available for this effort is a computerized information retrieval system. In fact there are two such systems we are using--DOE's RECON system and the Dialog system. RECON is composed of approximately 40 individual data bases. Dialog has approximately 150 data bases in its system. Some examples of the data [illegible] important to this study are shown in Table I. To date, we have done searches on all of these data bases. We are now in the process of reviewing the results and selecting the reports from those identified that are really appropriate. The format we selected for the printout includes an abstract, which makes the report selection easier, and all the keywords and categories under which the report was filled. This is helpful in identifying words or expressions that might have been missed on the first search and allows us to go back and refine or expand our search techniques.

We are hoping that as we go through the reports we can identify authors or institutions that have done a fair amount of work on the selected topics. The next, or possibly concurrent, phase of our efforts will be to contact these people directly to be sure we have the most up-to-date information on the subject. In this regard, I would like to encourage each of you to suggest reports you know of or other data bases or people actively working in this area that may be of assistance here.

Table I

INFORMATION RETRIEVAL SYSTEMS

DOE/RECON

Energy Data Base (EDB)

Nuclear Safety Information Center (NSIC)

Nuclear Science Abstracts (NSA)

Research in Progress (RIP)

DIALOG

National Technical Information Service (NTIS)

Electric Power Database (EPRI)

Doe Energy

Smithsonian Science Information Exchange (SSIE)

Conclusion

This program will result in the original analysis assumptions either being confirmed or modified. Once the required refinements are incorporated into the VAA and reactor sabotage vulnerability assessment procedures, it is expected that the NRC will be able to use the results with greater confidence that all the vital areas and equipment have been identified. In addition, some of the unnecessary conservativeness of the analyses may be removed and thus reduce the possibility of safeguards requirements adversely affecting the safe operation of the plants.

REFERENCES

1. W. A. Bradley, R. A. Haarman, and D. G. Rose, "Keeping Reactors Safe from Sabotage," Los Alamos Science 2 (2), 120-131 (1981).

2. G. B. Varnado and R. A. Haarman, "Vital Area Analysis for Nuclear Power Plants," Los Alamos National Laboratory report LA-UR-80-2407 (August 1980).

3. G. B. Varnado and N. R. Ortiz, "Fault Tree Analysis for Vital Area Identification," Journal of the Institute of Nuclear Materials Management, Fall, 438-447 (1979).

4. J. W. Bolstad and R. A. Haarman, "Summary of Thermal-Hydraulic Calculations for a Pressurized Water Reactor," Los Alamos National Laboratory report LA-8361-MS, NUREG/CR-1480 (May 1980).

NRC FORM 335 (7 77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*	NUREG/CP-0041, Volume 3
4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*	Proceedings of the Tenth Water Reactor Safety Research Information Meeting
2. *(Leave blank)*	
3. RECIPIENT'S ACCESSION NO.	
7. AUTHOR(S)	Compiled by: Stanley A. Szawlewicz, Consultant
5. DATE REPORT COMPLETED	MONTH: December; YEAR: 1982
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*	U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research Washington, DC 20555
DATE REPORT ISSUED	MONTH: January; YEAR: 1983
6. *(Leave blank)*	
8. *(Leave blank)*	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*	Same as Item 9.
10. PROJECT/TASK/WORK UNIT NO.	
11. CONTRACT NO.	
13. TYPE OF REPORT	Compilation of Conference Papers
PERIOD COVERED *(Inclusive dates)*	October 12-15, 1982
15. SUPPLEMENTARY NOTES	
14. *(Leave blank)*	

16. ABSTRACT *(200 words or less)*

This report is a compilation of papers which were presented at the Tenth Water Reactor Safety Research Information Meeting held at the National Bureau of Standards, Gaithersburg, Maryland, October 12-15, 1982. It consists of six volumes. The papers describe recent results and planning of safety research work sponsored by the Office of Nuclear Regulatory Research, NRC. It also includes a number of invited papers on water reactor safety research prepared by the Electric Power Research Institute and various government and industry organizations from Europe and Japan.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	Unlimited
19. SECURITY CLASS *(This report)*	Unclassified
21. NO. OF PAGES	
20. SECURITY CLASS *(This page)*	Unclassified
22. PRICE	$

NRC FORM 335 (7 77)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FOURTH CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH D C
PERMIT No G67

120555078877 1 ANRHRSRVRX
US NRC
ADM DIV OF TIDC
PDR NUREG COPY
POLICY & PUBLICATNS MGT BR
W-501
WASHINGTON DC 20555