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Critical Human Factors Issues in Nuclear Power Regulation and a Recommended Comprehensive Human Factors Long-Range Plan

Critical Discussion of Human Factors Areas of Concern

Prepared by C. O. Hopkins, H. L. Snyder, H. E. Price, R. J. Hornick,
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Human Factors Society, Inc.

Prepared for
U.S. Nuclear Regulatory
Commission

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ABSTRACT

This comprehensive long-range human factors plan for nuclear reactor regulation was developed by a Study Group of the Human Factors Society, Inc. This Study Group was selected by the Executive Council of the Society to provide a balanced, experienced human factors perspective to the applications of human factors scientific and engineering knowledge to nuclear power generation.

The report is presented in three volumes. Volume 1 contains an Executive Summary of the 18-month effort and its conclusions. Volume 2 summarizes all known nuclear-related human factors activities, evaluates these activities wherever adequate information is available, and describes the recommended long-range (10-year) plan for human factors in regulation. Volume 3 elaborates upon each of the human factors issues and areas of recommended human factors involvement contained in the plan, and discusses the logic that led to the recommendations.

INTRODUCTION

As described in Volume 2 of this report, the HFS Study Group became familiar with human factors issues in nuclear power generation from the regulatory position during Task A, and during Task B confirmed its findings regarding human factors problems and issues by holding a series of meetings with industry elements, conducting appropriate analyses, and reading a very large volume of technical reports from many sources. As a result of these activities, and in the context of the systems analysis approach described in Section 2, Volume 2, a number of nuclear power areas that are impacted by human factors were defined. These areas were analyzed, and the basic human factors issues were defined. Each of these human factors problem areas was then studied, evaluated for its relation to other system problems, discussed by the Study Group to reach a consensus, of opinion, and described in writing. These written descriptions were then used to generate the Recommended Comprehensive Human Factors Plan described in Section 4 of Volume 2.

These areas of human factors concerns are described in detail, in a consistent format, in this Volume 3. For each human factors problem area, the following categories of content are discussed.

(1) REQUIREMENT

This section defines the human factors problem area, the need for concern in the NPP context, the activity that is generally required, and the significance of the problem.

(2) CONSTRAINTS

This section discusses organization, regulatory, or personnel constraints that may prevent accomplishment of the desired requirement.

(3) PRESENT STATUS

The current status of the requirement is examined, along with any present activity which may partially or fully meet the requirement. Where appropriate, the present status is divided into activities by various organizations or groups of organizations, e.g., the NRC, utilities, architect-engineers, NSSS vendors.

(4) PLANNED ACTIVITIES

Any known activities planned for future initiation, and which may contribute to meeting the requirement, are discussed.

(5) MISSING ELEMENTS

In this section, the requirements are compared with current and planned activities to determine whatever additional activities are needed. Those missing but needed activities are then summarized.

(6) TECHNICAL FEASIBILITY AND PROBLEMS

Technical issues which may hamper or limit satisfaction of the requirement are discussed.

(7) INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Interaction with and dependencies upon other requirements are discussed in this section.

(8) RECOMMENDATIONS

This section summarizes the consensus recommendations as to what should be done, when, resources required, etc. It forms the basis for the comprehensive plan of Volume 2. It is further divided into the following parts.

(a) Technical Requirements. This is a brief objective or statement of work as to what should be done.

(b) Importance. This categorizes the relative importance of work in this problem area to NPP safety. Only problems considered important enough to support or pursue are described; however, to give some sense of the relative importance among the numerous items, we have elected to categorize each recommendation as having high, medium, or low importance.

(c) Schedule. The recommended schedule for carrying out the work is suggested in terms of urgency and duration. Urgency refers to the immediacy with which the work should begin. An urgency category of "immediate" recognizes that an unprogrammed, immediate start of a new activity may require unusual justification. However, it is deemed sufficiently important to justify this action. Other urgency categories refer to the start of new activities in time bands of 1-2, 3-5, and 6-10 years.

Each project is also described in terms of its duration, the estimated time required (in years) to complete the technical objective, given appropriate personnel, facilities, and funding.

(d) Resources. Professional labor (person-years), unique facilities, and other special resources required to complete the technical requirement are estimated.

(e) Implementation. In this subsection we note any particular professional skills, experience, or constraints that apply to insuring a result of necessary quality and completeness.

(f) Dependencies. Any way in which one task or activity depends upon another activity or task is noted, particularly if a design activity depends upon previous research results.

For convenience of the reader, the human factors problem areas are grouped into six major, somewhat homogeneous categories. Section 1.0 combines nine general problem areas that typically transcend a variety of specific applications. Section 2.0 contains Human Engineering problem areas, Section 3.0 summarizes Operational Procedures issues, Section 4.0 covers Personnel and Staffing problems, and Section 5.0 summarizes Training problems. Lastly, Section 6.0 relates to human factors issues of the NRC Incident Response Plan and Facilities.

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1.0 GENERAL PROBLEM AREAS

1.1 Professional Human Factors Qualifications In Nuclear Power

REQUIREMENT

Prior to the accident at TMI-2, the nuclear power community was, for all practical purposes, unaware of the human factors discipline and its relevance to power plant safety and operations. As a result of the several TMI-2 inquiries and reports, the NRC, utilities, vendors, and others attempted to fill the human factors void, either in fact or in appearance. This void still exists in many organizations.

The demonstrated relevance of a wide variety of human factors needs requires that all organizations involved in regulation, power generation, equipment and facility design, and support have the needed human factors engineering technical abilities, just as they have requisite abilities in other engineering specialties. This need must be met, in each individual organization, by a suitably sized staff of people with the following characteristics:

(1) Suitable academic training,
(2) Experience in human factors commensurate with job responsibilities and requirements, and
(3) Appropriate organizational placement to have the necessary influence.

The suitability of academic training must not be overlooked. To function usefully in human factors design/evaluation/research activities, the professional person must have essential skills and knowledge. These skills and knowledge areas include, <u>as a minimum</u>, formal training in the behavioral sciences (e.g., experimental psychology or engineering psychology), statistical evaluation, research methods, basic engineering and mathematical techniques, and human factors systems analysis. While it is certainly critical that the human factors engineer have an understanding of the application area (e.g., nuclear system) of his skills and knowledge, it is equally (or perhaps more) important that he have the human factors knowledge to apply to that system application. Contrary to the naive, but unfortunately common, belief in many segments of the nuclear industry, human factors science and engineering is far more than "common sense," and the naive introduction of superficial knowledge by untrained persons can often do more damage than benefit to a system. Thus, the formalized academic training, usually at the M.S. level, typical of a graduate program in human factors is considered essential for persons having individual responsibility for design and evaluation.

Further, the Ph.D. level training in human factors (or a related behavioral science discipline) is strongly desirable for those persons having research responsibility for human factors/system interaction. Only at the Ph.D. level does a person usually receive adequate training in methodology and analysis techniques to assure the meaningful generation of unbiased experimental data of human performance in complex systems.

Experience in human factors work can be meaningfully substituted for formal academic training. (For example, the Human Factors Society requires three years of relevant experience beyond the bachelor's degree for member status, and two years of experience can be substituted by graduate student work in human factors.) However, experience as a general substitute for formal training often leads to an individual too specialized and too narrow. In a research setting, such an individual is quite likely to be unaware of relevant, critical research methods. In a design environment, such an individual will often be unaware of pertinent experimental literature which can be applied directly to the design problem.

One often hears of control and instrumentation engineers who consider themselves human factors engineers because "people use the controls they design." This is totally fallacious logic, because "people" use (or maintain) everything every engineer designs. That many devices are in fact poorly designed without regard for human performance limitations is ample evidence that most control and instrumentation engineers lack the fundamental knowledge of human performance data to design controls and displays in an optimum fashion for human use. Once again, we need only to point to recent examples (e.g., TMI-2, the Teneriffe crash) to demonstrate that system design by competent engineers unknowledgable of human factors issues can lead to disaster.

In this report, we frequently refer to the need for a "career human factors professional" person. Such a person would meet the qualifications noted above, in education and/or experience, commensurate with the person's assigned job responsibilities. Such a career human factors professional is to be distinguished from persons competent in other technical areas, but lacking extensive adequate training in the skills directly pertinent to and part of human factors engineering. Thus, a career human factors professional is one whose entire career (or substantial portion thereof) has been predominately in human factors education, application, or research.

Employment of competent, trained persons is a fundamental requirement to introduce the necessary technical human factors knowledge into nuclear system design/research/evaluation. However, it is <u>also</u> necessary to place those persons in an organizationally logical and authoritative position. Staff supporting roles are often ignored when engineering decisions are made. If anything has been learned from NASA and DOD system design processes, it is that the human factors function must have line approval responsibility in the design and evaluation of complex systems. Mere staff support is totally inadequate. Thus, organizational placement and authority are key corequisites with suitable training and experience.

CONSTRAINTS

Two constraints are often offered for reasons behind inadequate human factors talent. The first constraint applies only to the NRC, while the second applies to both government and industry.

In the first case, the NRC has been hampered in its effort to hire adequate human factors talent. Employment freezes, salary caps, and organizational structures have not been conducive to obtaining the most qualified people. These are meaningful constraints, to a point, and will be addressed in a separate section. It should be noted, however, that the NRC has hired, in recent months, some extremely well-trained and suitable people in the human factors field. The adequacy of these numbers of persons is a separate issue.

The second constraint, often voiced by NSSS and utility persons, is that there "are [illegible]nough human factors persons knowledgable in nuclear pow[illegible] [illegible]his is a totally illogical argument. The human factors professional can, in a matter of a few weeks or months, learn all he needs to know about nuclear power to be effective in his assigned activities. (No human factors engineer ever flew a spacecraft, but without human factors design and research, the moon would still be virgin territory!)

Further, there are adequate numbers of competent human factors engineers available for employment in the nuclear industry. The employers need only advertise the positions in suitable publications, offer competitive salaries, create meaningful and useful jobs, and show logical awareness of their needs. While utilities may be unable or unwilling to create a team of six human factors professionals for each NUREG-0700 review in the next two years, that extreme viewpoint need not interfere with one utility hiring at least one competent human factors specialist for its current activities. In short, limited availability of qualified personnel is not a real issue, merely an excuse.

PRESENT STATUS

NRC. The NRC has recently hired several competent human factors specialists in the Human Factors Safety Division. It also has open requisitions for a few more. Similarly, the Office of Research has at least two openings for human factors persons in the Human Factors Branch. It seems likely that these positions can and will be filled with qualified people.

Utilities. Most utilities seem willing to hire human factors consultants rather than to add qualified personnel to their engineering staff. Those who have hired persons directly appear unknowledgable in criteria for selection, often confusing human factors skills with such distant (and distinctively different) academic areas as cognitive psychology and instrumentation engineering. In the long run, these utilities may have no substantial, useful human factors capability, although they may meet the short-term needs through consulting contracts. This approach can someties be myopic and unnecessarily expensive.

NSSS Vendors. All of the NSSS vendors claim to be doing "human factors" work at a significant level. Two of the NSSS vendors have human factors staffs with substantial training in human factors, although one of them is assigning their human factors people to other project areas as the company reduces its involvement as a prime vendor.

Two of the NSSS vendors have attempted to mold a human factors capability from persons whose academic training and experience are largely outside the human factors field. The results of this approach can be seen in activities which are often either (1) naive replications of basic studies and principles well known to human factors specialists, or (2) incomplete design approaches caused by lack of fundamental human performance systems data.

In all four vendor cases, the level of and quality of human factors involvement appear inadequate for the magnitude and complexity of the problems.

Architect-Engineers. Based on our limited, but probably representative, sample of A/Es, there is precious little real human factors work. Some A/Es simply feel human factors issues are not their responsibility, and the resulting plant designs unfortunately reflect this philosophy. Other A/Es are paying lip service to human factors, and proudly point to a few standard human engineering design handbooks in their libraries. Only one (EBASCO) appears to be taking a somewhat systematic approach to human factors issues in the design of only one plant (Waterford). Even that small group is lacking in personnel with adequate human factors training and experience, although their systems approach has considerable merit.

Consulting Firms. The TMI-2 accident created an awareness for human factors needs and a simultaneous business opportunity for many small (and some large) consulting firms. The results of this opportunity are (1) a few competent firms now offer meaningful, competent human factors consulting to the nuclear industry, and (2) many firms offer human factors consulting, but lack adequately trained and knowledgable human factors personnel. (At least one government laboratory also fits in this latter category.) Thus, the naive buyer can be badly burned by the present situation. Unfortunately, few buyers of these services make much effort to separate the bona fide human factors specialists from the "nouveau" experts.

PLANNED ACTIVITIES

At the present time, NRC plans to add additional human factors staff, both in the Office of Research and in the Human Factors Safety Division. We are not aware of any utilities, NSSS vendors, or A/Es who are seeking to employ qualified human factors professionals.

MISSING ELEMENTS

It is apparent that the work required by NUREG-0700 and NUREG-0801 cannot be accomplished by trained human factors experts within the utilities' current employ. Further, it is extremely doubtful that the current NRC staff can adequately evaluate this work, once done. Finally, while there are some competent human factors professionals in R&D consulting firms available to perform this work, those persons will be inadequate in number to meet all these needs should they exist simultaneously.

This condition creates the opportunity for the utilities to hire inadequately trained personnel for the NUREG-0700 reviews. NUREG-0801 prescribes the abilities/training criteria needed to conduct the 0700 review. It is especially critical that NRC remain firm as to personnel qualifications in that effort. It is also, therefore, critical that NRC be beyond reproach in its selection of human factors personnel who evaluate acceptability of the evaluation teams against the 0801 criteria. Whether this judgment is made by NRC personnel or contractors to NRC is not as key as are the qualifications of those persons making the 0801 decisions. There is some reason to doubt that adequate sensitivity to this issue exists within the NRC.

Each utility could clearly benefit from some professional human factors staff. While 0700/0801 activities are not planned to last forever and are clearly safety related, competent human factors staff personnel would, in the long term, significantly improve operating efficiency through improvements in procedures, design, training, maintenance aids, etc. To view the need for human factors personnel as a transitory requirement is as short-sighted as believing that TMI-2 could never again happen (and for the same reasons).

Thus, it is clear that the NRC needs to add to its current human factors capabilities to meet the 0700/0801 needs, at the very least. Even more critically, it is clear that the utilities must develop a long-term human factors capability, which is unlikely to result from arrangements with consulting firms, over whom the utilities have little day-to-day quality control. Lastly, should new plants be designed, it is absolutely mandatory that the AEs and the NSSS vendors increase their human factors abilities and/or involvement. Likewise, such increases in human factors input is needed as current plant designs move forward in the construction cycle.

TECHNICAL FEASIBILITY AND PROBLEMS

Adequately trained and experienced human factors professionals exist. Newly trained but inexperienced individuals are being graduated from several very good human factors education programs each year. The utilities, vendors, and AEs need to look to this supply line for candidates. While the 0700/0801 requirement may create a transient excess of demand over supply (and we actually doubt this), there is no reason to believe that competent career human factors professionals cannot be hired for the long term.

INTERACTIONS WITH OTHER SYSTEM REQUIREMENTS

This issue is a fundamental one which related to all other human factors issues in nuclear power. It therefore must be considered in the context of meeting all technical requirements addressed elsewhere in this report.

RECOMMENDATIONS

o Technical Requirement

The NRC, utilities, vendors, and A/Es must realistically assess their human factors staff needs, make a much better effort to understand the meaning of and role of human factors in their organizations, and take the necessary steps to meet that need. This action is of the highest priority in the human factors area. It is not acceptable to annoint a control engineer with the title of "human factors specialist," and to assume the necessary skills can be acquired immediately by attendance in a short course or seminar. Competent career human factors professional staff must be acquired, placed into suitable organizational positions, and assigned to projects involving man-machine systems. This process should be implemented immediately.

Importance: High

Schedule: Urgency - immediately

Duration - continuing indefinitely

1.2 The NRC Organization

REQUIREMENT

It is abundantly clear that human factors issues are central to both safety and economy in nuclear power plant design and operation. Plant designers must be knowledgable in applicable human factors design technology, and the utilities must be knowledgable in human factors principles relevant to operations, maintenance, training, personnel selection and staffing, and management. Finally, to discharge its responsibility for safety regulation, the NRC must have adequate human factors skills to monitor plant design and operations, and to evaluate and support research in critical human factors areas.

Thus, the NRC must have a broad base of human factors knowledge pertinent to both operations and research. Further, the NRC must be organized in such a manner that the required human factors skills can be applied completely and efficiently in the regulatory design and research processes. In this section of the report the current NRC human factors organization and personnel will be reviewed as to their ability to meet this requirement.

CONSTRAINTS

The current civil service salary cap, the Washington, D.C. area high cost of living, and the perceived low security of government employment mitigate against hiring competent senior human factors personnel at the NRC. In addition, the existing positions are at the GS 13/14 levels, and do not offer significant promise for progression and increased responsibility. Lastly, nearly all vacant positions report to supervisors who are not professionaly trained in human factors, but who are academically trained in other areas (e.g., nuclear engineering) and have recently been "converted" to the human factors discipline. Thus, candidates may be reluctant to accept a position reporting to a non-human factors trained supervisor. (In this context, we do not feel that a human factors short course, or two, plus a few months of experience compensates for the lack of formal education in human factors.)

PRESENT STATUS

In spite of the above noted constraints, the NRC has had excellent recent success in hiring two section chiefs in the Division of Human Factors Safety, along with a couple of senior human factors specialists. More of the same is urgently needed. While we recognize that a balance is desirable between human factors specialists and nuclear engineering specialists, we also suggest that critical human factors design supervision, analysis, and research planning <u>require</u> the training and experience of human factors specialists, not recently cross-trained persons from disciplines without a quantitative human performance data base.

At the present time, openings exist in both the Office of Research, Human Factors Branch and the Division of Human Factors Safety. It appears to be extremely important that these positions be filled with trained human factors personnel. Even then, the reporting relationship to non-human-factors-trained supervisors is likely to sustain a problem which has recently been voiced repeatedly by various staff human factors professionals. Specifically, many of these knowledgable human factors persons are disappointed with and frustrated by the lack of specific technical direction. They are apparently often left to define problems, select approaches, analyze existing information, and, most critically, establish technical human factors policies, all without supervisory guidance. This inappropriate result appears to follow a lack of technical knowledge and/or confidence on the part of branch and section supervisory personnel. For this reason, it seems mandatory that technical supervisory personnel responsible for human factors work be adequately trained in human factors engineering, i.e., be career human factors professionals.

There is also significant doubt as to whether the human factors staff is adequate in size. In the Office of Research, there is only one academically trained human factors professional, and two positions await filling. Even three such persons may be inadequate to formulate problems, select and monitor contractors, and keep current on all technical issues pertinent to human factors in nuclear power.

In the Division of Human Factors Safety, about half the staff is human factors trained, and half from other disciplines. Because of this hybrid grouping, discussions are fruitful and relevant. At the same time, however, there is an obvious naivete regarding the use and benefit of classic, fundamental human factors tools. The excessive dependency on task analysis as a sole means of uncovering design problems and verifying solutions is one example. Another is the simplistic, totally unwarranted discussion and requirement dealing with "operator workload" in NUREG-0700. Competent, experienced human factors professionals would be less likely to be misled by such issues.

PLANNED ACTIVITIES

The NRC will attempt to hire approximately six to eight additional human factors engineers. In addition, a decision will hopefully be made shortly regarding the Director of the Division of Human Factors Safety. The current Deputy Director is a senior, trained human factors engineer. Should he be appointed to the Director position, the new Deputy Director is likely not to be a human factors professional, if current indications are true. Should the Deputy Director retain his current position, it is likely that the new Director will not be a human factors professional.

In this latter case, a proper interpretation may be that NRC upper management and the Commissioners do not feel the need for professional human factors leadership at either the Division or the Branch level, thereby signifying a reduced importance of human factors knowledge and training to other NRC employees and to the industry.

MISSING ELEMENTS

Following the general logic presented above, it seems critical that the NRC place competent, experienced career human factors professionals in supervisory positions. This change is needed to (1) show proper emphasis on the human factors issues in safety and operations, and (2) provide the branch-level technical authoritative guidance currently required by the human factors staff. Without such visible management changes, it will appear that the NRC is merely paying "lip service" to human factors in the wake of TMI-2, and has no long-term commitment to the application of human factors knowledge to nuclear safety.

TECHNICAL FEASIBILITY AND PROBLEMS

Technical issues appear surmountable. What remains is a direct decision to be made by NRC management regarding the importance of human factors.

INTERACTIONS WITH OTHER SYSTEM REQUIREMENTS

This is a fundamental issue that relates to all other human factors problem areas in an obvious manner.

RECOMMENDATIONS

o Technical Requirement

In the Division of Human Factors Safety, the chiefs of the branches of Human Factors Engineering, Procedures and Test Review, and Licensee Qualifications should be replaced with career human factors professionals. These changes are needed to provide detailed technical guidance to the staffs based upon the perspective of the human factors literature, experience with other (non-nuclear) systems, and information achieved from career human factors professionals in other organizations. No criticism of the current branch chiefs is intended by this recommendation. Quite to the contrary, they have done an admirable job during the past two years considering their lack of appropriate technical background. However, in spite of the experience and contribution they have made, the Study Group believes the DHFS will be better served, in the long run, by placing career human factors professionals in these positions.

Importance: High

Schedule: Urgency - immediately

Duration - indefinitely

o Technical Requirement

Either the DHFS Director or the Deputy Director must be a senior career human factors professional, and recognized as such by his peers. This career human factors professional should take a direct hand in establishing technical policy for the division and in providing technical direction to the branch chiefs.

Importance: High

Schedule: Urgency - immediately

Duration - indefinitely

o Technical Requirement

The organizational visibility of human factors in the Office of Nuclear Regulatory Research should be increased to reflect the importance and magnitude of the human factors research activities. Alternative possibilities are (a) elevation of Human Factors to division status, (b) creation of a separate "pure" human factors branch, independent of quality assurance, and (c) creation of two parallel human factors branches, one concerned with hardware and software (control/display) research and the other with personnel, training, and procedural areas. Regardless of which of these alternatives is followed, the career human factors professional staff should be greatly enlarged to provide the specialized, experienced talent needed to plan, monitor, and understand the diverse human factors research programs.

Importance: High

Schedule: Urgency - immediately

Duration - indefinitely

Resources: Additional NRC staff

1.3 System Integration

REQUIREMENT

Significant system integration during design and development is necessary for the most effective performance of humans in the operation of a large scale, complex man-machine system. A major underlying cause of many different kinds of human factors problems in the field of nuclear power generation is the inadequacy of system integration during the design and construction of nuclear plants. The most fundamental of these human factors problems is the allocation of functions to the operators.

In the absence of formal, explicit system integration, various functions tend to be assigned to operators haphazardly and on a piecemeal basis. Sometimes functions are assigned to operators as an afterthought. Sometimes they are assigned on the basis of tradition, convention, historical precedent, convenience, or caprice rather than as a result of a functional requirement.

Any unsystematic method of assigning human functions may cause, in turn, several other kinds of problems. For example, requirements for simultaneous monitoring and control of different subsystems may not be taken into consideration. Such requirements may not even be recognized until a formal training or simulation program development is initiated. Worse still, they may not be recognized until there is an operational crisis. Also, the assignment of functions to operators in the absence of significant system integration may result in needless and potentially detrimental duplication of displays and controls. On the other hand, it may result in a failure to integrate related information from different subsystems in a single display or a relatively small number of displays when such integration is necessary for effective, safe human performance.

Thus, some of the most visible effects of the lack of system integration in nuclear power plants are found in the control rooms - in poor design and arrangement of controls and displays, panels, consoles, and other physical elements of the man-machine interface. (It is important to remember that system integration does not insure good human engineering design of displays and controls. Rather, it makes sound human engineering design possible.)

The control room is the focal point for transmission of information among the subsystems that comprise a nuclear power plant. Human operators are responsible for mediating the transmission of a large amount of this information and are responsible for monitoring and evaluating essentially all of it. Specific human factors problems in the design of displays and controls that constitute the physical elements of the man-machine interface are covered in Section 2, Human Engineering, in this volume.

Less obvious to the person relatively unfamiliar with human factors are the effects of inadequate system integration on the human or behavioral elements of the man-machine system. Representative behavioral elements include development and use of procedures and operator aids, determination of selection criteria and staffing requirements, and development and implementation of training programs. These and other behavioral elements of the man-machine interface are covered in Sections 3, 4, and 5 of this volume.

The effect of lack of system integration upon behavioral elements of the man-machine interface can be illustrated by using the development and implementation of procedures as an example. When the functions to be performed by the operator have not been assigned systematically and, consequently, the displays and controls have not been designed and arranged to permit a high level of error-free and timely performance by the operator, it may be impossible to develop and implement procedures that ensure safe, effective system operation.

In summary, the level of safe, effective operation of a nuclear power plant is determined by the adequacy of the man-machine interface. This adequacy is limited by the effects of system integration upon the various elements of the interface. To some degree, enhancement of some of the elements can compensate for deficiencies in others. Indeed, tradeoffs occur routinely during system design and development of most man-machine systems. In some cases, the tradeoffs may be made between physical elements and behavioral elements of the man-machine interface. However, these are a part of the normal system integration process, with explicit identification and evaluation of consequences.

When there is little or inadequate system integration the results adversely affect both the physical and behavioral elements. This places limits on the amount of improvement in the overall man-machine interface that can be achieved after the system has been built.

System integration is recognized as being necessary and is implemented successfully in the design of large scale, complex systems in other industries. Furthermore, human factors are included as a necessary aspect of system engineering and system integration.

System integration, as it is practiced in the aerospace industry, for example, has been almost completely unknown or ignored in the nuclear power industry. We do not know of any operating nuclear power unit that was designed and constructed with a degree of system integration approaching that which is accomplished routinely in the design and development of a complex military man-machine system.

Both our study of documents and our meetings with utilities, architect-engineers, and NSSS vendors indicate that control room design most often has not been the responsibility of a single organization. Sometimes, one organization would have responsibility for actual construction of the panels and mounting of displays and controls; however, the control room design was usually the result of the wants and desires of the various organizations involved.

The Rogovin Committee (SIG) investigated thoroughly the lack of system integration relative to the design of the TMI-2 control room. Among its findings were the following:

> The actual design of the Unit 2 control room was a product of the fragmented design process already discussed above. Principal responsibility for control layout and intrumentation lay with the architect-engineer, Burns and Roe. That company, in turn, consulted with B&W, but only on specific primary system instrumentation. Further complicating matters was the fact that the TMI-2 plant and its control room were originally designed for GPU to construct a second unit at Oyster Creek, New Jersey for Jersey Central Power and Light. Only after the preliminary design work was completed was the decision made to locate the plant at Three Mile Island. (128)

In the absence of any NRC criteria, Burns and Roe was primarily sensitive to construction costs, and to what its new customer, Met Ed wanted. What Met Ed wanted was a control room like TMI-1, but the expense of completely altering the design was considerable, and the cost factor prevailed. In the end, only minimal human factors considerations came into play. Indeed, the engineer from Burns and Roe who laid out the original placement of panels for the control room was not even aware of how many people would be required to operate the plant (128, Vol 1, p. 125).

Unfortunately, system integration in the ordinarily accepted sense cannot be accomplished after a system has been designed and built. The best that can be hoped for in terms of human factors concerns is to describe the system as it exists as accurately as possible and, by means of analysis, identify the human factors functions that are implied or specified by the necessary subsystem inputs and outputs, and the relationships among the subsystems. These human factors functions may not represent an optimum assignment in terms of human capabilities and limitations, in terms of equipment and subsystem characteristics, or in terms of overall system performance criteria. But, if they are identified and explicitly recognized, there is at least some basis for considering the possibilities of limited function reallocation, control room redesign and retrofit where required, elimination or addition of controls or displays as appropriate, display and control enhancement, improvement of procedures and operator aids, relevant and effective instructional system development, and other related factors.

CONSTRAINTS

For the future, there are no engineering or technical constraints that would prevent the performance of effective system integration during the design of a nuclear power plant.

However, organizational relationships and management responsibilities different from those of the past would be required.

For the present, the constraints upon system integration are severe. It is virtually impossible to integrate to any higher degree a large complex system that, quite literally, is cast in concrete. However, as was indicated in the preceding section, it is possible to analyze the system as it exists, and to state explicitly the relationships among subsystems in terms of the human operator functions, and requirements for displays and controls. There are no engineering or technical constraints that would prevent this activity from being accomplished.

PRESENT STATUS

The present status of activity relative to the requirement for system integration, including human factors in the design of new nuclear power plants, reflects the current status of plans for new nuclear power plants in the United States. We do not know of any plans for new plants beyond these that already have construction permits or are under review for construction permits. There is no activity currently in the NRC that addresses the requirement for system integration in the design of new systems.

Most of the engineering and management personnel with whom we met, representing all sectors of the nuclear industry, showed little indication of understanding the concept of system integration - what it involves and what it implies with regard to human factors. We met with representatives of one architect engineer firm and one nuclear steam supply system vendor who stated that their organizations have both the desire and capability to perform a true system integration function in the design and construction of new power plants should an opportunity occur at some future date. It was neither our purpose, nor would it have been appropriate for us, to try to evaluate the capabilities that were intimated.

Even given the existence of some organizations that have the motivation and capability to do system integration, the NRC will have to provide the stimulus to ensure that it will happen.

With regard to the requirement for performing analyses of existing plants and the designs of plants under construction, there is current activity both in the NRC and industry. (Again, we emphasize that this is not system integration after the fact. It is not even a poor substitute or compensation for the original lack of system integration. However, under present circumstances it is a necessary prerequisite to any serious attempt to improve existing man-machine interfaces.) The "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG 0660, requires (Section I.D.1) that operating reactor licensees and applicants for operating licenses perform detailed control room design reviews.

PLANNED ACTIVITIES

We do not know of any planned programs in the NRC that address the requirement for system integration in the design of new systems. NUREG-0700 contains an appendix that presents guidelines and references for comprehensive systems analyses designed to incorporate human factors into the design and development of new control rooms. However, many of the suggested activities can be accomplished effectively only in the larger context of a system integration program.

Planned activities with regard to the requirement for performing analyses on existing plants and designs of plants under construction include the publication of "Evaluation Criteria for Detailed Control Room Design Review," Draft NUREG 0801. Although NUREG-0801 will not present specific evaluation criteria for a particular methodology (for example, task analysis), it will provide general criteria to be used by the NRC in evaluating the objectives and results that were obtained using methods such as those outlined in NUREG-0700.

MISSING ELEMENTS

The chief missing elements are a policy position and a mechanism for ensuring effective system integration during the design and development of new nuclear power plants. The legal and traditional functional relationships among the NRC, utilities, A/Es, NSSS vendors, and other elements of the nuclear induustry are vastly different from the relationships among, for example, the USAF, an airframe manufacturer, an electronics systems manufacturer, and other subcontractors. Therefore, the system integration approaches used successfully by the military services in the procurement of large, complex man-machine systems are not directly applicable to the nuclear industry. Although the NRC, as a regulatory agency for nuclear reactors, cannot enforce system integration in the same way that the USAF, as the customer for a military system, can enforce system integration throughout the contractor-subcontractors organization structure, some mechanism for ensuring system integration must be developed or adopted by the NRC.

The difference between NPP design and military aircraft design, in terms of system integration, is not due specifically to differences between DOD and a regulatory agency such as the NRC. For example, system integration is a valid, useful concept in commercial aircraft design, construction, and certification, the safe operation of which are the responsibility of the FAA, another regulatory agency.

TECHNICAL FEASIBILITY

There are no technical feasibility problems that would prevent or even hamper the carrying out of system integration during design and development of a nuclear power plant.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The very essence of system integration is interaction among all elements involved in the design of a system. This interaction leads to the identification of subsystem input and output requirements, reconciliation of these among subsystems, cost/benefit tradeoffs, interface design requirements (including the man-machine interface), etc.

RECOMMENDATIONS

A single organization, either the utility, the A/E, or the NSSS vendor, should have responsibility for performing system integration during design of a nuclear power plant. Human factors, as well as other major functional subsystems, should be explicitly included in the system integration process. The NRC should ensure that system integration is accomplished.

o Technical Requirement

Establish within NRC a system integration organization. This organization will determine policy and procedures for NRC to use in ensuring effective system integration during the design and development of nuclear power plants.

Importance: High

Schedule: Urgency - 6 - 10 years, unless new licensee requirements demand faster response.

Duration - 1 year for policy formulation; organization to continue indefinitely.

Resources: NRC staff

Implementation: Special Skills: Members of this NRC organization should include legal personnel as well as technical representatives of major subsystems, including human factors. Head of the organization should be a career system engineering professional.
Dependencies: The intitiation of this recommendation is not dependent upon accomplishment of any other task. The execution of the recommendation will require interactions with all major functions and organizations involved in the design and development of a nuclear power plant.

1.4 Safety-Related Equipment Classification

REQUIREMENT

Equipment and subsystems in nuclear power plants are classified as safety-grade class (safety-related) or non-safety-grade class (non-safety-related) for purposes of NRC licensing design review. Design requirements are applied to the safety-related equipment, but generally are not applied to the non-safety-related equipment. The items that are not classified as safety-grade do not have to be reviewed. Furthermore, the non-safety-related equipments ordinarily are not inspected or tested.

Of the several kinds of problems associated with this procedure of classifying equipments on the basis of their relationship to plant safety, two are of particular importance for human factors in the control room. One of these is concerned with the basis for classification and the other involves possible interactions between safety-related and non-safety-related equipments.

The criteria for determining in which classification an item falls are not precise. The NRC considers that an item classed as important to the safety function is credited in the analysis of a design basis event or is specified in the regulations. However, it is recognized that ambiguities sometimes result so that criteria are not adhered to rigidly. Exceptions and modifications have been made in cases where it seemed appropriate to do so.

The TMI-2 Lessons Learned Task Force (LLTF) pointed out that "the interactions between non-safety grade and safety-grade equipment are numerous, varied and complex, and have not been systematically evaluated" (77, p. 3-3). The emphasis of the LLTF was upon physical interactions between non-safety-grade and safety-grade equipment. We do not minimize the importance of the types of interactions described in NUREG-0585. However, we believe that the interaction between safety-grade and non-safety-grade equipments and subsystems is equally, if not more, critical in the man-machine (M-M) interface. The reactor operator is an integral part of each of the subsystems for which there are displays and controls in the control room.

Two salient characteristics of human behavior permit functional interactions in the man-machine interface between equipments or subsystems that otherwise might be physically and functionally separate. Firstly, human behavior is characterized by limited time-sharing capability. Secondly, it shows a high degree of sequential dependency; i.e., current behavior is influenced by preceding behavior and preceding states of the system. For the most part, this latter characteristic of human behavior is adaptive and desirable. The human is able to respond appropriately to continuously changing stimuli in the environment but still exhibit stability and continuity of behavior. Sometimes, however, the effect of preceding behavior upon current behavior may be adverse, resulting in incorrect response, delay of response, or failure to respond.

Many of the nuclear power plant subsystem functions performed by the reactor operator are sequential and not highly time critical. Therefore, it should be expected that there will be little interaction among these functions in terms of the operator's performance. On the other hand, some different subsystem functions regularly, or under emergency conditions, may have to be performed as time-shared activities. Under these conditions, the different subsystem functions may adversely and mutually influence each other in terms of increased human response time and decreased accuracy.

The reactor operator is an important functional part of both safety-grade and non-safety grade subsystems. When there is a requirement for time sharing of responses by the operator among subsystems, the distinction between "safety-related" and "non-safety-related" is not useful. Such a distinction may be detrimental if it results in inferior human engineering design of displays and controls for the non-safety-grade subsystems or if it results in the development and provision of procedures for non-safety-grade subsystems that are incomplete, confusing or hard to use. Instances of both kinds of results are common. This happens because, traditionally, the non-safety-grade equipments (and by extension, some of the elements of the man-machine interface associated with them) have not had to meet NRC design criteria.

Because of the nature of the reactor operator's participation in the functioning of multiple subsystems, and because of the human behavioral characteristics of sequential dependency and of limited time-sharing capability, it is necessary to give careful attention to human engineering design of displays and controls and to develop and provide good operator procedures regardless of the safety classification of related equipment or subsystems.

CONSTRAINTS

There are no technical constraints on the design and development of the man-machine interface in terms of the system functions allocated to the operators, rather than in terms of the traditional, but detrimental, distinction between safety-related and non-safety-related systems. The only obstacles to specification of design and development of displays, controls, procedures, and training, on the basis of human factors system requirements are the existing attitudes, traditional practices, and established organizational relationships.

PRESENT STATUS

No overall program for evaluating the consequences of the safety-related and non-safety-related equipment classification upon the design of the man-machine interface has been indentified.

It should be noted here that the "Guidelines for Control Room Reviews" (NUREG-0700) do not refer to the safety classification of the system either in the review of system functions and analysis of control room operator tasks, the control room inventory, or the control room survey. The guidelines for assessment of human engineering deficiencies state that the general procedure is one of assessing the discrepancies (individually and aggregated) for their potential plant safety consequences. Aggregate assessment (either on a system/subsystem basis, work station basis, or control-room-wide) is important to ensure that the potential safety consequences of multiple discrepancies affecting an operator task have been considered. The concern here is that aggregate discrepancies (which on an individual basis might not have safety consequences) could affect operator performance with resulting safety consequences (85, p. 4-2).

These statements reflect a recognition of the possibility of interactions between safety- and non-safety-related systems in the use of control room displays and controls. They represent an improved conceptual approach to design.

PLANNED ACTIVITIES

We have not identified any planned programs concerned with potential effects of the safety-related and non-safety-related classification upon the design and functioning of the man-machine interface.

MISSING ELEMENTS

The missing element is some kind of action by the NRC to eliminate the possibility that overall system safety may be jeopardized by a design deficiency in the man-machine interface resulting from the application of the behaviorally meaningless distinction between safety-related and non-safety-related systems.

TECHNICAL FEASIBILITY

It is technically feasible to adopt a policy that requires all design and development of the man-machine interface to be done on the basis of system requirements and accepted human factors principles and practice.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Unless the NRC develops and adopts some overall system policy other than the present one regarding safety classification, the exemption of man-machine interface requirements from the present inappropriate classification probably would result in some interactions with other system requirements. All of these interactions and their potential effects cannot be identified without detailed review.

RECOMMENDATIONS

The NRC should either (1) eliminate the classification of "safety-related" equipment, as previously recommended (96, p. 148), or (2) classify all elements of the man-machine interface as safety-related. In the event the latter alternative is selected, then the following technical requirement exists.

o Technical Requirement

Determine the system interactions and effects of classifying all elements of the man-machine interface as safety-related and implement any necessary guidelines and regulations.

Importance: High

Schedule: Urgency - immediate

Duration - 1 year

Resources: 4 person-years and NRC staff

Implementation: Career human factors professional, I&C engineer, nuclear systems engineer

Dependencies: None

1.5 Analysis and Evaluation of Operational Data

REQUIREMENT

Within the nuclear power generation industry, methods for systematically collecting data on reactor operation are necessary to detect design and operating difficulties with safety implications. Specifically, data concerning human performance and unusual or abnormal events in operating power plants would be a valuable input to the mission of all NRC offices concerned with human factors and safety. Such data could serve two primary purposes. First, the data would be useful in revealing causes of human error; and second, the data could be used to evaluate the effects of changes.

While the NRC has had event reporting systems for many years, the importance and usefulness of such data have recently been emphasized, as evidenced by the following quotations:

> ". . . improved reporting requirements are necessary to upgrade reporting to include all events having public health significance . . . (and to) . . . include reporting on systems and components that may have safety implications and not just safety related." (TMI-2 Action Plan (82, Item I.E.6)

> "We found that in the past, the NRC and the industry have done almost nothing to evaluate systematically the operation of existing reactors, pinpoint potential safety problems, and eliminate them by requiring changes in design, operator procedures, or control logic. The lack of any such comprehensive program constitutes, in our view, an unacceptable situation that compromises safety and that cannot be allowed to continue." (128, Volume 1)

CONSTRAINTS

The major constraints on current reporting systems are those common to many self-reporting systems. For example, inconsistencies across utilities in interpreting reporting guidelines, biased reporting to create an impression, differing levels of investigation, and analysis of reportable events could impact the reliability and validity of resulting findings.

PRESENT STATUS

Two basic reporting systems exist. Additional reporting/communication mechanisms operate to supplement the basic system. The Licensee Event Reporting (LER) System is operated by the NRC's Office for Analysis and Evaluation of Operational Data (AEOD). Control and operation of the voluntary Nuclear Plant Reliability Data System (NPRDS) currently rests with the American Nuclear Society but will transfer to INPO effective January 1982.

Licensee Event Reporting (LER) System

Reporting requirements for LERs are included in Title 10, CFR, Parts 20, 40, 50, 70, and 73. The requirements are detailed in the following documents:

a) Regulatory Guide 1.16, "Reporting of Operating Information," Appendix A, Technical Specifications
b) NUREG-0161, "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File."

An LER form (see Figure 1) is to be prepared for each "Reportable Occurrence" as defined in Regulatory Guide 1.16. Data from all LERs are stored in a computer by NRC.

Originally, human performance was included as a proximate cause code of either Personnel Error or Defective Procedures. Personnel Errors could be further sub-coded by the specific type of personnel that were involved in the event.

During 1981 a new method of coding and storing LER data was introduced at the NRC (107). The purpose of sequential coding was to allow coding, computer storage, search, and reconstruction of the chain of occurrences noted in LERs. Previous coding and univariate retrieval methods lost the sequential, interactive nature of occurrences within an event. With the new coding and computer retrieval procedure, occurrence sequence is preserved.

As part of the sequential coding development process, new coding tables were developed. Each LER is assigned to one (or more) "Watch List" categories. The NRC staff can then compile and examine LERs by categories of interest. Table 1 presents the categories associated with human performance.

Within the actual sequential coding of an occurence, five general cause categories (Personnel Error, Procedural Error, Design Error, Mechanistic Failure, and Other) are available. Each of these general categories has between five and fifty specific causes which are used in the coding. Table 2 shows the categories pertaining to human performance.

The LER system has been reviewed and evaluated in several reports. One of the most recent (Potash, 1981) summarizes and comments on criticisms of the LER system. Appendix A contains the full Potash analysis. The major criticisms are listed here.

1. The purpose of the reporting system has not been clearly identified.

2. Feedback provided to the licensee and relevant NRC staff is inadequate.

3. Criteria for determining reportable events are not well defined.

4. The procedures used for data collection need modification.

5. The NRC forms (and accompanying explanations) used for generating the reports need some revision.

6. Better follow-up is needed to see that corrective actions are implemented effectively.

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

EXHIBIT A

CONTROL BLOCK: 1 6 (1)

(PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 — 7 8 9 LICENSEE CODE 14 (2) 15 LICENSE NUMBER 25 (3) 26 LICENSE TYPE 30 (4) 57 CAT 58 (5)

CON'T

0 1 — 7 8 REPORT SOURCE 60 (6) 61 DOCKET NUMBER 68 (7) 69 EVENT DATE 74 (8) 75 REPORT DATE 80 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2

0 3

0 4

0 5

0 6

0 7

0 8 — 7 8 9 80

0 9 — 7 8 SYSTEM CODE 9 10 (11) CAUSE CODE 11 (12) CAUSE SUBCODE 12 (13) COMPONENT CODE 13 18 (14) COMP SUBCODE 19 (15) VALVE SUBCODE 20 (16)

(17) LER/RO REPORT NUMBER — EVENT YEAR 21 22; — 23; SEQUENTIAL REPORT NO. 24 26; / 27; OCCURRENCE CODE 28 29; REPORT TYPE 30; — 31; REVISION NO. 32

ACTION TAKEN 33 (18) FUTURE ACTION 34 (19) EFFECT ON PLANT 35 (20) SHUTDOWN METHOD 36 (21) HOURS (22) 37 40 ATTACHMENT SUBMITTED 41 (23) NPRD-4 FORM SUB. 42 (24) PRIME COMP SUPPLIER 43 (25) COMPONENT MANUFACTURER 44 47 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0

1 1

1 2

1 3

1 4 — 7 8 9 80

1 5 — 7 8 FACILITY STATUS 9 (28) % POWER 10 12 (29) OTHER STATUS (30) 13 44 METHOD OF DISCOVERY 45 (31) DISCOVERY DESCRIPTION (32) 46 80

1 6 — 7 8 ACTIVITY RELEASED 9 (33) CONTENT OF RELEASE 10 (34) AMOUNT OF ACTIVITY (35) 11 44 LOCATION OF RELEASE (36) 45 80

PERSONNEL EXPOSURES

1 7 — 7 8 NUMBER 9 11 (37) TYPE 12 (38) DESCRIPTION (39) 13 80

PERSONNEL INJURIES

1 8 — 7 8 NUMBER 9 11 (40) DESCRIPTION (41) 12 80

LOSS OF OR DAMAGE TO FACILITY (43)

1 9 — 7 8 TYPE 9 (42) DESCRIPTION 10 80

PUBLICITY

2 0 — 7 8 ISSUED 9 (44) DESCRIPTION (45) 10 68 NRC USE ONLY 69 80

NAME OF PREPARER ______ PHONE ______

Figure 1. Licensee Event Report Format

TABLE 1

Watch List Categories Related to Human Performance

A.6 Deficiencies and Human Performance Concerns (750-849)

750. Design or Analysis Deficiency or Error

760. Fabrication Deficiency or Error

770. Procedural Deficiency or Error

780. Plant Personnel Deficiency or Error

- 781. Operating error
- 782. Testing or calibrating error
- 783. Maintenance or repair error
- 784. Maladjustment
- 785. Installation error

790. Fundamental Misunderstandings

- 791. Administrative, procedural, or operating errors resulting from a fundamental misunderstanding of plant performance or safety requirements

800. Administrative Deficiency or Error

810. Security Considerations

TABLE 2

Cause of Occurence Codes Related to Human Performance

B.1 Personnel Error

OA:	Maintenance	OG:	Fabrication
OB:	Installation	OH:	Administration
OC:	Surveillance/testing	OI:	Calibration
OD:	Licensed operator	OX:	Other
OE:	Nonlicensed operator	OZ:	Unknown
OF:	Radiation protection		

B.2 Procedural Error

PA:	Maintenance	PE:	Calibration
PB:	Installation	PX:	Other
PC:	Surveillance/testing	PZ:	Unknown
PD:	Operation		

From a human performance perspective, the major problem in the LER system is the undefined purpose of gathering human performance data. If the intent of the system is to reveal the relative contribution of people-related errors or deficiencies to safety-related (reportable) events, then the current system has some value. For example, a recent analysis of the more than 3,800 1980 LERs concludes that approximately 20 percent of them are attributable to human error (135). Potash (123) also reported that 20 percent of LERs were attributed to human error during 12 months of 1977-1978. However, if the purpose of LER data is diagnostic, only gross data are available -- as evidenced by the categories shown in Table 2. Information on errors of omission vs. commission, extraneous vs. sequential acts, error location, training, etc., are not available from LERs.

There is no documented indication that LERs were to provide human error rate data. However, attempts to extract data are available (58). Since LERs report only events with "safety significance," many human errors which do not lead as directly to a reportable event are corrected before a reportable event takes place, or are not reported to protect the operators, and are not captured by the reporting system. Therefore, any attempt at human error analysis using LER data will likely be highly conservative and underestimate the actual occurrence of error.

Attempts to study the role of human performance, be it operator error, procedural inadequacy, or management/administrative quality, have relied heavily on the LER data (24, 66). If LERs are to continue to be the primary source of human performance data for the nuclear power generation industry, consideration must be given to what data are required, how to code the data to meet the information requirements, and how to collect data of more consistent and reliable quality.

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP)

This data source is also known as the 766 File or inspection reports of noncompliance. The function of this program is to follow licensee performance in various areas, e.g., plant operations, maintenance, overtime. Inspections of licensee plant and operations result in reports about noncompliance with NRC requirements.

The nature of the inspection program gives greater emphasis to management/administrative and procedural areas. Thus, it is not surprising that 67% of noncompliances fall in those domains and that "almost all noncompliances. . .were attributed to human error" (123).

Differences between the LER and SALP, or 766 File, data are summarized in Table 3.

TABLE 3

Comparison of LER and SALP (766 File) Data (From 123)

- o LERs are generated by the licensee; noncompliance data are generated by NRC inspectors.
- o The LER data base has six cause categories, only two are related to human error. The 766 File data base has 18 cause categories, 14 are related to human error.
- o Approximately 20% of the LERs submitted to the NRC were attributed to human error (personnel error/defective procedures), whereas almost all noncompliances are attributed to human error. (This is a result of the different orientations of the two data bases, described below. Also, it is probably indicative of the writer of the report, and degree of responsibility of the writer for the error.)
- o The LERs (including those attributed to human error) are oriented heavily toward operation of safety-related plant equipment and subsystems, whereas the 766 File, or noncompliance data, is concerned with plant management and procedures related to safety and covers a relatively broad range of activities. This difference has been noted by Chakoff, _et al_. (24) also.
- o The cause categories used for the two databases are not comparable, each reflecting its different orientation. For

TABLE 3 (continued)

example, the 766 File uses a cause category "inadequate management" which accounts for nearly 28% of the noncompliances. The LER data base attributes "personnel error" to licensed and senior operators, nonlicensed personnel, maintenance and repair personnel, radiation protection personnel, etc., but has no category for "management" personnel.

Both Potash (123) and Chakoff, <u>et al</u>. (1981) conclude that the 766 File data are substantially different and cannot be pooled. There is the possibility that they may augment one another, e.g., LERs may be an "alert" to noncompliance; however, the mechanism and value of such a relationship is still to be determined.

Considering the 766 File from a human performance perspective, the major problem appears to be one of role identification. The data base reveals difficulties in certain areas of nuclear power plant operation, e.g., planning, procedures, and management, but does not appear to give in-depth causal or diagnostic information. Since 766 File data report noncompliance with established requirements, utilities must correct the deficiency. Management systems tend to be unique from one utility to another and more "root cause" information may not be useful in a generic sense. The 766 File serves as a specific inspection and enforcement function and may not be an appropriate base for human performance data. If it were to be considered for expansion or elaboration as a human performance data base, consideration should be given to the source and method of data collection, the purpose(s) such data would serve, and how the data should relate to the LER data.

NUCLEAR PLANT RELIABILITY DATA SYSTEM (NPRDS)

The present NPRDS is a voluntary program for the reporting of reliability data associated with safety class 1, 2, and 1E components and systems in nuclear power plants. Participation in NPRDS has been disappointing and as of January 1, 1982 the Institute of Nuclear Power Operation (INPO) took over revitalization and operation of the system. In its current form the NPRDS provides no consistent data for the analysis of human factors problems.

SIGNIFICANT EVENT EVALUATION AND INFORMATION NETWORK (SEE-IN)

Using LERs and Monthly Outage (operating) Reports, INPO and the Nuclear Safety Analysis Center (NSAC) screen these reports for significant events. Further analysis of the significant events is performed by the contributing organizations as outlined in Table 4.

No systematic additional human factors/performance data are collected. More in-depth case study data from an analysis of a particular event may emerge if human performance is involved in the incident.

TABLE 4

Outline of SEE-IN Functions (From 72)

1. Provide basic report of plant event. (Utilities)
2. Screen event reports for significant events. (Utilities, NSAC, and INPO with vendor and contractor input)
3. Provide backup data on contributing factors and probable causes and consequences. (Utilities, NSAC, and INPO, with contractor support)
4. Perform action analysis on significant events to evaluate possible options for short-term remedies and feasible remedies which might be implemented long-term. (Utilities, NSAC, and INPO with contractor support)
5. Disseminate information to the utilities along with alerting of potential implications. (NSAC and INPO)
6. Evaluate the decision to implement desirable remedies and obtain and deploy the required resources. (Utilities)
7. Feed back implementation actions. (Utilities - review by NSAC and INPO)
8. Periodic evaluation of the effectiveness of the process including Steps 1 - 8. (INPO)

PLANNED ACTIVITIES

Early in 1981 NRC proposed establishing an Integrated Operational Experience Reporting System (IOERS). This would have incorporated the NPRDS, so an integrated NRC system became redundant. Accordingly, on September 15, 1981 the Commission approved an advanced Notice of Proposed Rulemaking informing the public that rulemaking to establish the IOERS was deferred.

At the same meeting the Commission directed staff to develop a proposed rule to modify and codify the existing Licensee Event Report (LER) reporting requirements. Specifically, the Office for Analysis and Evaluation of Operational Data (AEOD) was required to:

1. Bring a revised LER rule to the Commission before the end of 1981.

2. Coordinate closely with INPO to minimize duplication between the LER and the NPRDS systems and between subsequent NRC and INPO analysis of NPRDS data.

3. Closely monitor the progress of INPO's management of the NPRDS. After INPO takes over the system, provide the Commission with semi-annual status report on the effectiveness of INPO management of NPRDS and the responsiveness of NPRDS to NRC needs.

If any change in the human performance data collection or reporting requirements is to be made, that input had to be given to AEOD immediately to meet the Commission's end of year deadline.

MISSING ELEMENTS

A concise statement of what human factors data are required for safety, regulatory, enforcement, or research purposes is not available. Until the various human factors information/data needs are explicit, it is difficult and inefficient to determine the best mechanism for collecting those data. The various action alternatives, e.g., modify the LER system collection or coding, create a new and separate reporting system, supplement the LER with special studies, collect human factors data as part of NPRDS, etc., cannot be assessed or addressed until data purposes, scope, and uses have been clearly defined.

TECHNICAL FEASIBILITY

There are no technical problems which would preclude the changes necessary to include more human factors data.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The design of a modified data collection and analysis system would be dependent upon the needs for specific data to support human error analysis and evaluation of human engineering, procedures and operator aids, and training.

RECOMMENDATIONS

o Technical Requirement

Initiate a project to accomplish the following:

1. Establish a program/mechanism to define the existing and long-term human performance data requirements from the various perspectives with NRC and utilities: safety, regulation, enforcement, operations, research.

2. Match data needs with existing data systems.

3. Determine unmet data needs and develop candidatemethods for filling data requirements.

4. Establish a program to complete the development and implementation.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 1 year

Resources: 2 person-years

Implementation: Personnel skills required - behavioral scientist and computer data management specialist.

Dependencies: None

1.6 The Human's Role in Increasingly Automated Systems

REQUIREMENT

There are two general but opposite positions which authoritative and knowledgable technical specialists have taken with respect to increased automation. Both of those positions also exist within the NRC regarding the ideal level of automation for nuclear power process control and, thereby, the design and facilities of the control room.

The position which argues in favor of increased high levels of automation points to such apparent advantages as reduced error, reduced staffing levels, a reduction in training requirements, and lower life-cycle cost. However, opponents to greater automation argue against those claimed benefits in the following ways:

(1) While human error may be reduced by minimizing the human involvement in system operation under normal conditions, when emergency or abnormal situations occur, the operators may then make criitical errors because of a lowering in skill level by virtue of lack of practical manual involvement. (2) While it may be true that a lower staffing level may be achieved for operators, there is evidence to suggest that this is offset by a need for a larger number of technicians for maintenance and/or repair of the more automated (and more sophisticated) equipment. (3) Similarly, though some training time may be saved for operation of highly automated systems, this may be offset by (a) increased training required for the technicians needed to maintain the system and (b) increased training for the operators to handle abnormal conditions in an attempt to mitigate the loss in skill discussed in #1 above. (4) There is no real evidence to show that a highly automated system has either a greater or lower life-cycle cost than does a largely manual system.

The basic requirement then is not whether to increase (or decrease) automation but rather to determine an optimal role for the human early in the design of new systems or modifications to existing systems. This is referred to as the allocation of functions process in the discussion of The Systems Approach to Human Factors, Chapter 2 of Volume 2 of this report.

CONSTRAINTS

Even after many years of NASA, DOD, and aerospace experience with manual, semi-automatic, and highly automatic systems, there is no solid evidence to suggest an ideal level of automation for in the control of complex systems. New technology in microprocessors makes many functions easier to automate, but obviously, the human element cannot be eliminated entirely. The proper mix of human and computerized control is not expressable as a simple algorithm, and is highly system, cost, and mission dependent. The criterion must be developed to optimally allocate functions to humans and machines, or more appropriately to optimize the integration of humans and machines in performing control functions.

PRESENT STATUS

Only pro and con arguments exist regarding the level of automation desirable for nuclear power plants. Currently, control rooms are primarily manual in start-up, normal, and even in many shut-down operations. The advanced control room concepts of the NSSS vendors incorporate sophisticated and highly automated control concepts. Research at ORNL has been ongoing for over two years to define the role of the operator and more recently to develop a method for function allocation and ultimately the evaluation of the human's role in computerized designs.

PLANNED ACTIVITIES

The ORNL activities are planned to continue into FY83.

TECHNICAL FEASIBILITY

Answering the question about the ideal level of automation seems to be quite impossible. No builder of a complex aircraft, spacecraft, or power plant has intentionally built two side-by-side systems (one manual, one highly automated) in order to compare directly the error rates, manning requirements, training needs, or life-cycle costs. It would be unreasonable to do so. One might argue that the problem could be studied at least partially by means of simulators. This would be an enormous task requiring resources that do not completely exist, and would require enormous amounts of software programming, extremely meticulous experiment construction to control relevant variables, and a large pool of carefully selected test subjects, to name only a few obstacles. Nevertheless, enough is presently known about human capabilities and limitations to develop a method and criteria for the allocation of functions early in system design to determine an optimal role for the human in a specific system. See Section 2, Volume 2 for a discussion of this approach.

RECOMMENDATIONS

The Study Group does not believe it to be appropriate to suggest a research effort to define the ideal level of automation. However, a modest research program to develop design criteria for function allocation should be continued. Additionally, there are indications that some European work may eventually provide some data about human performance and automation, and we suggest that the NRC continue to monitor that work as it progresses.

1.7 Risk Analysis and Human Reliability

REQUIREMENT

WASH-1400 (73) is a pioneering document that describes a methodology and the results of an assessment of accident likelihoods in nuclear power plants. The objective of the study was to make a realistic estimate of the public risks that could be involved in potential accidents in commercial nuclear power plants, to provide perspective, and to compare those risks with non-nuclear risks currently in existence. The results were expected to be "of help in determining the future reliance by society on nuclear power as a source of electricity" (73, p. 1).

Using this methodology, the authors concluded that, in general, the likelihood of reactor accidents is much smaller than that of non-nuclear accidents having similar consequences, and that the consequences are predicted to be much smaller than was previously believed. The results also indicated, however, that human reliability was a major contributor to overall system reliability; hence, better estimates of human reliability were required to obtain more precise estimates of event probabilities and system error rates.

As pointed out in Draft NUREG/CR-1278 (98, pp. 1-2), WASH-1400 presents only summary data on human error analyses, and the reader of WASH-1400 may not appreciate how the various human error probabilities (HEPs) were developed. Thus, NUREG/CR-1278 was prepared "to utilize human reliability principles more fully in plant design and operations" (98, pp. 1-2). NUREG/CR-1278 then elaborates on these concepts, data, and calculations used in obtaining HEPs.

The approach taken in NUREG/CR-1278 appears reasonable, and the authors are most careful to point out, repeatedly, that the HEPs are often estimated based upon non-empirical data, quite frequently their own experiences. They make it abundantly clear that empirically obtained HEPs simply do not exist for most of the tasks described in the handbook. Indeed, their purpose is to present a methodology, with examples, that can incorporate HEP data, when such are improved or validated.

As a result of the approaches taken in WASH-1400 and NUREG/CR-1278, there has arisen a general requirement for the acquisition of HEPs to flesh out the methodology with valid, acceptable data. A variety of research and analysis efforts have been undertaken toward meeting this objective.

CONSTRAINTS

There are three major approaches one can take in obtaining valid estimates of HEPs (or human error rates, HERs). First, one can look to the experimental psychology literature (or equivalent) for such; unfortunately, such data simply do not exist on any range of task-dependent bases.

Second, one could obtain data from records of existing nuclear plant operations, such as LERs. This approach has been attempted at Brookhaven National Laboratory and at Iowa State University, with limited success.

Third, one can obtain data in controlled studies using a nuclear power plant simulator. Unfortunately, simulator time is heavily committed and longitudinal operations for controlled data collection are usually pre-empted by intermittent, short-term trial training problems, thereby producing suspicious HERs when any sequential dependencies are pertinent.

The latter two approaches, currently being funded, will be discussed later. However, a key consideration in assessing the utility of any of these approaches is the ultimate application of the resulting HERs, assuming valid numbers can ever be obtained. Specifically, the intent is to use such data in deriving better estimates of event probabilities and likelihoods of certain classes of accident. Presumably, these improved likelihoods would then be used to (1) redesign, improve upon, modify, or perhaps even shut down existing plants having "unacceptable" likelihoods of certain accident classes, and (2) design new plants to have acceptable accident likelihoods.

We believe that either of these applications has problems in incorporating HEP estimates. First, there are well-established, empirically determined human factors engineering principles which have been shown to minimize HEPs for most applicable tasks in both operations and maintenance. We, therefore, advocate using these well-established principles in lieu of awaiting the results of HEP validation studies. Second, the number of new plant designs (new starts) which could benefit from these data is essentially zero. With the present economy and outlook, there may be no near-term application of these HEP data to new plant design. Thirdly, and most importantly, the application of HEP data and, by similar logic, probabilistic risk analysis (PRA) models, to new plant design or to existing plant modifications, presumes the application of traditional system engineering/system integration approaches to that design and/or modification. As pointed out previously in Section 1.3, there is no evidence that a system integration approach is being, or has been, taken in any plant design or modification.

Thus, the constraints to obtaining valid HEP data are substantial, and the application of such data, under current conditions at least, is virtually impossible.

PRESENT STATUS

NRC. The NRC is currently supporting or planning to support, through Sandia Laboratories, a variety of activities dealing with human reliability. These are briefly summarized below.

Supported by RES (Division of Facility Operations), Sandia Laboratories is preparing the final version of NUREG/CR-1278, due for release in September 1982. It is expected that this revised handbook version will be similar to the draft report, perhaps have more recent HEPs from some existing work, and contain a few more examples of calculations.

Sandia is also finishing a draft version of NUREG/CR-2254, a workbook that illustrates the use of NUREG/CR-1278. The draft is about completed, and a final version is scheduled for September 1982.

To circumvent the expenses and difficulties in obtaining empirical HEP data, Sandia contracted with Decision Sciences Consortium to develop and evaluate a family of psychological scaling techniques using expert opinion to obtain HEPs. NUREG/CR-1225, a literature review on those approaches was scheduled for publication in September 1981. It has not yet been published, to the best of our knowledge. A draft NUREG on the selected procedure was scheduled for completion by March 1932.

Sandia has contracted Human Peformance Technologies, Inc. to conduct a peer review of the application of NUREG/CR-1278 to a sample of about six HEP problems. Twenty U. S. experts (10 academicians and 10 industrial scientists) and 10 foreign reliability experts participated. The responses have been submitted and data analysis is underway. A NUREG describing the effort was due in March 1982. A separate survey of some of the respondents to this exercise indicated little agreement and even less confidence in the validity of the results.

Sandia has also contracted with General Physics Corporation to develop and evaluate a human performance data bank for obtaining HEPs. A program plan draft NUREG was due in October 1981. The program began August 15, 1981. Presumably, this two-year program planning effort will be followed by a major program to develop, computerize, and make available the data bank.

Another Sandia subcontract to General Physics Corporation is evaluating the usefulness of NPP control room simulators for collecting data that can be used to estimate HEPs. These data would then be compared to the estimates in NUREG/CR-1278. The two-year program began in March 1981. Preliminary results were presented at the October 27, 1981 NRC meeting in Gaithersburg, Maryland. These results appeared limited in scope, and the statistical analyses applied to the data thus far obtained were of questionable suitability. We question whether this effort will be of much use, in part due to the apparent limited experience of the investigators in conducting and evaluating human performance experiments. Only perceptual-motor operator tasks are involved in this first two-year effort. A follow-on simulator program, scheduled to begin in April 1983, will include cognitive-based tasks. That work is also scheduled to be conducted by General Physics Corporation.

In addition to the above RES programs, contracted through Sandia Laboratories, the Division of Risk Analysis is heavily involved in the IREP (Interim Reliability Evaluation Program). Conducted by the NRC, this program emphasizes methodology development and four trial applications. It will be followed by a National Reliability Evaluation Program (NREP) on all licensed reactors, to be conducted by the reactor owners. These programs are reliability analysis, not risk analysis, programs. They recognize that human errors contribute an estimated 50% of all risk, however, and therefore demand good HEP data. Sandia Laboratories human factors personnel are supporting these efforts, but the intent is to apply existing data (e.g., NUREG/CR-1278) wherever possible to reduce the need for qualified human factors experts when all operating reactors are evaluated under the NREP. The Office of Research is funding, through ORNL, a subcontract to Applied Psychological Services, Inc. to develop and validate maintenance-related human error models, risk prediction techniques, and means of applying these models to nuclear plant risk assessment. In FY81, the contractor and ORNL were to have surveyed existing models and methodologies, initiated a task analysis of NPP maintenance activities, and developed a comprehensive plan for future work in this area. The subcontract to Applied Psychological Services, Inc. was scheduled to terminate in December 1981.

Lastl , Brookhaven National Laboratory is conducting statistical analyses of HEPs, based upon data from LERs. Modelling activities thus far have compared HEPs for valves and pumps with estimated HEPs from NUREG/CR-1278. The differences between the early BNL results and the NUREG/CR-1278 HEP estimates range from quite similar (factor of 1.87) to somewhat discrepant (factor of 9). This work is continuing.

As a related, but minor effort, BNL is evaluating the feasibility of methods of coding, storing, and retrieving human error data from LERs. This study should be completed in FY82.

Utilities. Our meetings with utilities revealed that they have little interest in, and make no use of, PRA or HEPs in design, operations, or maintenance. Further, their staffs do little to track NRC work in this area, although they will ultimately have to get involved in the NREP.

NSSS Vendors. The NSSS vendors similarly have little familiarity with and interest in PRA and HEPs.

Architect-Engineers. The architect-engineers we have talked with have not considered PRA or HEPs as a tool in plant design. This is to be expected inasmuch as they apparently do not take a system engineering approach to plant design.

Consulting Firms. Consulting firms, not surprisingly, have been heavily involved in human reliability research, but it must be noted that their interests are largely in the generation of HEP data, conduct of simulator studies, and creation/analyses of human performance data banks. Since their interests are largely in science and scientific support, rather than in the utility of these data for plant design and modification, such involvement is to be expected.

PLANNED ACTIVITIES

The RES-sponsored efforts at Sandia Laboratories, BNL, and ORNL are scheduled to continue as described above and in Section 3 of Volume 2 of this report.

MISSING ELEMENTS

Under the assumption that HEP research applied to NPPs should be continued, the existing and planned programs are adequate to cover all possible sources of HEP data and data generation. No addition work appears warranted.

TECHNICAL FEASIBILITY

The enormous variety of human tasks in NPP operations and maintenance tends to defy the creation of a valid HEP data base. Further, the heavy dependence on performance shaping factors (PSFs) causes additional complexity in empirical data base generation or construction. Thus, the feasibility of this overall effort, to an acceptably valid level, is questionable at best. Without question, however, is the need for competent, experienced career human factors professionals in directing and conducting this research, supported by able computer programmers, statisticians, and engineers. There is considerable doubt, in our opinion, as to the adequacy of training and experience of some organizations and personnel currectly involved in this work.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

A valid HEP data base, if it existed, could be helpful in operator selection, performance and specification, and training program development. However, those activities must proceed at a more rapid rate than will be possible if they depend on the valid results of HEP research.

RECOMMENDATIONS

The preceding discussion can be summarized as follows:

(1) There is scant empirical basis for HEP provided in NUREG/CR-1278. Improvement of this deficiency would be tremendously expensive, and validation of the resulting HEP is, while theoretically possible, practically infeasible.

(2) While the human reliability estimation process is reasonable and logical for a well-trained analyst, the process seduces the user into believing that resultant probability values are valid, in spite of the nonvalidated input HEPs.

(3) The present state (and predicted future state of NPP design and modification) disregards good system engineering/system integration concepts, and therefore cannot make design use of PRA or HEPs.

(4) If proven system design techniques associated with human engineering of workstations, personnel selection, operator procedures and aids, and training systems are applied to NPP operations and maintenance, then HEPs will be minimized and human operator performance will be maximized. Under these conditions, no further improvements are likely and predictions resulting from HEPs become superfluous, even if generated from an improved, valid HEP data base. (The potential argument that PRA might distinguish between the relative merits of two different designs, each based upon the same, proven human factors design techniques, is fallacious, for the PRA/HEP data base will always be less valid (more "noisy") than will tried and proven design concepts based upon empirical human performance data).

Accordingly, it is recommended that the current high level of research activity in HEPs be greatly reduced to only an awareness of and very minor support of performance measurement activities.

o Technical Requirement

Maintain awareness of other tasks (e.g., Section 1.8, this volume) that might provide useful, empirical data on HEPs. Attempt to shape those tasks, where feasible, such that valid HEP data can be obtained at little or no additional cost or effort.

This effort can be an NRC staff (RES) function, and requires no substantial resources.

1.8 Evaluation Criteria

GENERAL SYSTEM REQUIREMENTS

Possibly the most general requirement related to attempts by the NRC and the Nuclear Power Industry to improve safety and efficiency of powerplant operations is the need for objective evaluative criteria for use in validating proposed or mandated changes. This is equally true whether the need is to evaluate control room "enhancements," operator examination standards, purported improvements in training programs, specification of simulator features, assessment of operating and maintenance procedures, improvements in personnel selection, or recommended work/rest cycles. What is lacking in all instances are objective measures of effectiveness (criteria) against which to validate the system improvements. It is also generally true that the data required to support evaluation are not readily available (see Section 1.5).

CONSTRAINTS

The principal constraint in the area of evaluation criteria is associated with the difficulty of objectively assessing all important facets of operator and maintainer performance. There are two problems: (1) collecting performance data should not interfere with normal operation or change the way in which personnel typically perform; and (2) collecting a representative sample of behaviors reflecting the full scope of critical skills, knowledge, and cognitive processes can be an enormous data collection/reduction task.

PRESENT STATUS IN MEETING REQUIREMENTS

The need for objective performance criteria is apparent in many of the differences of opinion expressed between industry and the NRC. For example, industry is critical of the NRC for having ill-defined acceptance criteria with respect to man-machine interface design, function allocation, and other human engineering requirements associated with NUREG-0700 and NUREG-0801. The NRC is criticized for being both too general and too prescriptive at the same time. There appears to be widespread industry concern over the possibility that the criteria applied in determining whether or not the guidelines are met will be elusive and subject to change at the last moment, or will be subject to the biases of the evaluator.

A more fundamental concern has to do with how it can be shown that a change in design truly improves reliability, safety, or performance. For example, it is suggested by some industry representatives that certain add-on features purportedly aimed at increasing safety of operations may actually have adverse effects because of their impact on maintainability. There is general concern in industry with how much real benefit will come from mandated control room changes. The perceived enormous documentation effort required by NUREG-0700 is viewed as unlikely to have commensurate benefits. The view is held by some that the NRC is concerned only with safety and not with broader criteria of performance effectiveness. Industry feels the concern must be with both.

To date, objective performance criteria have not been employed in any systematic way to validate changes in control panel designs and information displays, although currently planned minor and major control room modifications clearly necessitate measures of performance <u>under stress</u>. NUREG-0801 provides numerous examples of dependency on subjective assessments. Yet, considerable design change is already taking place. In some cases, display changes are being "validated" against operator preference criteria though it is not clear that the procedures used would stand up under more rigorous evaluation methods. Nevertheless, this method seems preferable to selecting formats designed by vendors who, some industry representatives feel, fail to understand the operator's problems adequately.

Similar concerns have been expressed with respect to the validation of procedures. The "adequacy" of a procedure is difficult to define. Again, subjective judgment appears to be the only available criterion. How does the inspector assure himself that a given procedure "works" during a walk-through? Industry points out that the methods of carrying out procedures can vary considerably in practice. What acceptance criteria can the inspector justifiably apply? One industry observer notes that many routine procedures are never validated (NRC's emphasis is on emergency procedures) and that there is a big difference between what is written and what is actually done. It is similarly felt that job performance aids are rarely validated. These concerns also interact with the issue of "strict compliance" with procedures.

Critics of training, as carried out by industry, also focus on the criterion problem. It is pointed out that because well-defined control room performance objectives are lacking, industry doesn't really know what it is training for. Industry (through INPO) is trying to address this problem by developing training standards and model training programs focused on such criteria as task difficulty and frequency of performance. These efforts should be useful, but the requirement for objective performance measurement remains. The vendors of simulators emphasize that the technology exists for automatic performance evaluation in the simulator but that the customers have lacked sophistication in asking for features (obviously involving significant investments) that would enhance performance evaluation and feedback.

The absence of objective performance criteria also limits improvement of personnel selection methods and development of more operationally relevant licensing standards. Industry has made substantial investments in studies aimed at improved personnel selection but these studies have been limited by their dependency upon subjective evaluations as criteria against which to validate new test variables. While it is likely that performance ratings (properly structured and carefully collected) reflect some important aspects of an ultimate criterion of performance, the tenuous relationship often observed in other settings between supervisory ratings and objective measures of technical performance raises considerable doubt that selection procedures will be optimal unless objective performance measures are also included in the composite criterion. Similar observations can obviously be made about the problem of validating licensing examinations. This latter problem appears well-recognized within the research arm of the NRC though possibly less so within other groups, although the DHFS has a project beginning in FY82 to validate the current examination.

Industry has not been unaware of the need for objective performance measures, however. In a sizable research effort, EPRI has sponsored the development of a computerized Performance Measurement System (PMS) as an adjunct to simulator use. This project has as its objective the development of a data base for operator reliability evaluations, for evaluation of control room design and procedures, for enhancement of training, and for the generation of data to support operator selection research. The NRC is making use of the PMS in studies of "Safety Related Operator Actions" being carried out by ORNL.

The PMS provides measurements of how the operator responds to plant indicators by recording what switch and control manipulations are made in response to a problem scenario. Data are recorded on magnetic tape containing indications of all control room gauges, annunciator lights, and switch and knob positions, with time to one-second accuracy. When any change occurs, a data record is written. The resulting data form a sequence of "snap shots" of the simulator, each containing the status of every light, meter, switch and knob at each point in time. By evaluation of a series of data records, operator time response, errors and continuous control behavior can be evaluated (60).

The key to the success of this approach depends upon the ability to specify the responses of an "ideal" operator to the problem scenario. For example, it should be effective in determining how well operators follow specified operating and emergency procedures, but a complication arises whenever there is more than one correct path of operator actions that will deliver the plant to the desired condition.

The PMS is also capable of objectively recording certain continuous varibles associated with controlling plant states in a stable manner and staying within operating limits. It is noted (60) that absolute criteria for stable performance within technical specification limits are subject to debate. However, the system is certainly capable of capturing examples of highly unstable performance.

To date, this program has demonstrated that it is possible to automatically record the status of the plant and associated operator control manipulations. Unfortunately, a large part of the truly significant behavior of the operator, including the basis for most decision processes, cannot be directly captured and must be inferred from the data in post-exercise analyses. Nevertheless, the PMS is a significant step in the direction of much needed objective performance measurement.

The NRC has established a task with Oak Ridge National Laboratory to collect and evaluate data on nuclear power plant operator performance relevant to safety-related operator actions. The program, which uses the PMS system, is intended to provide information that will assist in the assessment of perforance in responding to emergency conditions. A significant part of this effort is aimed at objective measurement of operator response time and error rate as a function of various independent variables, including performance shaping factors, plant characteristics, and accident sequences that influence those criteria. Performance shaping factors that have a significant impact on operator performance are to be identified together with simulator experiments that can be used to validate and quantify the effects of those factors.

NRC has a contract with Sandia National Laboratory for human performance modeling of nuclear power plant operations. The thrust of this effort is the estimation of human error probability for use in reliability and risk assessment analyses. Some elements of this effort clearly relate to evaluation criteria including (1) evaluation of human error probabilities in relation to error rates being developed from LER reports and in simulator studies at ORNL; (2) development of a program for establishing a human performance data bank; and (3) design of a program and methods to collect human performance data on power plant simulators.

The Task Group on Human Performance Evaluation of the IEEE Human Factors Working Group SC 5.5 has conducted a survey of models and data bases relating to human performance in nuclear power plants. Data and models from academic, aerospace, military, and power industry sources were reviewed. It was concluded that, although a large amount of data exists, its applicability to nuclear power plants may be quite limited. In some cases the tasks are very different, in some the data base is quite small, in some it contains only anthropometric information, and in some it reflects performance only in simulators which, it is felt, may pose a problem of "transferability" of the data. It is concluded that further investigation will be needed to evaluate how applicable and sufficient the available data and models are to the nuclear power industry. Further inputs are being sought in an iterative program aimed at refining the survey information.

Modeling work in the area of maintenance performance is being supported through Oak Ridge National Laboratory with the objectives of (1) identifying information needs concerning maintenance-related risk prediction; (2) documenting available quantitative methods for predicting human performance of maintenance tasks; (3) conducting, as necessary, a task analysis of nuclear power plant maintenance activities, including data relating to time, accuracy, and potential error; and (4) developing a program plan for the development of a model, data base, and methodology. A planned maintenance task analysis by INPO during FY82-83 will be monitored to determine the extent to which task analysis will be necessary under this program. Similarly, INPO/DOE task analyses of support personnel whose activities are closely related to plant safety are being monitored and additional analyses will be performed if deemed necessary.

In addition, Battelle Pacific Northwest Laboratories has been tasked for support in the area of human factors and procedures guidelines in relation to maintenance, with the objective of generating guidelines and criteria in support of the licensing process. This work will focus on (1) identification of equipment failures and nonavailability that could be attributed to personnel errors; (2) assessment of "dominant causes" of maintenance problems and errors; (3) investigation of the processes by which maintenance procedures are prepared; (4) analysis of likelihood that maintenance personnel may induce common mode failures; and (5) study of administrative controls related to maintenance. Though this work does not specifically address the issue of a criterion of maintenance procedure effectiveness, it clearly should generate much useful information related to that objective.

PLANNED ACTIVITIES

The planned NRC activity most fundamentally related to the need for evaluation criteria is a contractual effort that started in December 1981 for a control room operator task analysis. If successful, this effort will provide not only a comprehensive inventory of operator tasks, but will also identify cognitive requirements, control requirements, information requirements, likely response times, and a number of knowledge and training requirements associated with normal, off-normal, and emergency procedures. These data are fundamental to a host of validation programs and in some cases can be used directly as validation data. For example, the task analysis should provide an objective basis against which to judge the comprehensiveness of training programs and examination procedures.

With this exception, and the continuing work at ORNL on the measurement of safety-related operator actions, there do not appear to be any NRC activities directly planned to attack the criterion problem. Rather, it appears that suitable evaluative criteria will have to be generated in conjunction with other efforts (e.g., control room modifications, SPDS designs, the development of plant drills, the validation of operator qualification examinations, the validation of procedures, assessment of work schedules, etc.). There are projects involving the development of evaluative criteria at INEL, LLL, and ORNL. Clearly, not all evaluative objectives can be served by the same criterion measures. However, the most general need appears to be for objective measures of control room operator performance that can be related back to the higher level cognitive and decision making requirements of the operator's jobs at all operational levels (e.g., RO, SRO, and Shift Supervisor). As noted earlier, some of these programs not only call for performance measurement, but under conditions of operator stress.

MISSING ELEMENTS

As noted throughout this report, objective criteria against which to validate virtually all regulatory decisions can, at the present time, be considered missing elements. This conclusion applies equally to plant operation and plant maintenance.

TECHNICAL FEASIBILITY AND PROBLEMS

There are different technical problems associated with different classes of criterion measures. In the area of control room operator performance, a practical constraint stems from the very large universe of operator behaviors and control room designs that should be sampled in the development of any comprehensive performance criterion. Associated with this, as previously mentioned, is the problem of translating objectively captured response data into dimensions of performance that can be related to the procedures or changes that are being evaluated. The investment in both simulator (or control room) time and the time of qualified operators is likely to be substantial. Many important evaluations cannot be carried out in the operating plant, thereby necessitating dependence on simulators which at the present time can duplicate some, but not all, of the important operational problems of interest. There may be questions concerning the degree of "stress" that occurs in simulators as opposed to actual plant operations.

Obtaining reliable criterion measures can be a significant technical problem. In addition to comprehensiveness, any criterion measure of performance used for evaluative purposes must, of course, be reliable (i.e., repeated measurements must generate similar outcomes with respect to the effects of independent variables). Meeting this measurement requirement usually implies a fairly lengthy performance test and sometimes necessitates measuring a large number of comparatively small segments of performance as opposed to a more global index reflecting more general outcomes. These are difficult problems to solve in any operational context and they may prove technically difficult in the context of power plant operations.

The extent to which similar technical problems will exist in developing evaluative criteria of maintenance performance is not presently clear. This is an area that deserves attention during currently supported efforts.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The need for objective evaluative criteria interacts with every other system requirement. A specific example where interactions are not being well-coordinated is in the Emergency Response to Utilities (NUREG-0696). The Office of Inspection and Enforcement is developing several criteria for ERFs (NUREG-0814) while the Division of Human Factors Safety in NRR is developing acceptance criteria for the Safety Parameter Display System (NUREG-0835), which is a part of the ERF.

RECOMMENDATIONS

1. Research should be conducted to identify objective performance criteria by which to evaluate proposed changes to

design, procedures, training, personnel selection, qualification standards, work schedules, management practices, and so forth.

2. To the extent possible, the criterion measures should reflect performance on a represenetative subset of the universe of actual operational (or maintenance) tasks. This subset should be identified. The feasibility of defining a common set of performance criteria based on this subset of tasks that will serve a diversity of evaluation needs should be determined. Prior work at various laboratories should be evaluated with respect to scope and measurement properties.

3. The practicality, cost, and technical feasibility of employing unobtrusive data collection methods relating to the evaluative criteria should be determined for: (a) actual operating plants; (b) full-scale simulators; and (c) part-task simulators.

4. Research should also be conducted to define useful secondary criteria such as progress through training, licensing examination scores, supervisory ratings on various dimensions of performance, frequency of involvement in "events" or critical incidents, and turnover rate.

5. Research should be directed toward the development of a comprehensive criterion of performance effectiveness for operator and maintenance personnel. This criterion should reflect not only technical competence but other job relevant considerations such as performance under stress.

Importance: High

Schedule: Immediate start on evaluative criteria that will be required for evaluating near-term changes or developments. More general effort starting in 1-2 years.

Resources: 5 professional person-years per year. Access to control rooms; unrestricted use of simulators.

Implementation: Expertise in human performance measurement, statistical methodology, plant operations, plant maintenance, application of computers to performance measurement.

Dependencies: Completion of NRC task analysis desirable but not essential.

1.9 System Engineering of the Regulatory Requirements

The NRC has issued many new requirements since TMI-2 that impact human factors; and many existing requirements have been updated and reissued to reflect new policy. Essentially all of the items in Chapter I of NUREG-0660 (and clarifications in NUREG-0737) are items which will affect changes in some human factors areas including personnel and staffing, training, procedures, and control room design. All changes should, therefore, be carefully planned and integrated in order to obtain the optimum benefit from such changes and to maintain some overall human factors integrity.

REQUIREMENT

A common concern emerging from the nuclear industry is the lack of integration of NRC requirements which impact human factors and control room instrumentation. The Study Group's review of relevant documents supports this concern, which may be identified as a lack of "systems engineering of the regulatory requirements." Because some of these requirements are currently being implemented by the utilities and other related requirements are imminent, a clarification which integrates all of the human factors related requirements should be issued. This clarification should encompass at least the following requirements which impact control room design and personnel performance and are frequently dependent and possibly conflicting:

- Development of emergency operating procedures (reference Item I.C. 1(3) and Item I.C. 9 of NUREG-0660; also NUREG-0799).
- Development of a safety parameter display system (reference Item I.D. 2 of NUREG-0660; also NUREG-0696 and NUREG-0835).
- Upgrading of Emergency Response facilities (reference Item III.A.1.2 of NUREG-0660; also NUREG-0696 and NUREG-0814).
- Development of improved control room instrumentation (reference Item I.D.5 of NUREG-0660; also NUREG-0700 and NUREG-0801).
- Changes in requirements for training and staffing (reference Items I.A.1 and I.A.2 of NUREG-0660).
- Instrumentation for light-water-cooled nuclear power plants to assess plant and environs conditions during and following an accident (reference Regulatory Guide 1.97, Revision 2).
- Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (reference Regulatory Guide 1.47).

- Meteorological Measurements Programs in Support of Nuclear Power Plants (reference Regulatory Guide 1.23).

These items have in common (1) a need for some type of control room personnel task analysis and (2) instrumentation and control requirements during both normal and emergency conditions. The implementation of these requirements ranges over several years.

This lack of integration of requirements creates problems which jeopardize the system approach necessary to implement these requirements and thereby ensure maximum impact on reducing human error.

CONSTRAINTS

No constraints can be identified which would preclude integration of all the requirements identified above, at least from the human factors point of view.

PRESENT STATUS

The NRC has no identifiable effort at present to integrate the various regulatory requirements. NUREG-0801 has a very simplified chart. It does indicate a general relationship between a few of the requirements, but is completely inadequate for any planning.

Industry has not prepared and published any integrated plan known to this Study Group. It is known that the Atomic Industry Forum has sent a letter to the NRC expressing concern about the lack of integration of requirements affecting control room design.

The utilities are currently planning and putting together project teams to accomplish all of the items listed earlier. However, there is a reluctance to proceed based on the lack of integration with respect to sequencing specific actions and the dates of submission and implementation.

PLANNED ACTIVITIES

As this report goes to press, the DEDROGR has submitted a plan addressing some of these issues (157). This plan is discussed in Volume 2 of this report, which points out its shortcomings in attempting to integrate the several human factors issues.

MISSING ELEMENTS

Other requirements beyond those identified above may also impact human factors and necessitate integration, but the identification of these is beyond the scope of this project.

TECHNICAL FEASIBILITY AND PROBLEMS

Not applicable to this concern.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

As stated above, this is the focus of the concern about the lack of system engineering of regulatory requirements.

RECOMMENDATIONS

- Technical Requirement

The NRC should issue a clarification as soon as possible which will integrate the individual activities of these major efforts and will account for the dependencies, conflicts, and compliance dates. Such a clarification would not only strengthen the end results but would also tend to reduce the variance in methods of approach and levels of effort contemplated by the utilities.

Importance: High

Schedule: Urgency - immediate

Duration - no more than 3 months, including publication

Resources: NRC staff time

Implementation: NRC staff

Dependencies: None

2.0 HUMAN ENGINEERING

INTRODUCTION

Proper human engineering practice has not existed within the nuclear industry, nor within the NRC. Even as a result of Three Mile Island it is not obvious that consistently valid and effective human engineering practices will occur, in spite of some good intentions by some people within key elements of the nuclear industry and the NRC. Perhaps this report will increase the probability that better human engineering practice will be more consistently experienced in and applied to the design process. However, that depends ultimately on the implementation of the research/development issues addressed in these Volumes and on the level of commitment given to this need by management of the NRC, utilities, NSSS vendors, and A/Es.

The thrust of this report is to recommend a human factors plan for the future. While there is no need to restate the number and types of human factors deficiencies of power plants in great detail, it is desirable to identify the data that were available to the nuclear community so as to alert the reader that there were substantive warnings by qualified scientists and observers which were not heeded by any segment of the nuclear community -- at least not by any responsible or influential decision-making segment.

There is no excuse for the nearly total avoidance of human engineering application in the nuclear industry. The human factors discipline has existed since World War II; design standards have been in aerospace circulation for decades; studies have been conducted within the industry which identified serious problems as well as recommendations for resolving them; and some knowledgable people have urged attention to this area affecting nuclear safety. The Kemeny Report, as well as the Rogovin Report, brought the results of disregard for human factors well into focus. Still, the expressions of some people within the industry today demonstrate a distant permissiveness to an outright pejorative attitude which strongly suggests that successful implementation of good human engineering design will be a distinct challenge for those who wish to pursue it.

Countering that pessimism is a hopeful attitude which exists on this Study Group. There are expressions of key people in the nuclear power community which pay, at least, "lip service" to human engineering concerns. Further, many management personnel do have a deep conviction that good human engineering is necessary for safety of plant operations. And, still further, some elements of the nuclear power industry are in the process of hiring human engineering consultants, if not full time human factors specialists, within their own companies.

A qualitative human engineering functional difference exists between control rooms and the other plant areas. The control room primarily serves the operational control of the nuclear energy process; it must be serviced and maintained on a scheduled or irregular basis -- but that is a supportive function. In other plant areas, the primary manual activity is that of maintenance, with very limited control functions being accommodated secondarily.

It is undertandable that, since TMI-2, the primary interest of the nuclear community has been in the control room since serious errors and design deficiencies were discovered there following the accident. However, there is no question that failures in proper maintenance activities led to serious events in the TMI-2 control room. Consequently, the human engineering part of this long range plan focuses not only on the control room operational design features, but also on the maintainability of the entire plant.

Significant effort pertinent to human engineering prior to TMI-2 is seen in WASH-1250, WASH-1400 (NUREG-75/014) and EPRI NP-39. In the same vein since TMI-2 are EPRI NP-1118, EPRI NP-1567, NUREG/CR-1278, and NUREG/CR-0400 reports. It is worth examining these briefly for some of their general comments and observations before exploring specific human engineering issues in subsequent sections. In those sections, additional studies and reports of a human engineering nature will be identified.

WASH-1250 (140) addressed the matters of safe operation, redundancy, inspectability, testability prior to acceptance, and instrumentation and controls. WASH-1400 (73), sponsored initially by the AEC and completed under the aegis of the NRC in 1975, suggested that the risks of accidents in nuclear power plants were "comparatively small" to non-nuclear accidents having similar consequences, but it made no judgment as to what level of risk is acceptable to society in general. It did discuss human error, testing and maintenance, and hardware design problems. Those two reports did identify concerns, but did so within the general premise of low-risk probabilities. However, WASH-1400 (p. 162) stated that "human failure probabilities can be quite high when compared to component failure probabilities." It further criticized the design of controls and displays and their functional arrangement on console panels as deviant from the human factors standards commonly adopted by the aerospace industry, as well as by other commercial industries.

As a result, an EPRI Task Force initiated a human engineering analysis and review of operational control rooms. In that review, Project 501, conducted by Lockheed Missiles and Space Company, control room design problems were identified and described in great detail pertinent to workspace, arrangement, control board configuration, traffic flows, visual accessibility, and environmental factors (such as illumination and noise). Included were concerns with manning, functional grouping, mirror-imaging, annunciator/alarm systems, and others. In addition, that study (32, completed in 1976) argued for future research efforts related to those issues, as well as for the adoption of realistic standards, design guides, better alarm/annunciator systems, decision-aiding and maintenance features in order to reduce error potential and selection and training requirements.

In its Summary Report for EPRI NP-1118, Project 501-3, the research team concluded forcefully that the technology of human factors has not been "generally or consistently applied to the design of nuclear power plant operational work spaces." Here again, the problems in design of control rooms are well documented, with extensive use of photographs depicting the problems. Significant in its "Summary and Conclusions" was the statement ". . .there is no consistent, uniform, or formal concern for the human factors engineering aspects of control room design." Further:

> "There is no standard pattern of relationships in the design process between participants; namely, the client, A/E, NSSS vendor and the NRC. This is not to say that human factors concerns are totally ignored. Rather, the responsibility for the man-machine interface is assumed to be everyone's concern. No one individual or group is exclusively responsible for the man-machine interface. . . . Also, there are no standards, guides. . . for measuring the adequacy of new design from the human factors standpoint." (38)

This is undoubtedly the most significant conclusion of the EPRI NP-1118 Report -- the lack of an integrated systems and/or human factors approach to control room design.

A significant review of power plant maintainability is found in EPRI NP-1567 (Project 1126). Though completed after TMI-2, the 1126 Research Project was initiated in late 1977 as a result of the EPRI NP-309 study in which note was taken of plant outages which had been caused or prolonged by human engineering problems associated with maintenance. In its summary report, the review team observed that:

> "It was evident from discussions with design organizations, utility personnel at all levels, and NRC representatives that no one group or agency has assumed or has been assigned the responsibility to ensure that power plants are designed for safe and effective maintenance activities." (39, p. 1-45).

The study identified and photographically documented major human engineering problems regarding access for maintenance, labeling, coding, communications, overcrowded workshops, equipment mobility, and poor environmental factors. It also identified non-human engineering, but distinct human factors problems, in maintenance training, inadequacy of procedures and manuals, job performance aids, and inter-organizational coordination.

NUREG/CR-1278 (98) and NUREG/CR-0400 (95) are concerned with statistical data regarding human reliability and risk assesment, respectively. The emphasis in NUREG/CR-0400 is methodological with respect to probability analysis statistics and consequences of an accident. However, a three-page topic on human factors pointed to the need for better human data in assessing human performance in complex systems. In NUREG/CR-1278, the authors make several cogent points:

1. No systematic human engineering technology is incorporated in power plant design of man-machine interfaces.
2. Violations of conventional human factors practices are the general rule rather than being exceptions.
3. The incorporation of good human engineering in the design of plants could result in "substantial improvements" in human reliability.

It is clear, therefore, that a well-documented history of poor human engineering design of nuclear power plants exists. From such documentation and from the Study Group's intensive contacts with the NRC, utilities, NSSS vendors, and the A/Es, the specific issues which require attention have been identified and are presented in the following sections.

2.1 Design Induced Error

REQUIREMENTS

The most important concern about safe operations in nuclear power plants is that of human error. From a human factors point of view, human error can result from many causes. Detrimental environmental factors such as excessive noise, temperature extremes, inadequate lighting or illumination glare, poor ventilation, etc. are all known to affect human performance negatively -- and all exist in nuclear power plants. Inadequate training results in human error, and training programs in the nuclear industry have been found wanting. Poorly prepared or inaccurate procedural manuals are another source of error, and manuals are found to be inaccurate and/or difficult to use. Fatigue, boredom, and stress are personal factors producing human error, and all exist in a nuclear power plant. All of these factors are found in nuclear plants, but are differentially operant in the control room versus other plant areas. For example, ambient illumination may be too low for good maintenance in an auxilliary building; and, it may be bright enough in a control room, but glare may be reflected on the surface of indicators from the light sources. Fatigue mav plague maintenance personnel from excessive work hours, or may be boredom-related in a control room due to long periods of monitoring.

In a sense, all of the factors mentioned above are design-induced error sources -- poor environmental design, poor training program design, poor design of procedural manuals, poor control of working shifts/durations/rotations, and so on. However, from strictly a human engineering standpoint, design-induced error usually refers to the controls and displays associated with operational or maintenance procedures.

The human engineering concept of "design-induced error" refers to the concept of determining if an error is caused by a basically poor design rather than by the apparent cause. Rather than accept "operator error," or "stress," or "poor training," or "mindset" as explanatory, the human engineering specialist seeks further to determine if the design of the equipment itself fosters error, or is inadequate in stress situations, or defies any amount of training, or is able to induce improper "mindsets" by violating population stereotypes.

The many deficiencies in the LER format, use, and implementation are well-documented in the "Analysis of Licensee Event Report (LER) and Noncompliance Data Related to Licensee Performance Evaluation," especially for the purpose of determining "root causes." However, that report, the NSAC #9 (70) and #35 (71) reports dealing with screening LER's, and an analysis of NUREG-0161 (75) (Instructions for Preparation of Data Entry Sheets for LER File). all suggest a heavy involvement of design-induced error -- both operational and maintenance.

Specific control room deficiencies in which design-induced error can be expected are well documented in the EPRI NP-309, Project 501, and EPRI NP-1118, Project 501-3 reports. Among these are:

1. Glare and reflection from lighting on instruments.
2. Mirror imaged control boards.
3. Subsystem controls widely separated from their associated alarm annunciators.
4. Poor location of some controls permitting inadvertent activation.
5. Use of qualitative instead of quantitative indicators.
6. Improper use of major, intermediate, and minor scale markings on meters.
7. Meters that fail with the pointer reading in the normal band of the scale.
8. Chart recorders with too many parameters.
9. Annunciator systems that are too complex along with complex equipment/procedures for

acknowledging, silencing, testing and resetting alarms.

10. Inconsistent color coding within a control room (for example: red to indicate normal circuit closed or normal flow and also red to indicate danger or abnormal conditions).
11. Poor labeling practice, including inconsistent abbreviations (for example, on one panel, four different ways of indicating "pump" were PU, PP, PMP, and PUMP).
12. Poor use of shape coding.
13. Adjacent meters that must be compared by an operator which have non-identical scales.
14. Recorder printouts that are illegible.
15. Lack of barriers for critical switches or control knobs.
16. Placement of controls and displays beyond anthropometric reach and vision envelopes.

All these problems and a number of additional similar ones were observed by the Study Group during its plant visits. Some of the problem areas obviously are inter-related and part of a larger generic problem. A few of these merit more detailed discussion.

Remote Emergency Shutdown Panels

One design deficiency that could result in a serious error situation is the difference in design between control room boards and remote emergency shutdown panels. Emergency shutdown panels are to be used in the event that the control room is disabled and the operators must control a shutdown in the plant itself. There appears to have been no attempt to make the emergency panels and controls as identical as possible to the panels and controls that are used for shutdown in the control room. Typically, the emergency panels are difficult to access, poorly lit, and covered or locked for security reasons. In the event of an incident that would require use of the remote panels, the difference in design, coupled with a high stress situation, represents a high risk for design-induced error.

Color Coding

Color coding implications exist for the various indicators on control panels; within annunciator panels, and in computer-generated displays on CRTs in advanced control room designs. The use of colors to enhance information transfer between the system and the operator is a technique well-known in military, as well as in civil, applications. Usually, color coding follows population stereotypes. For example, red refers to "stop" or emergency situations; green refers to "running" or operational status; white may indicate standby or available status; and amber or yellow appropriately indicates cautionary or warning situations.

The reality is that nuclear powerplants not only deviate frequently from such a convention, but that inconsistencies may occur within a panel, a console, and/or a control room for basic indicators. The same is true for annunciators.

In a typical control room, red indicators indicate "normal" (such as valve open, normal flow; or, circuit closed, electrical circuit functioning) while green indicates the opposite (valve closed; circuit open). A basic question is whether the general population stereotype should replace the historical convention that has existed in the power industry. What training would be necessary to overcome the ingrained habits of experienced operators? Should the coding practice favor the older experienced operators who may not be part of the operational system for many more years, or should it favor the stereotypes of new, inexperienced personnel who will live with the system for many future years?

Related to those questions is the concept of the "green board." Here, a control panel would have green-lit indicators for all functions that are operating in a normal, or in tolerance, condition for the current operating mode. In this framework, a low temperature, or high pressure, or low flow would not appear as amber or red based on the absolute value of the parameter, but would be indicated as green if that condition were normal for the mode existing (cold start-up, normal operation, emergency shutdown, fuel reload, etc.) at the time.

For CRT display formats, careful consideration must be given to the optimal use of color. The Study Group's exposure to the developers of advanced control room concepts shows that there is a general assumption that color coding enhances performance. Experience in the aerospace industry supports the assumption -- but not without two precautions. First, not just any type of color coding will enhance performance -- indeed poor color coding may actually degrade performance from that with a monochromatic display. Second, too many colors may cause confusion rather than improvement. The latter factor is based on the fact that the human may have difficulty in discriminating between various colors (blue from magenta, orange from red, etc.) based on the type of phosphor used, persistence, contrast and brightness, color purity, saturation, etc. The requirement here is that any color coding scheme must be based on empirical performance data rather than on hunches, expert guesses, or general feelings such as "the more colors the better".

Indicators and Instrumentation

NUREG-0578 (TMI-2 Lessons Learned and Short Term Recommendations), Section 2.1.3, called for direct indications of PORV and safety valve position as well as instruments for directly monitoring inadequate core cooling (or low reactor coolant level). Section 2.1.8 of that document calls for instrumentation that follows (and is able to help analyze) the course of an accident.

NUREG-0660 (NRC Action Plan Developed as a result of the TMI-2 Accident) section II.F calls for:

- additional accident monitoring instrumentation;
- additional instruments to aid in identification of and recovery from conditions leading to inadequate core cooling;
- instruments for monitoring accidents (Regulatory Guide 1.97 which specifies design criteria and the range for each instrument).

NUREG-0660, Section II.D.3, also requires the direct indication of relief and safety valve positions.

Naturally, it is essential that any additional instrumentation be integrated with current instruments in the control room. Merely adding indicators without a thorough analysis of the operator tasks involved in their use and of the associated design of related indicators may result in further error inducement. Additional instrumentation must improve the man-machine inerface, not degrade it.

General Control Room Design

Though specific design deficiencies have been discussed, it is apparent that the entire control room must be compatible with the personnel who perform the operations. The control room must be an integrated entity, not merely an area where all of the required controls and displays are placed on the basis of cabling, separation, and seismic criteria. The human engineering principles must comprise a major criterion in the design tradeoffs that are part of an intentioned system design approach.

CONSTRAINTS

For nuclear plants being designed currently, there is no unusual constraint that would prevent control rooms and plants from evolving systematically, thereby reducing/eliminating design-induced error. The importance of using human engineering design criteria in the trade-off analyses is essential to achieve that goal.

For plants in operation, there are real constraints which will not permit complete conformance to established human engineering criteria. Much, however, can be done to reduce design-induced error by "enhancement" -- the paint-label-tape process, as illustrated by the Summary Volume and Volume 1 of EPRI NP-1118.

A potential constraint for the design of future control rooms mentioned by several utility representatives is that regarding the use of large, unwieldy, and poorly human engineered meters, indicators, and controls. The argument is that these are all that exist which meet several NRC requirements, especially that of being seismically qualified. However, it is our opinion that if the utilities must pass the human factors requirements of NUREG-0700, and if they insist on controls and displays that are well-designed from a human engineering standpoint, the laws of supply and demand will prevail. When there is a need and a market, manufacturers will produce the items required.

PRESENT STATUS

There is a large amount of activity by all elements of the nuclear community in the majority of problem areas described above.

1. The NRC in NUREG-0660, Task 1.D.1 requires all licensees to complete a control room design review (CRDR) to "identfy and correct design deficiencies."
2. The NRC funded Essex Corporation to produce NUREG-1580, the draft for CRDR guidelines (July 1980).
3. The NRC issued NUREG-0659, a staff supplement to 1580 providing sample checklists and draft evaluation procedures.
4. The NRC published NUREG-0700 in September 1981, "Guidelines for Control Room Design Reviews." This document describes the CRDR process; suggests objectives and responsibilities; describes review of system functions, task analysis, CR survey, discrepancy compilation, requirements for workspace, annunciators, controls, displays, labeling, process computers, panel layout, and so on.
5. The NRC requires CRDRs to be completed by the utilities by April 1982 according to NUREG-0660.
6. The utilities have initiated enhancement improvements to existing control rooms.
7. The NRC has sponsored two workshops, organized by the IEEE in 1979 and 1981, on "Human Factors and Nuclear Safety."
8. The NRC contracted with the Human Factors Society to conduct this study and development of a research plan.
9. The NRC is developing NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Reviews," and is soliciting inputs from interested parties. This document is intended

primarily for internal guidance, but will also serve industry needs.

10. The NRC is developing a draft NUREG-0835, "Human Factors Engineering Design Review Acceptance Criteria for the Safety Parameter Display System." In its present form, it requires functional human engineering design criteria, but provides a desirable amount of latitude for specific applications.
11. The NRC has established (July 1981) a new Human Factors Research Review Group, with designated members from NRR, I & E, and RES. One major function to be addressed is to identify human engineering research needs relevant to the operator-machine interface focusing on the consequences of functional allocation and control room design.
12. The NRC has sponsored research to identify the instrumentation for status monitoring during accident conditions relevant to the latest revision of Regulatory Guide 1.97.
13. EPRI subcontracted a 22-month study in September 1980 to perform a critical evaluation of the data sources behind the criteria in MIL-STD-1472B and NUREG-0700.
14. EPRI is sponsoring RP 501-4, "Human Factors Review of Enhancement Approaches for Nuclear Control Rooms," being conducted by Honeywell with a scheduled completion date of December 1981. Its purpose is to produce a practical "how to" enhancement guide.
15. EPRI is sponsoring RP 1637, "Human Engineering Guidelines for Operations" with a multi-disciplinary team (Essex, Combustion Engineering, Babcock & Wilcox, Ebasco, and Bechtel). Its objective is to develop a guideline document focusing on advantages and disadvantages of alternative design options.
16. EPRI has sponsored RP 891-5, "Plant Safety Status Monitoring," to evaluate different approaches to safety status monitoring.
17. INEL is conducting for the NRC (FIN #A6119) a human factors review that includes a survey and investigation of potential negative impacts of control panel retrofits.
18. NSSS vendors are developing "advanced control rooms" with a professed goal of eliminating design features associated with human error.
19. The IEEE has a Working Group S.C. 5.5 on Human Performance. Two of its task groups are working on two key documents. One has developed a draft document, "Human Factors Engineering Requirements for Nuclear Powerplant Sytems, Equipment, and Facilities" (June 1981) modeled

after DOD's MIL-H-46855. The other has developed a draft "Recommended Practice for the Use of Color Coding in Nuclear Power plant Panels, Controls and Displays." The latter document is intended to provide guidance for intraplant color coding consistency rather than set industry-wide standards.

20. The IEEE has developed IEEE STD 566-1977, "Recommended Practice for the Design of Display and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations" and ANSI/IEEE STD 567 (October 1980), "Trial Use Standard Criteria for the Design of the Control Room Complex for Nuclear Power Generating Stations." Both documents are, however, quite general and not very detailed. Both are representative of attempts to achieve better man-machine interfaces; neither provides specific design criteria.
21. ORNL sponsored a "Review of Standards and Requirements Affecting Human Factors in Nuclear Power Plant Control Rooms," completed by Science Applications, Inc. (ORNL # 62B-13819C/62X-11). It assessed the existing OR standards and those currently under review/development by the NRC and ANSI/IEEE.
22. IEEE is attempting to compile a human factors bibliography pertinent to the nuclear industry.
23. INEL is conducting a study for the NRC (FIN No. A6308) on CRT display design and evaluation (FY81 obligation is $500K). Some goals include techniques for presenting information to operators in a more intelligent and efficient manner; development of diagnostic graphics for detection and identification of basic accidents.
24. The LOFT (Loss-of-Fluid-Test) reactor facility of INEL includes a highly instrumented nuclear reactor operated by the DOE for establishing nuclear safety requirements. Human engineering research is being performed there (see Item 23). An example is, "A Method for Quantifying Selected Human Engineering Design Standards," by W. W. Banks and M. P. Boone which the NRC is considering using or validating (April 1981).
25. INEL is conducting a study for the NRC on Plant Status Monitoring (FIN A6294) (FY81 obligaton is $450K) which includes operator information aids.

26. INEL is conducting a study for the NRC on Human Factors Reviews (FIN A6119) (FY81 obligation is $100K) which includes consideration of operator performance related to changes in displays, colors, and control relocation.

PLANNED ACTIVITIES

In addition to the activities indicated above as continuing in nature, there are the following planned activities:

1. The NRC has reorganized and established a Human Factors Safety Division in NRR which is to have about 300 people, about 10 of which will be dedicated to human factors.
2. The NRC plans to continue INEL work in CRT Display Design and Evaluation throuh FY87 (FIN A6308), Plant Status Monitoring through FY83 (FIN A6294), and Human Factors Reviews through FY87 (FIN A6119).
3. The NRC plans Control Room Studies through FY87, but no specific funding plans are yet approved for this.
4. The NRC plans analysis and classification of LER-derived human error data.
5. The NRC is developing a request for proposals to perform task analysis relative to control room design.
6. The NRC is soliciting letters of interest and capability relative to evaluation of computerized information displays (Commerce Business Daily, Issue No. PSA-7912, September 9, 1981).
7. The NRC has placed high priority ratings on future research or evaluation of human factors engineering data and on validation of control room modifications.
8. The NRC is initiating FIN B-2364 with Battelle Pacific Northwest Laboratory with incremental funding into FY82 for "Verification, System Status, Automatic vs. Manual." Part of the effort deals with "unambiguous indication of status at the system level."
9. The NRC plans review of SPDS designs from FY82 through FY84.
10. The NRC plans CRDRs from FY82 through FY84.
11. The NRC plans to develop criteria for an automatic system status monitoring system (now given in R.G. 1.47). The activity is to be completed in FY84 with guidance and criteria to be incorporated in regulatory guidance documents.
12. The NRC plans development of guidelines and criteria for control stations and panels

outside the control room, with funding planned for FY82-84.

13. EPRI desires to evaluate computer-based operator support systems in FY82 but has not received funding approval.
14. INPO plans to issue human engineering guideline documents to advise utilities on how to accomplish backfitting or enhancement fixes to control rooms.
15. INPO plans work to improve LER format, data input, and analysis.
16. NSSS vendors express desires to continue concept development of advanced control rooms with great use of computers and CRTs. However, the activity level here appears to be very low due to industry's lack of new orders.
17. The IEEE plans to continue its standards development.

MISSING ELEMENTS

If all of the current activities mature, and if all the desired "plans" are conducted, the goal to reduce design-induced human error to an acceptable minimum could be realized. It seems clear that the NRC and the nuclear industry know what needs to be done technically. However, there are two critical elements that currently are lacking -- inadequate human factors engineering technical expertise within the NRC; and inadequate numbers of career human engineering professionals working with the rest of the nuclear community.

At the NRC there are too few human engineering people who would be deemed as "qualified" by peers in the human factors profession. A set of good intentions by NRC managers is not sufficient by itself. Competent human factor professionalss must be obtained and given responsibilities commensurate with the magnitude of the problem.

Some NSSS vendors do have some qualified human factors professionals, either working directly on nuclear powerplant design problems or, at least, available to them from another division in the company. The utilities are not as fortunate, as a group, and find themselves searching for potential employees or for consultant services. The A/E firms have even less awareness of the need to employ, either on a full-time staff or a consultant basis, qualified human factors specialists. But, given the need that exists, the reality is that there are relatively few consultants or potential employees who possess a desirable mix of human engineering expertise and power industry experience. Consequently, the industry must either compete for such services or be content to hire qualified human engineering personnel who can be trained to learn the specific power plant technology. It is our opinion that (a) the few organizations and people who possess the desirable human factors-power industry mix are spread too thinly to be fully responsive to the nuclear industry needs and (b) the industry must take active and positive steps to attract human engineering specialists into their business.

TECHNICAL FEASIBILITY

There is no technical reason why the issues addressed under this "design-induced error" topic cannot be solved. Given the industry awareness of the problems to be attacked, and given the research and development plans of the nuclear community, the acquisition of qualified human factors professionals is necessary for eventual solution.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

It should be obvious that the attack on design-induced error is one which involves a generic human factors approach. Consequently, a systematic human factors approach will enable resolution of training, procedures, simulator, job performance aids, staffing, and organizational management kinds of matters.

RECOMMENDATIONS

1. NUREG-0700, or any subsequent improvement thereon, should be implemented as a requirement rather than as a guideline. The NRC should review license applicants not only in accord with NUREG-0700 and NUREG-0801, but also in accord with the recommendations of EPRI NP-1118. A guideline document for human engineering maintainability features that is similar to NUREG-0700 should be published.

Importance: High

Schedule: Urgency - immediate

Duration - 2 years

Resources: NRC Staff

Implementation: N/A

Dependencies: Maintainability data for NUREG-Maintainability Document

2. The NRC should produce a guideline document which requires license applicants to achieve designs of emergency shutdown panels with controls, displays, and layouts as similar as possible to those in the control room used for the same required functions.

Importance: Medium

Schedule: Urgency - 3-5 years

Duration - 1 year to produce Guidelines; 1 year for utility implementation

Resources: NRC Staff

Implementation: N/A

Dependencies: N/A

3. The local control stations, such as those located in the radiation waste control rooms, should be equipped with controls and displays that meet the human engineering design criteria of NUREG-0700 and NRC should produce a guideline document for that purpose.

Importance: Medium

Schedule: Urgency - 3-5 years

Duration - 2 years - 1 year for guideline; 1 year for utilities to implement

Resources: NRC Staff

Implementation: N/A

Dependencies: N/A

4. A serious study of the use of color coding, especially the use of red and green, and a serious study of the "green board" concept should be conducted both empirically and analytically.

Importance: Medium

Schedule: Urgency - 6-10 years

Duration - 3 to 4 years

Resources: 30-40 person-years

Implementation: Software, laboratory, reconfigurable simulator, test subjects, career HF professionals, behavioral/statistical analysts.

Dependencies: None

5. The NRC should contnue research and development on advanced display technologies such as that work currently being performed by LLNL (see 3.1.2.1.1 f) and INEL (see 3.1.2.2.1 e).

Importance: Medium

Schedule: Urgency - 1-3 years

Duration - 2 years

Resources: Contractor or National Laboratory
FY82: 2 person-years
FY83: 3 person-years

Implementation: N/A

Dependencies: N/A

2.2 Inconsistent Control Room and Plant Design

REQUIREMENT

There are two aspects to consider regarding design differences in control rooms specifically and in power plants generally. One aspect is that of the mere existence of differing designs; the other is whether these differences have any significant human engineering implications.

The reality of large differences in control room designs is dramatic, not only among utilities, but within the same utility for the same type of reactor system. The reasons for the wide variation have been identified as a lack of a systems approach and the lack of specific standards from the NRC.

When the wide variation in control rooms and the power plants is realized, it is not uncommon for people to suggest that there should be more commonality of design, or even full standardization. Reasons offered include:

- cost savings
- easier transition for employees who have been with another utility or in some other plant within the utility
- consequent reduction in training requirements
- less likelihood of error during stressful conditions.

Those reasons are suggestive of a requirement for as much standardization as possible. Therefore, the issue is considered here.

CONSTRAINTS

A greater degree of consistency in design is possible with a system approach. While the application of a system approach will only improve the situation, it is probably unrealistic to expect full consistency. Indeed, it may even be undesirable to pursue that objective too greatly. Certainly, if a plant contains more than one control room, it is desirable from many standpoints that they be highly similar. But the reality of technological advance, as well as the incorporation of new regulations, will make full consistency impossible when control rooms are designed at different points in time. It would be foolish not to incorporate technological improvements simply to match a previous design.

It is important to assess the human engineering factors regarding consistency. Certainly, there are broad human factors considerations relative to staffing, transfer of training, personnel practices, and so on. But, strictly from a human engineering standpoint, the real question is not whether or not plants are identical, but rather whether the specific features are error-inducing in themselves. For example, a system approach and standards for aircraft design will not result in identical cockpits for all aircraft. Similarity in layout is achievable, but the specific design features of the man-machine interfaces must be examined to determine if they result in error or unsafe performance.

PRESENT STATUS

There is no activity within the NRC which is actively directed toward achievement of identical control rooms and power plants. Instead, the development, revision, and drafting of documents, such as R.G. 1.47, R.G. 1.97, NUREG-0700, NUREG-0801, and NUREG-0835 are directed toward getting to better overall designs for nuclear powerplants from a human engineering standpoint.

The utilities, generally, do not acquire several nuclear power plants simultaneously. In the cases where there is simultaneous design, there is great similarity in control rooms, though there are examples where mirror-imaging unfortunately was used when the control rooms are co-located.

NSSS vendors have developed advanced control rooms which each vendor hopes will be flexible enough in its software to accommodate future modifications/additions without hardware changes. Examples are Combustion Engineering's "System 80" and G.E.'s "Nuclenet."

PLANNED ACTIVITIES

There are no formal plans to develop identical control rooms or power plants.

TECHNICAL FEASIBILITY

It is not realistic or feasible to retrofit existing control rooms to achieve identical control rooms or plant layouts, though moving from one plant to another, and potential for reduced licensing time.

RECOMMENDATIONS

1. The NRC should continue to develop standards such as NUREG-0700 to achieve specific design features that are based on sound human engineering principles.

2. NSSS vendors and utilities should continue their efforts in advanced control room design with a view to modularity and software flexibility so that future developments affecting control room functions will have minimal impact on hardware design.

2.3 Annunciators and Alarms

REQUIREMENT

The seriousness of the control room annunciator (visual or auditory alarm) problem is well-documented. Often 1000 to 2000 of these may exist in a control room to alert the operator to abnormal or emergency situations. Most of the visual indicators are placed into matrix panels above the control board. Each matrix may consist of 50-80 indicator "titles," each about 2 x 3 inches with a legend on the tile face.

Many specific problems exist in typical annunciator complexes because no standards have existed for their design features. Consequently, color coding is inconsistent, legend terminology varies, flash rates vary, faults are not presented in a hierarchical manner, various schemes are used for alarm acknowledgement, and auditory alarms have different characteristics of pitch, intensity, on-off cycle, etc.

The Kemeny Report (60) observes that, at "Three Mile Island, over 100 alarms went off in the early stages of the accident with no way of suppressing the unimportant ones and identifying the important ones." The EPRI NP-1118, Project 501-3 Final Report identified the problems (36, p. 4-1):

- Operators are given more information than they can reasonably assimilate when a major anomaly occurs.
- There is generally no differentiation between major and minor annunciators beyond isolation of "first out" annunciators.
- Legends are not sized to be read reliably from the operators normal workstation.

- Annunciators are not always located near their associated quantitative displays or controls.
- Legends often are inadequate or contain too many options.
- There are many "nuisance alarms" (those which indicate that a particular system is working normally).
- Good coding techniques are underutilized.

A DOE study conducted by INEL (114) found those problems as well as

- inadequate organization
- lack of a system engineering approach to annunciator design
- a lack of filtering out collateral alarms
- some types do not contain press-to-test circuitry
- alarm disabling procedures are unsatisfactory.

In order to have a good overview of the plant status, operators need a simplified presentation; but they also need detailed information to support diagnostic work. A systematic analysis of the alarm/annunciator system is needed so that such systems are compatible with the way humans solve problems, i.e., collecting only the minimum amount of information in order to arrive at a solution at the highest leve of abstraction. A review of current annunciator systems is essential; good systematic analysis of future annunciator systems is necessary.

CONSTRAINTS

The key point underlying the technical inadequacies, again, is the lack a system engineering approach. There are some technical constraints which will limit how much current control room annunciator systems can be improved. However, for designs being initiated currently, there is every reason to expect that a dramatic increase in control process warning system improvement is achievable.

PRESENT STATUS

Fortunately, the annunciator problem is receiving much attention in the nuclear community.

In NUREG-0700, there is a major and detailed section (6.3) with guidelines for "Annunciator Warning Systems." It treats parameter selection, multichannel or shared alarms, prioritization, coding, labeling, arrangement, readability, controls, response procedures, and other similar factors. Though these are guidelines rather than detailed specifications, adherence to the guide will eliminate many of the currently experienced problems.

INEL analyzed its own test reactor alarms with a view to modifying them and to establish some information for future work in alarm integration (EGG-SSDC-5088, April 1980).

INEL completed a study of "Nuclear Control Room Annunciators: Problems and Recommendations," (NUREG/CR-2147).

Danchak observed that a survey of existing CRT alarm systems "revealed no standard content for alarm messages." (26)

The NSSS vendors all have incorporated CRTs and a heavy utilization of computer-generated data in what they call their "advanced control rooms." Proper derivation of display formats should be able to reduce greatly, if not eliminate altogether, the myriad number of separate annunciators. In the NSSS vendors advanced control room concepts, however, are display formats which are based on engineering and operator judgments (guesses). The use of color codes and equipment symbology are highly individualized. Consequently, the same caution is necessary here as would have been desirable for the more traditional annunciator systems. That is, the display formats used in CRT presentation should be derived from a systematic analysis of the operators' information requirements. Then candidate formats should be developed and evaluated on the basis of human performance. The CRT terminal should provide the basis for additional data entry, querying, and corrective control initiation.

PLANNED ACTIVITY

The NRC is initiated a study of "Design of Annunciator Systems" (FIN B2365) with Battelle Pacific Northwest Laboratories, with $100K in FY81 and incremental funding planned for FY82. It includes work to analyze purposes of alarms in each mode of reactor operation, examine modifications required to accommodate DAS and SPDS installation, analyze alarm prioritization, and determine to what extent annunciator systems should be upgraded by use of computer conditioning of alarms.

In addition, high priority NRC research plans include annunciator concern as a part of operator task analysis, emergency procedures, and control room modifications, all topics in Group A (highest priority) of the NRR's Division of Human Factors Safety research plan.

EPRI has an approved research project for FY82 on "Evaluation of Annunciator-Warning Systems," the details of which are not known at this time. INPO plans to conduct a study of alarm presentation, information processing and display, but no concrete programs are known.

One of the most promising avenues for the better management of process control alarms is that of computer-generated faults and diagnostics on CRTs. This is an activity associated with the advanced control room concepts discussed earlier.

TECHNICAL FEASIBILITY

See CONSTRAINTS above.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Annunciator systems research and development activity has significant interactions with those relative to CRDRs, retrofit or backfit fixes, SPDS and DAS, and the "green board" concept.

RECOMMENDATIONS

The following specific recommendations are made:

1. The NRC should initiate appropriate rulemaking activity to require adherence to Section 6.3 of NUREG-0700 for existing annunciator system. Utilities should analyze the systems and identify changes which can be made toward compliance in order to enhance their effectiveness.

Importance: High

Schedule: Urgency - Immediate

Duration - 2 years - 1 year for rulemaking; 1 year for utility analysis and report.

Resources: NRC Staff

Implementation: N/A

Dependencies: None

2. NRC should sponsor studies to extend that reported in NUREG/CR-2147. The end product should be a standard or specification for annunciators of the traditional type dealing with flash rates, acknowledgement/silencing procedures, location, color coding, etc.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 1 year

Resources: 1 person-year plus NRC staff

Implementation: Career human factors professional

Dependencies: None

3. NRC should encourage industry studies to determine the requirements for development of logic systems aimed at

filtering or restriction of alarms. These studies are to be structured with a system approach to include consideration of operator information requirements, mode-dependent signaling, task analysis, functional hierarchies, prioritization, and other similar factors.

4. The NRC should sponsor studies and then issue an alarm requirements document for <u>advanced</u> control rooms using CRTs and computer-generated displays.

<u>Importance</u>: High

<u>Schedule</u>: Urgency - 3-5 years

Duration - 3 years

<u>Resources</u>: 12-15 person-years plus NRC staff; NUREG

<u>Implementation</u>: Laboratory and computer facilities, flexible programming, system modeling, test subjects, career human factors professionals, experienced nuclear engineer.

<u>Dependencies</u>: None

2.4 Design for Maintainability

REQUIREMENT

It is evident to the Study Group that an area of great significance is that of design features which affect the relative effectiveness of maintainability throughout the entire plant. The enormous number of valves, pipes, pumps, heat exchangers, condensers, generators, test and calibration panels, and special instruments require scheduled or emergency maintenance. That maintenance, of course, demands the use of a large number of personnel with specialized skills. Besides the consideration of cost of personnel, protective clothing, tools, and instruction manuals, another significant concern is with the radiation exposure limits for the maintenance personnel.

Clearly, if a maintenance person cannot easily find the desired piece of equipment, if that person has difficulty in gaining adequate access to it, if he or she must search through a poorly prepared manual, and if that person does not have the proper tools to use on the piece of equipment, then the maintenance person is potentially exposed to radiation for longer periods of time than would be otherwise necessary. This factor influences the number of personnel required, thereby resulting in additional costs to the utility (and to the public). Further, the likelihood of serious maintenance errors is increased, especially when such activity occurs in the presence of temperature extremes, noise, and inadequate lighting. Those errors may result in significant control room problems and errors.

The major human engineering needs are for:

- o better accessibility
- o better identification of equipments
- o better control of environmental factors.

The important Summary Report of EPRI NP-1567-54, Project 1126, identifies other human factors concerns for maintenance, such as procedures and training.

CONSTRAINTS

The primary constraint on optimizing good design for maintainability is that of space limitation. It is true that space for a plant is not limitless, and some improvement in maintainability design would require some additional space in some specific instances. However, it is likely that remarkable improvements could have been achieved within the existing envelopes of current plants if a systematic engineering design process had been followed. An example of a factor overlooked by designers is the need to accommodate maintenance personnel wearing protective clothing. Without a systems engineering approach, it is not surprising to find the makeshift ladders, scaffolds, and individualized attempts to improve equipment identification (labels on valve handles, painted arrows on pipes, stick-on-tape identifiers on pumps, felt-tip-pen schematics on walls, etc.).

CURRENT STATUS

Unfortunately, with the current emphasis on control room design, there has not been much attention paid within the nuclear community to this matter. To our knowledge, no major effort has existed within the NRC for improvement of maintainability design features from a human engineering standpoint. Instead, maintenance documents address specifications for materials inspection requirements, calibration frequency, and the like. Indeed, in an NRC memorandum identifying "priorities for DHFS Human Factors Research," plant maintenance was in the lowest of three priority categories with an overall priority ranking of 15 out of 16 in the categories (151). However, in a draft memorandum received from the NRC in late November 1981, Plant Maintenance was raised to a priority position of 10 out of 15 research needs (158). The NRC has a FY81 research program funded at $150K at ORNL on "Maintenance Error Model" (FIN No: B0461-1). However, it does not deal with design; it focuses, instead, on development of statistical human error models, based on task analysis, toward risk prediction techniques. The NRC has proposed future plant maintenance studies with a desired start date in FY83, but these are of the same nature as the current work.

PLANNED ACTIVITIES

The NRC is initiating a study to develop "Maintenance, Human Factors, and Procedures Guidelines," (FIN B2361) to be performed by Battelle Pacific Northwest Laboratories. Its total funding is for $510,000 with completion scheduled for October 1982. It includes the following scope and objectives:

- o survey documentation and several operating plants;
- o identify equipment, facility, personnel, procedural and organizational factors that result in maintenance errors and critical delays;
- o review equipment malfunctions resulting from improper maintenance practices, poor design, and inadequate training of personnel;
- o develop recommendations regarding human engineering maintenance guidelines.

EPRI, after Project 1126, has been considering a more thorough maintenance task analysis. In addition, EPRI is assessing the utilities' priorities in power plant maintainability with a survey form asking for high, medium, or low interest in EPRI's research candidates in this area. Candidates include human engineering questions relative to standards, design requirements, protective devices, outage analysis, mockups, and others. The specific work anticipated is unknown at this time.

INPO may conduct a maintenance study under DOE, and INEL has proposed a test maintenance error survey, but again for the FY82 or later time frame.

The A/Es represent a wide range of concern with this problem area. Expressions range from ". . .we've always designed plants this way and never had any complaints" to that of building scale models to assess piping routes, clearances, space for access, etc. Even in situations where models are used, there is no systematic process for the effective integration of personnel and equipment systems. The prevailing sentiment of A/Es is that the facilities they design and provide are in accord with the basic specifications to which they are contractually committed. It appears that the utilities commonly expect that the A/E will consider maintainability factors, but do not adequately specify the human engineering requirements therein.

A concrete step in the direction to assure human factors engineering in plant maintenance is evolving in the IEEE's development of a "Guide for Human Factors Engineering Requirements for Systems, Equipment and Facilities of Nuclear Power Generating Systems." As of June 26, 1981, there exists a proposed draft, Revision A-4, that is modeled after the DOD's MIL-H-46855 (a military document that provides guidelines for the conduct of a human engineering program), directed to all significant human interfaces throughout the entire power plant.

MISSING ELEMENTS

The NRC historically has not been vitally concerned with human factors aspects of maintainability. Utilities have been content to follow traditional maintenance practices combined with radiation protection requirements regarding exposure and protective garments. A/E firms may examine major features related to equipment removal and passage clearance, but detailed maintenance tasks have not received special scrutiny. With the exception of the NRC's project FIN B2361, human factors maintainability problems appear to have a lower level of general nuclear community attention that those related to control room design.

TECHNICAL FEASIBILITY

Granted that power plants are complex facilities, there are no technical problems to prevent significant improvement in maintenance from a human factors standpoint. Even with space limitations, it is highly probable that significant improvements in plant maintainability are relatively easy to achieve.

There is an abundance of OSHA requirements coupled with the DOD's MIL-STD-1472 standards that clearly provide for better access, labeling, illumination, tools, and other maintenance features.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Better maintenance design is closely aligned with design-induced error, safety of operations in the control room, procedural manuals development, training requirements, and staffing needs.

RECOMMENDATIONS

1. Emphasis of the NRC's research in the human factors maintenance area should be shifted from error models and risk assessment to design analysis. For example, we believe more reduction in error is possible by the development and application of maintenance guidelines rather than by the development of a maintenance model, for example.

2. The NRC should publish a Guideline document similar to NUREG-0700 that defines human engineering design criteria for maintenance. As part of that document, the NRC should require utilities and A/Es to demonstrate, through models, mock-ups, and task analyses, that critical maintenance tasks can be performed in an acceptable human factors manner.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 1 year

Resources: NRC Staff

Implementation: N/A

Dependencies: This should be done immediately upon completion of FIN B-2361 scheduled for completion in October of 1982.

3. The NRC sponsor empirical and analytical studies on development of (a) better protective garments and (b) better tools and instruments used for maintenance in a radioactive environment.

Importance: Medium

Schedule: Urgency - 3-5 years

Duration - 3 years

Resources: 9 person-years

Implementation: Requires personnel experienced in biomechanics, environmental physiology, biochemistry, and human factors.

Dependencies: This should be coordinated with EPRI work.

2.5 Design Freeze

REQUIREMENTS AND CONSTRAINTS

"Ratcheting" is a term commonly used throughout the nuclear industry. It refers to the process wherein the NRC may require the addition of instrumentation or control/display devices to planned or existing control rooms based upon operational experience. If a particular kind of incident occurs with some regularity in operational plants, or an accident of significant magnitude occurs, post-incident analysis may suggest that certain additional features would reduce its likelihood of occurrence. Since the design of a control room is usually complete prior to construction, and since the construction-to-operations cycle may consume some 10 years, the final appearance of a control room may be substantially different from its initial design. This process is a significant one because of (a) redesign costs (wherein large numbers of drawings, wiring diagrams, and specifications may be affected) and (b) the distinct possibility that a basically "good" design from a human engineering standpoint may become less than good because of clutter, intrusion on functional grouping, interference with controls, etc. Further, the change or addition may have questionable validity if measured against objective effectiveness criteria. A thorough examination of the effects of the ratcheting process is needed.

PRESENT STATUS

Currently, the utilities have little recourse when ratcheting occurs. They must comply with the specific requirements, or be prepared to defend a position that other design features already exist to serve the newly required function, or that alternatives to the specific requirement are equivalent to or better than that being mandated.

PLANNED ACTIVITY

To our knowledge, there is no specific effort within the NRC to study the ratcheting process or to evaluate its consequences -- either positive or negative. And, there is no effort on the part of industry to evaluate it beyond the comply or defend postures previously mentioned.

The imminent release of NUREG-0700 and NUREG-0801 should be helpful in resolving some of this problem. For the short term, utilities will be able to anticipate more precisely the human engineering requirements and evaluation criteria for control rooms currently designed or under construction. For designs in the more distant future, these requirements and criteria can be incorporated as part of the on-going system design process, especially if the IEEE's proposed "Guide for Human Engineering Requirements. . ." becomes effective as part of a solid system engineering effort.

TECHNICAL FEASIBILITY

Many technical additions to CRs may be ratcheted for other than human engineering reasons, yet they have human engineering impacts. The incorporation of SPDS, DASS, bypassed and inoperable status indications (REG GUIDE 1.47), instrumentation to assess post-accident plant conditions (REG GUIDE 1.97), relief and safety valve position indications, auxiliary feedwater flow indication, and others (all from Chapter II of NUREG-0660) are indicative of future features that may be perceived at times as "ratcheting" by some elements of the industry.

RECOMMENDATIONS

1. The NRC should perform an analysis to determine the relative merits of using a "design freeze" process vs. the currently used racheting process for the design of power plants. The design review process of the DOD should be used for the

design freeze model. Comparison factors should include a cost/safety tradeoff.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 1-2 years

2. If a design freeze process is found to be advantageous, any subsequent changes which are thought to be vital should be assessed from the standpoints of validity and significance. A potential approach that could be followed for that assessment is suggested below:

A. Determine the validity of the change. Does the change truly affect human operator performance in a positive manner? Data rather than hunches are needed. Subject the contemplated design change to "before and after" performance evaluation. Here, the NRC should use a subcontractor who might use mockups and/or simulators to evaluate design configurations without and with the design feature under consideration to collect actual performance data to establish the validity of the change.

B. Determine the significance of the change. The previous step may indeed show a performance improvement, but the magnitude of the benefit must be evaluated versus the cost of implementing it. If change shows only a trivial performance improvement, it may not be worth incorporation. If a large error reduction occurs, and the error is of a type that can have important operational consequences, the change becomes significant. The data collected by the subcontractor in the step above can be used by the NRC in conjunction with analysis of industry-wide experience to make a judgment about the significance of the tentative change. Admittedly, the decision may have a judgmental element, but it would be based on performance data as well as operational data.

3.0 Problems in Procedures and Operator Aids

The overall requirements for procedures and operator aids are driven by the application of the systems approach to the design and development of nuclear power plants. Firstly, the requirements for procedures are identified. Secondly, in conjunction with the identification of the training requirements, the requirements for operator and maintenance job performance aids (JPAs) can be defined.

The problems in the procedures and the JPA areas can be reduced by establishing standards that will help ensure consistency and adequacy across types of procedures and across plants. After the standards, i.e., specifications, have been established:

(1) The procedure development process needs to be defined.
(2) Format options for both operator and maintenance aids have to be considered.
(3) Adequate methods for (a) procedure implementation and revision, (b) performance verification, and (c) shift relief and turnover practices can be defined.

3.1 Standards and Specifications Governing Procedure Development

REQUIREMENT

The nuclear power plant, like any complex man-machine production process, requires procedures for operations, maintenance, and administrative control. The procedures serve as a blueprint for the human actions in the system. When defined as a resultant of the system analysis, the procedures provide the human with all the information needed to operate and maintain the nuclear power production process.

For operation, personnel have to:

a. Start-up the plant.

b. Operate the plant under normal conditions.

c. Operate the plant under off-normal operations (when normally available equipment that does not affect safe power production is unavailable, e.g., maintenance).

d. Operate the plant during transients that might occur, i.e., a normally operating system that does not affect safe power production may fail during normal operation.

e. Operate the plant during an emergency, i.e., the loss of primary equipment will require actions to continue safe operation or initiate a shut-down.

f. Shut-down of the plant. This includes the actions required to isolate an equipment or a component for maintenance.

For maintenance, personnel have to:

a. Conduct surveillance activities. These include the periodic inspection, testing, and calibration of equipment.

b. Perform maintenance by periodically cleaning and lubricating, removing, repairing, and replacing equipment.

For administrative control, personnel have to:

a. Determine what technical guidance is required to define the technical content of the operation and maintenance procedures.

b. Determine what developmental guidance is required to define the development process for the operation and maintenance procedures.

c. Determine the nature and extent of job performance aids that will be used along with the procedures and training to operate and maintain the plant.

d. Define the format guidelines to follow for the development of the procedures and job performance aids.

e. Define the process for procedure and job performance aid implementation, verifying operational and maintenance performance, and revising procedures and job performance aids.

f. Determine the requirements for passing shift responsibility to the incoming shift that ensures the incoming shift is made aware of all completed and ongoing maintenance and operator actions.

Given the scale of a nuclear power plant and the procedure categories identified above, there are literally thousands of procedures that have to be generated. Thus, specifications for all the procedure categories should provide a standard that would help ensure consistency across procedures. A well-developed specification eases the burden of the procedure developer by providing guidance that defines the scope of the procedure development activity. Besides providing guidance on the procedural steps, specifications can provide guidance on format, job performance aid development, implementation, and updating practices.

Within the government the use of specifications for the development of procedures is widespread. Particularly in the military, a complex man-machine system in which the number of subsystems is in the thousands, has specifications for the development, format, implementation, quality assurance and revision of technical documentation for operation and maintenance. These specifications have as a common goal the consistency across technical documentation, regardless of the subsystem for which the documentation is developed and vendor from which the subsystem and documentation are purchased. Specifications by themselves do not ensure consistency of technical documentation. Specifications, however, developed from a system analytic approach that considers the users' information needs will reduce the variability in the technical documentation across vendors to a minimum.

Within the system approach, regulations have to be considered prior to the identification of system requirements. Specifications are written to satisfy the criteria established in the regulations. Thus, well-defined specifications would provide a sound basis that identifies the constraints placed on the system requirements analysis.

CONSTRAINTS

Obviously the nuclear power industry is not the same type of organization as the DOD. The DOD is the customer who decides what procedural information it wants and writes the specifications stating how that information is to be developed. In the nuclear power industry, Nuclear Regulatory Commission (NRC) has a different role. Since the purpose of the NRC is to assure the safe production of power from nuclear fuel, the NRC assumes a role of defining the specifications for the customer, the utilities. The difference in roles, however, does not mean that the purpose of specifications has to be different.

If the utilities spoke as one and generated their own industry-wide specifications, the NRC could assume the role of review and approval. Unfortunately, when the nuclear power industry began, all the utilities were not created equal in their knowledge and understanding of nuclear power. It was a seller's market for the vendors and industry-wide specifications for the development of procedural information did not exist. Thus, the NRC chose to decide what was best and safest. In the area of procedures, however, the NRC was not very active. For the new nuclear technology, containment of the physical parameters was the prime concern. At Three Mile Island, however, the lack of information in the emergency procedures and a maintenance procedure that did not provide for the recording of a maintenance action contributed to the incident.

Since the utility is the customer, it can certainly generate its own specifications for procedure development. The problem that utilities run into, particularly in the maintenance area, is that vendors already have their technical manuals developed, i.e., off-the-shelf manuals. For the utilities to require a different manual would mean increased development costs that management may not want to assume. The utility may even be placed in the position of buying a particular product with its off-the-shelf manual or not buying the product at all.

The issuance of specifications by the professional organizations has helped to some extent. IEEE and the ANS have taken the responsibility of generating specifications to facilitate compliance with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50). Most of the IEEE and ANS specifications are joint issues of the respective professional organizations and ANSI. Unfortunately, the NRC has its own standards group to generate Regulatory Guides to satisfy the criteria of 10 CFR 50. While the Regulatory Guides can be used to endorse the IEEE and ANS specifications, exceptions to those specifications are sometimes made. This diverse generation process for specifications has contributed to the sporadic effort of specifications that are directly concerned with the development of operational and maintenance procedures.

PRESENT STATUS

After Three Mile Island, the NRC and the industry realized that the operational, maintenance, and administrative procedures used in the nuclear power plants were inadequate. This conclusion was emphasized at the 1979 IEEE Standards Workshop on Human Factors and Nuclear Safety. The workshop concluded that procedures need to be adequate, consistent, and standardized. To accomplish this, industry and NRC must review all operational and maintenance procedures systematically and comprehensively, investigate increased automation for rule-based procedures, and write procedures with the proper level of detail.

The NRC issued several documents that included sections on procedure problems. NUREG-0578 identifies the needs for a functional analysis prior to developing procedures and training and for vendors to generate guidelines to improve the technical content of emergency procedures. NUREG-0585 directs attention to emergency operating procedures by recommending the development of guidelines for content and format, the establishment of acceptance criteria, and the determination of which operational and maintenance procedures require independent performance verification. NUREG-0660 has as its objective for operating procedures the improvement of the content, wording, and format of operational, administrative, maintenance, testing, and surveillance procedures. Specifically, Tasks I.C.1 address emergency procedure revisions, I.C.2 shift turnover procedures, I.C.6 performance verification, I.C.7 vendor review of emergency procedures, and I.C.9 general updating of all procedures. Finally, NUREG-0737 clarifies I.C.1 by defining the analysis that must be performed to generate the content for emergency procedures, requiring the preparation of emergency procedure guidelines and upgrading the emergency procedures.

The recognition of the problem and the recommendations for improved guidelines are the first steps. Specifications, however, are required to assure achievement of an overall acceptable standard. In a review of existing specifications (31) the following IEEE and ANS standards were found to have some applicability to procedure development:

a. ANSI N42.7-1972/IEEE Std 279-1971
Topic - Criteria for protection systems
Comments - Tells what information is required, but not how to obtain it. Incomplete technical information will lead to incomplete procedures.

b. ANSI/IEEE Std 308-1978
Topic - Criteria for class 1E power systems
Comments - Establishes need for preventive maintenance testing, but does not tell how to determine the test intervals. Procedures development requires complete technical information.

c. ANSI N41.9-1976/IEEE Std 334-1974
Topic - Standard for tests of class 1E motors
Comments - Requires periodic testing, but does not provide an adequate means for determining test frequency. Without complete technical content, the procedures will be inadequate.

d. ANSI/IEE Std 338-1977
Topic - General principles for reliability analysis
Comments - Recognizes need for test procedure and operator feedback. Establishes need for preventive maintenance testing. Requires performance verification during test procedure. Requires the following of procedures as written. Requires test procedures to be periodically updated. Establishes a format for test procedures.

e. ANSI/IEEE Std 381-1977
Topic - Criteria for tests of class 1E modules
Comments - Requires preventive maintenance testing, but does not define time intervals.

f. ANSI/IEEE Std 387-1977
Topic - Criteria for diesel generator units
Comments - Requires preventive maintenance inspection and testing in accordance with manufacturer's recommendations. Procedures are to be generated based upon vendor's off-the-shelf manuals and supplemented with information obtained during operating experience.

g. ANSI N14.26/IEEE Std 497-1977
Topic - Criteria for post-accident monitoring
Comments - Requires identification of all operator actions. Provides guidance on what to do but not how to do it. Requires decision aiding in the form of post-accident displays.

h. ANSI/IEEE Std 498-1975
Topic - Standards for calibration and control of measuring and testing equipment
Comments - Requires calibration of equipment, but does not specify that the verification procedure should be performed by personnel different than those that performed the calibration.

i. IEEE Std 566-1977
Topic - Recommendation for design of display and control facilities
Comments - Stated as recommendations, this "standard" becomes a very weak document. Recommends functional grouping and match-up of functions to controls. Addresses accessibility, readability, and comprehension of information in displays, but does not offer definitions for these attributes, nor does it specifically address written procedures.

j. ANSI/IEEE Std 603-1977
Topic - Criteria for safety systems
Comments - Requires means for manual initiation of safety systems, but does not consider what decision aids the operator will require. It will be difficult to write comprehensive emergency procedures if decision points are not identified.

k. ANSI/ANS 22-1978
Topic - Earthquake instrumentation
Comments - Requires maintenance procedures but states that personnel with "normal skills" may not require a step-by-step format and fails to define normal skills.

l. ANSI N18.7-1976/ANS 3.2-1976
Topic - Administrative controls and quality assurance
Comments - Requires written procedures for maintenance, operation, and surveillance testing. Requires procedures to be reviewed prior to use, but does not state what the review should consist of. Does not address validatoin and verification. Requires procedures to be reviewed for updating at least once every two years. Establishes requirement for administrative controls and quality assurance. Provides format guidance for all types of procedures. Format guidance is nonspecific and incomplete.

m. ANS 3.5-1979
Topic - Simulators for training
Comments - Addresses use of simulators for operator training, but does not consider simulators for use in validating operational procedures.

n. ANSI/ANS 4.1-1978
Topic - Criteria for safety systems
Comments - Establishes need for effective procedures, but does not define what procedures will be required.

o. ANSI 18.2-1973/ANS 18.2a-1975/ANS 51.8-1975
Topic - Design criteria for PWRs
Comments - Recognizes human component and establishes a need for analysis to determine operator actions prior to developing procedures.

p. ANS 51.10-1979
Topic - Auxiliary feedwater for PWR
Comments - Requires validation of procedures under conditions as close as possible to full operation.

q. ANS 52.1-1978
Topic - Design criteria for BWRs
Comments - Recognizes human component and establishes a need for analysis to determine operator actions prior to developing procedures.

r. ANSI/ANS 58.4-1979
Topic - Criteria for technical specifications
Comments - Establishes requirement for defining surveillance and administrative control procedures.

The only Regulatory Guide that has been issued governing procedure development is 1.33 - "Quality Assurance Program Requirements." In its most recent revision (Revision 3 - July 1981) it adopts the latest revision to ANS 3.2 - "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants" (Draft 8 - April 1981). The NRC accepts ANS 3.2 as complying with the quality assurance program requirements in procedure development of Appendix B to 10 CFR 50 with the following additions:

a. For shift turnover procedures, there should be a short tour of the plant and there should be enough overlap to complete and sign a relief turnover checklist that defines specific plant parameters and equipment availability.
b. When following a procedure required by the technical specification, temporary changes can be made if approved by plant management.
c. For performance verification, personnel have to be documented with respect to the individual's knowledge of the system involved.
d. For review, approval, and control of procedures, all operating and maintenance procedures have to be validated by a step-by-step walk-through by the users.

The NRC Office of Nuclear Reactor Regulation has also revised the Standard Review Plan (90) used to review applications for construction and operation of nuclear power plants. Section 13.5.1 covers administrative procedures and Section 13.5.2 covers operating and maintenance procedures. Each describes areas of review, defines acceptance criteria, and lists the review procedures.

The NRC has recently issued NUREG-0799, criteria for preparing emergency operating procedures. This document does provide information on how to develop emergency procedures. It defines the technical guidelines that are required to generate procedural content. It covers the area of accessibility, availability, consistency, replacement, reproduction, presentation of information, and validation. For presentation of information, guidance is provided on organization, format, style, and sequencing of content. This document apparently takes the place of a specification in that it will be used to satisfy the emergency procedure requirements of 10 CFR 50. While this document is a start, development criteria are still required for normal operating, surveillance, maintenance, and administrative procedures.

While much has been written that addresses procedure development, only NUREG-0799 is directly related to the entire process of procedure development, and that only covers one type - emergency procedures. ANS 3.2 is a vast improvement over the earlier drafts in providing information on the control and preparation of all plant procedures. The basic approach, however, differs little from the 1976 version. Rather than requiring a systems approach to determine what and how many procedures are necessary to operate and maintain a nuclear power plant, it only lists the types that should be considered. Requiring specific types of procedures is a long way from defining the procedures should be generated. NUREG-0799 is hopefully the start of "how to" specifications.

PLANNED ACTIVITIES

The introduction of NUREG-0799 states that it represents the first step in developing a long term program for upgrading procedures, action item I.C.9, of NUREG-0660. Item I.C.9, however, goes far beyond emergency procedures and requires:

a. the integration and expansion of NRC and industry efforts in writing, reviewing, and monitoring of all procedures,

b. a clear concise format for all procedures, and

c. the determination of the inter-relationships among administrative, operating, maintenance, test, and surveillance procedures.

The effort to generate these specifications will be considerable.

The Institute for Nuclear Power Operations (INPO) has initiated a program that should ease the requirement for additional procedure specifications. INPO is developing writing guides that it hopes the utilities will be able to use to generate their own procedure specifications. A possible problem, however, is that each utility may develop radically different specifications. INPO's writing guides will have to be prescriptive to a degree, while at the same time allowing each utility to generate the specification it feels is the best.

MISSING ELEMENTS

The emphasis, justifiably, is on the generation of procedures. Job performance aids are an additional aspect of procedure development that should be considered. If portions of the procedures can be better accomplished through the use of job performance aids, the aids should be a part of the procedure. Thus, the specifications that govern procedure development will have to be broad enough to provide guidance on job performance aid development. If not, additional specifications will have to be generated that may be in conflict with the procedure specifications.

Another missing element is the role of the professional societies. Early on, the IEEE and the ANS provided many standards that satisfied the requirements of 10 CFR 50. Few standards had any direct relation to procedure development. If the IEEE and the ANS, in conjunction with the Human Factors Society, cooperated with industry and the NRC, then adequate and usable specifications could be written.

TECHNICAL FEASIBILITY AND PROBLEMS

As already mentioned, the precedence exists for the generation of procedure specification. The aerospace industry, in general, has generated its own specifications to provide customers with clear, concise, and adequately illustrated procedures in the areas of operation and maintenance.

One of the problems associated with the generation of specifications is a clear cut objective and well-defined scope for the specification. This requires the total cooperation of the community to which the specification is directed.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Procedure development specifications will have a direct impact on the system requirements analysis because the specifications should define the criteria for meeting the regulations of 10 CFR 50. Obviously, specifications will also impact the procedure development block by defining format requirements. Less dramatic, and depending on the extent of the specifications, procedure development specification could also impact:

a. the task analysis because it will define the technical content available to develop operational and maintenance procedures,

b. the personnel requirements because it will define the user population for which the procedures will be written,

c. the training requirements because procedures will be a part of training, and

d. job performance aids because some procedures may incorporate job performance aids.

RECOMMENDATIONS

o Technical Requirement

NRC assume the responsibility for developing non-plant-specific specifications for procedure development that define the criteria for meeting the requirements of 10 CFR 50. Specifications are needed for plant operation, plant maintenance, and plant administration. Each specification should address development, format, validation and verification, implementation, quality assurance, and revision. NRC should elicit the support of the professional societies (IEEE, ANS, and HFS) and coordinate the specifications development effort.

Importance: High

Schedule: Urgency - Begin in 1-2 years;

Duration - Complete in 3 years

Resources: In addition to the effort expended by the professional societies, NRC should be able to complete the effort as a staff function.

Implementation: NRC staff participants will require career human factors professionals who have a background in technical data development and presentation techniques.

Dependencies: Coordinate activity with requirements of NUREG-0799, ANS 3.2-Draft 8, and Regulatory Guide 1.33-Revision 3.

3.2 Procedure Development Process

REQUIREMENT

The failure to incorporate the systems approach for procedure development has been well documented. Swain and Guttmann (98) note that problems occur with written procedures because they are diffficult to read, difficult to locate, and inconvenient to use. A review of Swedish nuclear plants came to the same conclusions. In an analysis of the incomplete control rod insertion incident at Browns Ferry (56) it was found that the procedures were not detailed enough to provide the information the operator needed to recover from the equipment failure.

For maintenance, the problem appears to be worse. Seminara and Parsons (39) found that approximately half of the maintenance technicians interviewed in five nuclear power plants described their procedures as inadequate. Specific deficiencies included no page numbering scheme, materials and time required for tasks not listed, improper tag-out procedures, wide variety of vendor formats, vendor manuals not reflecting the nuclear environment, and inadequate detail for inexperienced users.

The systems approach when applied to procedures development will alert the developer to these problems. Adherence to the approach will identify the user population and provide appropriate detail for the population. The application of system integration to the development of procedures includes the types of analyses discussed below. The analytic process tracks the overall flow diagram described in Volume 2.

System Requirements. The system requirements establish the need for the operation and maintenance of the nuclear power plant. Existing documentation is the starting point, with the Technical Specifications as the source document. In addition, all specifications that have been developed will have to be considered at this point as these may affect the required output. The system requirement should account for:

a. The control room and equipment operators. Operators will require information to understand and learn how the plant functions.

b. The maintainers of the plant. While the operators may perform surveillance tests and calibrations to identify problems, the maintenance personnel perform the removal, replacement, and repair of the equipment.

c. Other support personnel who will need information on parts and design.

d. Administrative controls for formatting and implementing the above.

Function Allocation. With the identification of the system requirements, the function allocation will divide the various plant functions between person and the machine. Appendix B of NUREG-0700 describes as the starting point for function allocation the identification of the nuclear plant operating and safety functions that must be controlled and maintained to achieve the control room objectives of 10 CFR 50. This includes the integration and interaction of functions. In addition, the decision and information requirements needed to carry out the operating and safety functions have to be identified. Considerations of the information processing capabilities and limitations of both the human and the machine will determine what functions will be automatic (machine) and what functions will be manual (human).

To allocate functions, the design criteria of 10 CFR 50, along with applicable specifications, can be used to identify all the components, equipments, subsystems, and systems that have to be maintained and operated. The equipments are grouped together functionally, beginning with a top level breakdown to identify current and fluid flow to and from each functional group. The results will be block diagrams with all the input and output requirements identified for each block. The functional breakdown should continue until detailed block diagrams have been developed for all the identified systems, and all the plant components (valves, circuit breakers, controls, etc.) have been included within the various blocks.

The next step in the function allocation is the failure mode analysis. The failure mode analysis will identify equipment failure symptoms that are observed, heard, or felt. Symptoms have to be identified for all modes of operation in terms of their functional outputs, such as displays, vibrations, odors, noises, etc. Failure symptoms have to be identified for all components contained in the functional block diagrams. Additional outputs that are required for the procedure development process include:

a. Functional descriptions - narrative detailing the purpose, input, output, and interaction, of each functional block.

b. Measurement identifications - listing of the input and output measurements and the procedures needed to obtain them. The measurement procedures have to be identified for all modes of operations, e.g., plant start-up, hot shut-down, etc.

c. Schematics - detailed wiring and piping flow diagrams developed for all the functional blocks identified in the block diagram.

While the functional allocation lays the groundwork for the procedure development process, it is the task analysis that is the major starting point.

Task Analysis. The major product of the task analysis for the procedure development process is the task identification matrix (TIM). The TIM should list all the equipments that have to be operated and maintained within the nuclear power plant in a column. As column headings, the following task categories should be listed:

a. Operate - to control the various functions through manipulation of controls.

b. Adjust, align, calibrate - to maintain, change, or correct a component to achieve the required performance within the various modes of operation.

c. Checkout-troubleshoot - to verify the operability of a component, equipment, subsystem, or system and to isolate the source of a malfunction or failure.

d. Remove/replace - to interchange a malfunctioning or failed component with an operable one.

e. Repair - to restore a component.

f. Service - to periodically clean, inspect, and/or lubricate a component to maintain required performance. Service may require the disassembly and assembly of equipment.

The cell entries should indicate whether the task is automatic or manual. In addition, manual operation should indicate whether the task will be performed by the control room operators, equipment operators, maintenance technicians, or other support personnel. The TIM must track the function allocation. Thus, TIMs should be developed for the top-level function breakdowns and continue until all the personnel and man-machine interface requirements have been determined, and until all the procedures required to operate and maintain the plant under the various modes of operation have been identified.

While the procedure development specifications will aid in identifying task categories, a complete task analysis is required for each plant to ensure that all tasks unique to that plant are identified. The task analysis will serve as the major input to help identify personnel requirements subsequent training requirements, procedure development requirements, and required job performance aids.

As already mentioned, the failure mode analysis is required to identify equipment failure symptoms. This analysis addresses all failures, but a prime concern is with failures that will require initiation of emergency procedures. Therefore, the task analysis has to be extensive enough to identify all the symptoms that will be observable to an operator during an emergency. The operator will then be able to respond immediately to the initiating symptoms of an emergency without first having to decide what caused the emergency. The symptoms, however, should be correlated with functions to identify decision points for the operator to determine what plant function is being compromised. Then the operator can take the appropriate actions to deal with the emergency.

Personnel Requirements. An important aspect of the personnel requirements is the user description. The user description will determine to what level of detail the various procedural and training information has to be developed. The user description identifies the intended users of the procedures and should address:

a. Job relevant skills, knowledge, and experience, and

b. Reading ability.

A user description should be developed for each category of personnel who will be required to operate or maintain equipment within the nuclear plant. The personnel requirements have to correlate with the man-machine requirements to identify possible trade-offs. For example, a function identified in the task analysis as manual may, based upon the user description, have to be allocated to an automatic function.

Man-Machine Interface Requirements. As mentioned in the personnel requirements section, the man-machine interface requirements should correlate with the personnel requirements. The functions originally designated as automatic may not be within the state of the art or may for cost considerations be considered inappropriate for automatic functions. Therefore, personnel requirements may have to reconsider additional functions that will be under manual control. On the other hand, the output of several manual functions may serve as an input to a display that can then be developed as an operator aid. In addition, the procedures identified in the task analysis have to be analyzed to determine if a requirement for an operator or maintenance aid exists. The types of operator and maintenance aids considered should include mimics (both static and dynamic), graphics (developed from the functional blocks and schematic diagrams), and computer generated composite displays.

Control and Display Design. The man-machine interface requirements identify what types of operator and maintenance procedures should be developed. In the control and display design, the requirements for the design of various operator and maintenance aids are defined (i.e., what are the operator or maintenance personnel information needs?). With the information needs as criteria, formats can be planned for displays that are easy to use and that do not overload the user's information processing capacity.

Workspace and Workstation Design. The workspace and workstation will have an impact on the design requirements of various operational and maintenance procedures and aids. If a CRT or other electronic aid is going to be used, power requirements will have to be established, particularly for the loss of off-site power. In addition, consideration will have to be given to illumination, humidity, and cooling requirements. Access will also be a factor in determining the usability of the procedure.

The workstation influences the design of procedures and aids, and these design requirements influence the workspace. Thus, there will be trade-offs between control and display design, and workspace and workstation design to obtain the optimum mix and to provide the best design for operator and maintenance procedures and aids.

Training Requirements. The completion of the task analysis will provide a complete listing of all tasks that have to be performed. Input from the personnel requirements and man-machine interface requirements will provide a listing of manual tasks that will have to be performed by the operators and maintainers. While procedures will have to be developed for all these tasks, the indivi· .l training requirements relating to these tasks will also ·ave to be identified.

Procedure Development. In the task analysis all the tasks that ar· required to operate and maintain a nuclear power plant are identified. Implicit in the identification are the procedures required to accomplish those tasks. The first step of the procedure development process will be the decision - to what level of detail do the procedures have to be developed. As in the training requirements where a training/job performance aid trade-off will have to be made, the procedure development process also has to reflect a similar trade-off process. Procedures that are going to serve as a technical database for training materials should be developed to a different level of detail than the procedures that are going to be developed into job performance aids because the job performance aids will, for the most part, be the technical data.

Procedures that will become part of the training curriculum do not have to be as detailed as the procedures in the job performance aids. Training implies learning the procedure to a certain criterion. Thus, if a procedure requires opening valve 2A23, the training materials may simply state, "open valve 2A23." The assumption is that location of the valve and the performance required to open the valve will be learned when the procedure is practiced. On the other hand, the same procedure, if relegated to a job performance aid (a procedure that has not been practiced), will require details as to location and correct operation. This does not imply that procedures learned in training will not be documented in adequate detail. On the contrary, procedures, along with the information from the function allocation and task analylsis, are the source data and have to contain all the information that is required to accomplish all the identified tasks. The format of the procedure, however, will differ between training materials and job performance aids. Therefore, there has to be a correlation between the training requirements and the procedure development processes.

Procedures determined to be developed as job performance aids are held in abeyance until the job performance aid development process. The remaining procedures will be developed in a format compatible with the information requirements of the training materials. The format should provide a complete description of the procedure that is required to achieve the necessary end result, i.e., the correct performance of the operating and safety functions identified in the function allocation analysis. The format will be mostly narrative and will be supplemented with the necessary references to tables and graphics that satisfy the information needs of the task performer.

Procedures will have to be developed for all phases of plant operation. These include:

a. Normal operating procedures.

b. Non-normal operating procedures that cover operational transients and subsequent emergency procedures.

c. Start-up procedures.

d. Shut-down procedures.

e. Surveillance procedures - periodic checks and tests of the operability of various components, equipment, subsystems, and systems.

f. Maintenance actions - the removing, repairing, and replacing of malfunctioning components.

g. Administrative procedures - actions that affect the power output of the plant, procedures for tagging-out a particular equipment for maintenance, development and implementation of plant specific procedures and job performance aids, revisions, etc.

The technical content for the emergency procedures will come from the technical guidelines developed by the vendors and the owners' groups. These will have to be supplemented with plant specific information to generate functionally-oriented emergency procedures. Other procedures (e.g., normal operation, routine maintenance, calibration, etc.) can be developed based upon a particular event. The emergency procedures, as well as the maintenance troubleshooting procedures, have to be symptom-based. With symptoms as the initiating event, the operator can respond to the symptoms without having to determine the cause. This mode will probably not compromise plant safety. Decision points will have to be identified, however, to determine what plant function is being compromised to initiate the appropriate actions to terminate the emergency condition.

With the technical content defined, procedures have to be written, validated, and distributed. Writing requires technical writers taking the technical content and putting it into a form that is readable and understandable to the user population. The writing portion also requires reviews by editors and subject matter experts to assure that the material is written as intended. Validation is necessitated by naive users, i.e., operators and maintainers who are representative of the user population, but were not involved in the writing of the procedure, performing the procedure as written. All errors or deviations in performance have to be evaluated to determine if corrections have to be made to procedures. Ideally, a group (4-5) of naive users should perform the task and results should be compared across users.

With tasks that cannot be performed (e.g., some emergency procedures) alternatives such as walk-throughs or use of the plant simulator should be used. After the procedures are validated and produced, they are distributed to the appropriate plant facilities. Distribution also requires that the necessary approvals are obtained, and that updates and revisions are incorporated.

As procedures are developed, it will become evident that the earlier division between automatic vs. manual functions may have to be revised. In particular, certain procedures may have been identified as manual that upon more detailed development are better suited for automatic accomplishment. Thus, the procedure development feeds back into both the training requirements and the man-machine interface requirements.

CONSTRAINTS

The procedure development process begins with the vendors. Since the vendors design and supply the major equipment components, they have to develop initial operating and maintenance instructions. These instructions, by necessity, only address interaction with other components at a very basic level. For example, while the steam supply system is generally supplied by a single vendor, the turbine is not. Even when it is the same company, usually the systems are developed by different divisions or departments that may or may not talk to each other.

On the other hand, each plant is required to develop technical specifications that meet the criteria of 10 CFR 50. Thus, each plant, usually the architect/engineer, has to describe what the components in the plant can and cannot do. These specifications provide the foundation for the procedures development. If the system approach is not applied, as previously described, the architect/engineer will have to make too many assumptions about the interactions between system components and provide the utility with an inadequate base for the development of procedures.

These technical constraints are the greatest in the area of maintenance. The vendors for the many valves, pumps, motors, etc. are supplying off-the-shelf items with the existing technical manuals. With the large number of different vendors that supply the balance-of-plant components, the quality of the technical information varies greatly. The unfortunate thing is that the utility is faced with the choice of accepting the existing manual or having no manual. When faced with the prospect of rewriting all the technical manuals into a set of consistent maintenance procedures, the utility will accept the vendors' manuals and leave it for the maintenance staff to develop their own personal procedures for plant maintenance.

Organizational constraints also affect procedure development. Historical precedence relies upon the operational staff to develop the operating procedures. While the importance of incorporating operators as part of the procedure development staff cannot be overestimated, technical writers are also important to ensure clarity, conciseness, and consistency. In addition, a specification that provides specific guidance for procedure development would be helpful.

Validation of procedures, especially emergency procedures, is a real constraint. Obviously, all of the emergency conditions cannot be generated to validate the technical content of a procedure. Walk-throughs, talk-throughs, and use of the simulator have to be used to ensure the adequacy of emergency procedures. Other procedures, however, should be validated in the operational environment.

Generally, procedures are reviewed by subject matter experts to assess the technical content. Unfortunately there is a tendency to stop at this point. While the expert review is an important step, hands-on validation is required to ensure that the procedure can be followed and understood by the prospective users.

Within the maintenance area there is a real need to validate the procedures because of the potential for radiation exposure. For example, reseating a valve in a fossil fuel plant becomes a different task in a nuclear power plant if that valve is located in a contaminated area and requires the wearing of protective clothing. Time and accuracy, then, become problem that have to be resolved and has to be addressed during the procedure development process.

Finally, administrative procedures have to be established for the production and distribution of completed procedures. Procedure development is a dynamic process. Changes in requirements, modifications to components, and use all provide input for changes in procedures. Lack of specific administrative policies for production and distribution can lead to an inadequate procedures system even though the initial development process provided sound technical content.

PRESENT STATUS

Both the President's Commission and the NRC's report to the commissioners reviewing the accident at Three Mile Island identified procedures as a contributing factor. Kemeny (60) stated that the procedures were inadequate and Rogovin and Frampton (128) found the procedures:

a. incomplete,
b. lacking clarity and conciseness,
c. internally and externally inconsistent,
d. in noncompliance with ANSI N18.7,
e. did not contain useful decision aids,

f. did not identify the information operators needed, and
g. did not state how to verify that an action was completed.

The NRC, recognizing these problems were not unique to Three Mile Island, issued an action plan (NUREG-0660). The objective for Task I.C., in which operating procedures are specifically addressed, is:

> "Improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, explicit, and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing, and surveillance will be included. . ." (p. I.C-1).

In particular, four action items provide more detailed requirements. Item I.C.1 states that for the short term an accident analysis shall be conducted for each plant and revised emergency procedures shall be developed. Item I.C.7 provides for an in-depth review of selected emergency procedures for verifying correct performance of operating activities which have to be developed. Item I.C.8 establishes a pilot program in which the NRC reviews selected emergency procedures of near-term operating license applicants. Finally, item I.C.9 addresses long-term programs for upgrading all procedures. The NRC will develop a plan that will:

a. integrate and expand on current efforts in writing, reviewing, and monitoring plant procedures, and

b. detail the interrelationships among administrative, operating, maintenance, test, and surveillance procedures.

Shortly after the issuance of NUREG-0550, the NRC issued NUREG-0737 to clarify Item I.C.1. In upgrading the emergency procedures, the NRC required that the accident analysis:

a. address multiple failures,

b. be carried far enough to assure that all relevant thermal, hydraulic, and neutronic phenomena are identified,

c. consider operator errors, and

d. be supported with detailed documentation including (1) methodology, (2) function diagrams, (3) identification of the bases used for multiple failures, (4) supporting analyses, and (5) relationship of generic results to plant specific applications.

The Procedures and Test Review Branch (PTRB) in the DHFS is presently involved in two major efforts, a short-term emergency procedures review for NTOL applicants (Task I.C.8) and long-term upgrading of all procedures in a nuclear power plant (Tasks I.C.1 & I.C.8). For the short-term program, the PTRB will review vendors' guidelines for selected procedures and compare the guidelines to the NTOL's plant specific procedures. After the procedures have PTRB's comments incorporated, each of the selected failures will be walked through in a simulator. Changes will again be incorporated, and a single selected failure will be walked through in the plant. The results will be included in the safety evaluation report (SER). An underlying assumption is that the other procedures are written in a similar manner. The process for NTOL has no systematic data collection process.

This short-term program is useful, but the extent of its usefulness is difficult to determine. With no systematic application of evaluation criteria, the variability of the effectiveness of the emergency procedures can be great. Instead of walk-throughs on a simulator, procedure validation with the anticipated user would be a much better indicator of procedure effectiveness.

In addition, the PTRB uses the Standard Review Plan (NUREG-0800, Section 13.5.2) to review license applicants' plans for development and implementation of operating and maintenance procedures. The Standard Review Plan comprises areas of review, acceptance criteria, review procedures, evaluation findings, and implementation.

With the appearance of the draft criteria for preparation of emergency procedures (NUREG-0799), the NRC (in this case the PTRB) has taken a step in establishing consistent guidelines for procedure development. The first attempt at providing guidance for procedure development was part of NUREG/CR-1580. Unfortunately, the section on procedures was very broad and did not address the various types of nuclear power plant procedures.

Prior to the formation of the PTRB, the Office of Inspection and Enforcement (IE) issued two contracts for the development of two checklists for evaluating procedures, one for maintenance, test, and calibration procedures (NUREG/CR-1368), and one for emergency procedures (NUREG/CR-1970). Both of these development efforts required the of a methodology to ensure the adequacy and usability of procedures. Unfortunately, since NUREG-0799 has been issued only in draft form, and no document has been issued that delineates the criteria for the development of maintenance procedures, it is difficult to understand how these checklists will be used. The checklists do provide a standardized evaluation instrument that IE inspectors can use to sample and assess individual plant procedures.

The regulation of the procedure development process involves several players in the NRC. The LQB of the DHFS evaluates the utility's plan, i.e., the utility's commitment to develop administrative procedures. In a similar way, the PTRB evaluates the utility's plan to develop operating and maintenance procedures. The PTRB also evaluates the technical content of emergency procedures. The completed procedures, however, are evaluated by IE. Operating plants' procedures are evaluated on a continuing basis using a sampling technique by the resident IE inspector.

Recognizing that procedures for nuclear plants are not always systematically or efficiently developed, e.g., procedures are sometimes written by individuals not familiar with the operation of the particular system, Nelson, Clark, and Banks (69) investigated the application of functional analysis to reactor operations. Nelson, et al., state that functional analysis can be used:

a. to develop operating procedures,
b. to determine physical and informational aspects of control panels and CRT displays,
c. to structure the content of training for all plant personnel, and
d. to develop diagnostic procedures for troubleshooting.

They proposed a methodology that comprises system state diagrams, functional decomposition of operator actions, flowcharts, and extracting flowchart information to develop tables of specific actions.

Parallel to the methodology, Nelson et al. offer the following guidelines:

a. If the consequences of error are large, do not rely on memory. Refer the operator to appropriate cross-reference procedures or list steps in the current procedure if there is not enough time available.
b. Do not rely extensively on cross-referencing.
c. If a display or control reference is necessary, use a specific designation to facilitate information gathering.
d. Specify criteria for making decisions, e.g., verify it is often used without any indication of how.
e. Emergency procedures should be symptom based.

Hollenbeck, Krantz, Hunt, and Meyer (46) did a functional analysis of operations for the LOFT control room. The functional analysis was used to determine operator information needs during normal and emergency operations by:

a. Looking at the system to determine relevant operating modes and transfers between modes.

b. Developing mode-to-mode transfer procedures.

c. Flowcharting to determine the operator's decision points.

d. Tabularizing the information according to:
   (1) required decisions,
   (2) information required to make decisions,
   (3) source of information,
   (4) time available to act,
   (5) feedback associated with correct action, and
   (6) alternative action available if malfunction occurs.

The result of this functional analysis was used to make recommendations for improving procedures, and for the design of the CRT displays for an advanced control room.

Nelson et al. (69) investigated the use of response trees for detection, diagnosis, and treatment of emergency conditions. Response trees correlate specific accident sequences with a set of generic emergency procedures. The operator can use response trees to monitor system status, differentiate accident sequences, and select the correct procedure to follow.

Response trees differ from fault trees. Fault tree generation employs a technique that monitors characteristic parameters and evaluates deviations from normal values. Finally, an appropriate response is recommended. Response trees, on the other hand, do not require the predetermination of all potential accidents and do emphasize immediate response. Because the methodology is symptom-based and not event based, accidents are treated after they occur. The logic of the response trees illustrates system and component relationships. The effect of support system failures can be determined. All accidents are treated equally. Also, response trees can facilitate the detection and correction of multiple failures.

In addition to the research and guidance performed by the NRC, the utilities, vendors, and owners' groups have focused on upgrading nuclear power plant procedures. Based upon established priorities, the bulk of the effort has concentrated upon emergency procedures.

The typical role of the NSSS vendor in procedure development consists of the vendor developing emergency procedure guidelines, which in the past were event based, along with a system description, operation requirements, and interface requirements. The utility, which is resonsible for the final safety analysis report (FSAR), would then integrate these items to develop the plant specific emergency procedures.

With the requirement for symptom based procedures, the vendor in conjunction with the respective owners' group, would use its engineering department's subject matter expertise to determine what system components had to be monitored and what operator actions were required. A rationale for each decision was also compiled. The owners' group would receive the described actions and compile block diagrams. At this point the prose of the guidelines would be reviewed several times by various subcommittees in the owners' group. A simulator would be used to resolve discrepancies.

The finished product is supplied to the utilities along with background information. The background information documents for each procedure in the analytical process are used to develop the plant-specific procedure. This information can then be used as input for training by the utility.

The owners' groups have expended considerable effort in developing technical guidelines that are reanalyses of the emergency procedures that apply to each vendor's type of steam supply system. Rather than focusing on a particular event, the analyses for these guidelines try to determine what symptoms are likely to be present and then set out actions for alleviating the symptoms until a decision can be made that will identify the initiating event. At least one owners' group has attempted a function analysis to develop the functional relationship and the function flow among the plant safety systems. Using event tree analyses of probable events and best-estimate computer analyses of various branches of each event, diagnostic procedures are developed that describe actions based upon process parameter indications, thus eliminating the possibility that a response will be made to an incorrectly diagnosed event.

These technical guidelines represent the first step for upgrading emergency procedures. It will remain for each utility to take these guidelines and develop plant-specific procedures that correlate unique plant controls with proper identifying information to the functions that have to be performed.

To provide a strong base, the technical guidelines will have to be validated, to the extent possible, on simulators and supplemented with walk-throughs. This will be a requirement of NUREG-0799.

Utilities have developed administrative procedures that are well-documented. For example, Philadelphia Electric (123) has prepared a document on tagging and blocking that tells what is to be done and why it is to be done. Of course, if the procedures for tagging and blocking are not applied and followed, an explicit set of administrative procedures is useless.

PLANNED ACTIVITIES

A 1981 memorandum on research needs describes the requirement to address all plant procedures. Four needs were identified:

a. a survey of plant procedures to identify deficient procedures,
b. development of guidelines based on research for upgrading non-emergency procedures,
c. development of alternative ways of presenting procedures to operators and other plant personnel, and
d. collection of data to determine the effect of the recent changes in emergency procedures for the purpose of evaluating future changes.

To guide the research, a preliminary taxonomy for procedures was developed by the PTRB. The major areas in the taxonomy are personnel, environment, and information presentation. Under personnel, training requirements, experience, and behavior have to be considered. Environment includes physical conditions and information availability. Finally, for information presentation, consideration has to be given to format, time compatibility, level of detail, the use of job performance aids, and technical content. To evaluate procedures, a methodology has to include measures for comprehensibility, readability, utility, and acceptance by the user.

In prioritizing the research needs, NRR grouped the needs into three groups, an FY81/82 start, an FY82/83 start, and an FY83-87 start. Of the four needs related to procedures, all are in the first group.

The Office of Nuclear Regulatory Research (RES) responded to these needs by proposing an FY82 start that includes:

a. Summarizing existing procedural guidance from NRR, IE, and INPO.
b. Monitoring the Halden project related to computerized presentation of procedures.
c. Performing a scoping study of the cost-benefits of more rigorous compliance to procedure guidelines.
d. Examine available simulator data on performance with revised emergency procedures.
e. Develop an experimental design for continued evaluation of simulator data, if necessary.

In comparing RES's response to NRR's research needs, there does not appear to be a very close match. Apparently, RES feels that enough work has been done on development guidelines and all that needs to be done is to summarize the NUREG reports, e.g., the IE checklists and NUREG-0799. Halden's computerized display system for procedures will provide the information needed to evaluate an alternative presentation method.

RES appears to doubt the validity of the technical content in plant procedures and feels the emphasis should be placed on procedure adherence. Thus, RES is proposing a study to determine the costs of strict compliance with procedures as written and complete technical review of all procedures by NRC. There is no question that technical content is of prime importance. Proper presentation of that information, however, is equally important. To separate the two can only lead to future problems. Procedures have to be developed in accordance with the user's needs and capabilities. A research effort that would evaluate the effect of format changes on performance will help identify the best way to package the technical information. Unfortunately, RES does not think such an effort is required.

NRR also has a technical assistance contract for FY82 with Battelle Pacific Northwest Laboratory on maintenance problems at operating nuclear power plants. Part of that effort includes interviewing operators and maintenance technicians to obtain their views on the problems with maintenance procedures. The maintenance procedures themselves will also be reviewed.

Among the program objectives listed in the NRC's long-range research plan (NUREG-0740), there is one on recommendations to implement improved systems and procedures. Within the program plan there are research areas for role definition and accident response procedures. Under role definition, a task analysis is proposed to clearly define the roles of operating, maintenance, operating support, and management support. Completion is expected in FY83. With accident response procedures, the goal is to enhance the quality and utility of written procedures. This effort will require the analysis of accident sequences to identify necessary content revisions. The effort will incorporate evaluation of operator comprehension, operator acceptability, and procedure readability. Unfortunately, completion of the effort is not anticipated before FY87.

MISSING ELEMENTS

Because of priorites, the emphasis for the present and near future is on emergency procedures. NUREG-0700 requires a systems review that will provide the analysis needed for upgrading most of the control room procedures. Unfortunately, the area that requires the most attention is given the least -- maintenance. Many incidents are traced to inadequate and incorrect maintenance procedures.

Although the utilities and NRC have moved from event-based emergency procedures to symptom-based emergency procedures, research is needed to determine if better presentation techniques can be found. At Three Mile Island, the operators followed event-based emergency procedures, assuming the wrong event led them to the wrong procedure. The symptom-based approach permits the operator to treat the symptoms regardless of the cause and maintain plant integrity indefinitely without determining the cause. An approach that incorporates diagnostic information and decision aids would permit the operator to treat the symptoms initially, while at the same time leading the operator to the initiating event so that it can be corrected.

In general, procedure development beyond emergency procedures appears to be a low priority item. With the efforts expended on emergency procedures going well, upgrading of other plant procedures is not a major concern. It is obvious that RES is non-responsive to NRR's needs. NRR, however, has not interacted as extensively as needed to provide input for RES's research efforts. Part of the problem is that NRR requires a quick response time to specific needs and cannot appreciate long contractual lead times. Lack of communication also contributes to the problem. NRR and RES have to interact more closely. Physical separation does not help matters.

A parallel to the symptom-based emergency procedures is the development of troubleshooting aids for maintenance that are symptom oriented. The system review will have to be extended to identify the functions and tasks associated with maintenance. In addition to determining the remove, replace, repair, and service tasks, a troubleshooting analysis will be required to determine the observable malfunction symptoms.

Another missing element in the maintenance procedures is a planned maintenance schedule. Utilities should initiate a program with the maintenance vendors to schedule the clean, inspect, and lubricate tasks that will contribute to overall plant efficiency and safety.

Besides types of procedures that relate to the safe operation and maintenance of nuclear power plants, administrative procedures have to be established within each utility to ensure specific guidelines or development, production, and distribution of all procedure types. Administrative procedures are required to establish a continual updating policy that will ensure that all procedures are reviewed on a periodic basis, and that a mechanism is available to permit feedback from tasks performed in the real-world environment and incorporated into procedure revision.

TECHNICAL FEASIBILITY AND PROBLEMS

The development of readable, usable, and technically complete emergency, normal operating, and maintenance procedures is feasible and has been accomplished for many complex military and industrial systems. Unfortunately, the trend in the engineering world is to have the design engineers write the procedures. Engineers are fully capable of designing the many complex systems. But, operation under normal and emergency condition, and maintenance are very different from design. Technical writers, along with subject matter experts, are required to generate good procedures that become better through a rigorous review system, validation, and verification.

There is no question that reanalysis, rewriting, and development of improved procedures can be an extremely costly undertaking. When incorporated as part of the system approach on new plants, the additional cost, if any, is minimal. Even the rewriting required to update the procedures in existing plants will not be that costly because each plant will be required to perform some sort of systems analysis. With this analysis as the source data for improved operations, maintenance, procedures, and training, the task of upgrading the procedures will be straightforward.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

As discussed under the Requirements section, inputs to procedures are the personnel and man-machine interface requirements. The user population will greatly influence the level of detail required for procedures, as will the available displays, controls, and workspace.

The development process itself has to consider the training requirements and the trade-offs that will be made, i.e., what procedures and what degree of learning will be incorporated into the training. The greater the reliance on training, the lesser the level of detail required for the procedures.

Procedure development impacts the hardware/software requirements, too. For example, if the procedures are going to be automated, the development may be different than a totally hard copy set. The use of computers and CRT displays present a wider flexibility that will affect procedure content differently than reliance on hard-wired displays.

Finally, the procedure development process will be impacted greatly by the incorporation of job performance aids. For example, if specific graphs and flowcharts are developed to facilitate the understanding of certain tasks, then these aids must be integrated into the procedures to which they apply. If tasks are going to be supported entirely with a job performance aid, the need for a procedure disappears. As already discussed, the automated job performance aids will have the greatest impact.

RECOMMENDATIONS

Each utility should develop plant specific guidelines for emergency operating, normal operating, maintenance, and administrative procedures. Each guideline document will identify covered procedures and provide information on technical content, development process, format, validation and verification, implementation, quality assurance, and revision. The development process should recognize the importance of a technical writing group consisting of a career human factors professional, technical writers, and staff subject matter experts from operations and maintenance. These guidelines should reviewed by NRR against the non-plant-specific specifications for compliance.

Importance: High

Schedule: Urgency - Begin in 3-5 years

Duration - Complete in 1 year

Resources: The utilities will require a 2 person-year effort and the NRC will require an additional person-year to review the guidelines for compliance.

Implementation: Both the utilities and the NRC will require career human factors professionals with technical data development experience for the compilation and review of the guidelines, and nuclear engineers for the input and review of the technical content.

Dependencies: Effort cannot begin until the non-plant-specific specifications have been generated. The system analysis/review of NUREG-0700 has to be completed prior to procedure development to ensure complete coverage of required procedures.

3.3 Job Performance Aids

REQUIREMENT

The written procedures are, by definition, job performance aids (JPAs) because most are used to perform a procedure. In reality, however, the procedures are a technical data source. With emergency operating procedures, the individual is expected to follow the procedure as written line by line. In contrast, maintenance personnel seldom follow written instructions while performing a particular action. A job performance aid is supposed to help a person complete a task without requiring extensive training for that task. Within the nuclear power industry there is a place for both paper job performance aids and computer-based job performance aids.

Although research in the area has centered on maintenance of military systems, evidence (57, 130, 131) exists that supports the importance of JPAs:

a. in reducing task performance time,
b. for increasing task accuracy,
c. in allowing inexperienced users to accomplish the task, and
d. in reducing the selection and training requirements.

The JPA role in nuclear power plants can be just as dramatic, but that role has to be defined.

Mimics are already being used within the industry. Static mimics functionally group controls and displays together to provide the user with a perceptual representation of the system logic. This reduces the cognitive load on the operator. Dynamic mimics provide feedback information upon completion of certain actions that permit the operator to upgrade the status of his cognitive model of the plant. Graphs and tables can also be used to present information in a concise form. Functional flow diagrams that integrate schematic information with panel controls provide a valuable source of information to the operator or maintainer.

With the addition of computer display terminals, the JPA role should expand. Dynamic mimics can be displayed that can present upper-level system representations with searching-to-function representation at lower levels. With the proper software, alarm information can be distilled and integrated to provide the operator with a manageable subset of information.

In maintenance, JPAs can be used to alleviate many of the present problems. Exposure, for example, can be reduced to a minimum if tasks are reviewed prior to performing. This would require the establishment of a photo or illustration file of the various plant components and a presentation medium (e.g., a slide projector or computer display). By reviewing and rehearsing the task just prior to performing it, exposure time in contaminated areas will be reduced because the technician will know what the task consists of before doing it. A maintenance job performance aid that simply lists the tools required to perform a particular task goes a long way in ensuring correct performance of a task.

Performance verification is another area that would benefit from using job performance aids. A computer-based management system can facilitate scheduling of planned maintenance. When a procedure for one component requires isolation from other components, a computer or a rigid checking procedure can facilitate isolation and reintegrating the component when the task has been completed.

For all these JPA uses, however, a commitment has to be made to incorporate the JPA aspect of the procedure. In some cases the JPA, when constructed for a specific task or group of tasks, will take the place of the procedure. To accomplish this, the system approach has to be applied because JPAs will be impacted by the personnel and training requirements. Based on the training requirements, JPA/training trade-off ground rules have to be established to determine what should be put into JPAs. The trade-off rules determine what tasks the operator or maintainer has to perform using:

a. training alone,
b. procedure alone,
c. JPAs alone, and
d. combinations of the three.

The personnel requirements will provide the user description.

The trade-off ground rules are generated from consideration of what the operator or maintenance technician has to do to operate and maintain the power plant (the task analysis), and what they are capable of doing (user description). While all the manual tasks can be learned via training, it is uneconomical to do so. In addition, for stressful situations, complete reliance on training (memory) could contribute to performance decrement. Thus, in generating the training/JPA trade-off ground rules, the following factors should be considered:

a. Ease of communication - learning vs. book (JPA)
b. Criticality of the task
c. Complexity of the task
d. Time required to perform the task
e. Frequency of the task or similar tasks
f. Psychomotor component of the task
g. Cognitive component of the task
h. Equipment complexity and accessibility
i. Environmental constraints
j. Consequences of improper task performance.

While a particular set of ground rules that would be applicable for all LWR power plants has not been established, the following general rules (adapted from 59) should be helpful:

Put into training:

a. Tasks that are easy to learn through experience on the job, e.g., day-to-day control room operation and plant maintenance.

b. Tasks that are hard to communicate through words, e.g., difficult valve adjustments.

c. Tasks that require extensive practice for acceptable performance, e.g., replenishment of fuel.

d. Tasks that are critical, i.e., where the consequences of error are serious, e.g., the immediate operator actions required for an emergency procedure.

e. Tasks that are performed frequently.

f. Tasks with a response rate that do not permit reference to a printed instruction, e.g., start-up procedure that requires constant monitoring and shifting of range monitors.

g. Tasks that require a team effort.

Put into job performance aids:

a. Tasks that require long and complex behavioral sequences, e.g., the subsequent actions of an emergency procedure.

b. Tasks that are rarely performed, e.g., the control room operations required during a refueling outage.

c. Tasks that require verification of readings and tolerances, e.g., valve alignment procedure.

d. Tasks that would benefit with the inclusion of illustrations, tables, graphs, flow charts, and schematics, e.g., a mimic that illustrates the responses of a specified sequence of actions.

e. Tasks that require branching, e.g., a diagnostic or decision aid that list failure mode symptoms.

f. Tasks that are extremely costly to train, e.g., simulation of all possible emergency procedures in which all the immediate and subsequent operator actions are learned.

g. Information that can be integrated and presented in a concise format.

h. Information that will reduce exposure time, e.g., preview a procedure.

Some tasks or parts of tasks may require coverage by both training and JPAs, e.g., an operating procedure that includes reference to a valve alignment table. The application of the ground rules to the manual tasks is a preliminary step. During the actual development of the training curriculum and JPAs, the analyst may discover additional trade-offs that have to be made. In addition, the procedure development process may uncover tasks that have to be considered in the training requirements analysis.

After it has been determined what tasks will be learned through training and what tasks will be developed as job performance aids, a behavioral task analysis is required for those JPAs that will be used as procedures.

The behavioral task analysis lists each task element of a given task sequentially with each element analyzed in human performance terms. The delineation of the behavioral task analysis requires the analyst to identify and describe exactly what the user perceives and needs to do to accomplish the task. The above requirement usually means that the analyst will have to either do the task himself or observe another doing it. Each task element has to be carefully analyzed and described in detail. Along with each task element, the following should be described:

a. Hardware interface - the controls, displays, support equipment, etc. the individual performing the task will encounter

b. Criticality - what are the consequences of performing the task incorrectly

c. Cue - what does the individual see, hear, smell, and feel to initiate the task

d. Response - what action is required by the task performer when the task is initiated

e. Feedback - what indication does the task performer have that the task element was completed correctly

f. Performance criteria - what are the time and accuracy constraints of the task

g. References - what was the source data (Technical Specificatons, etc.) used to generate the task element

h. Graphics - what illustration or graphic representation is needed to supplement the task element

A task that is supported entirely by a JPA is a complete start-to-finish, step-by-step action arranged in a logical sequence of occurrence. Steps are the separate instructions within the JPA. Graphics are usually used to support the JPA task and are not intended to serve as the main focal point of the JPA. The required level of detail for both text and graphics will be a function of the characteristics of the intended users.

The final step for JPA development is the validation of the JPA. Once the JPA has been written, personnel representative of the intended users should perform the task on the equipment with no information other than that contained in the JPA. Successful performance will be an indication of the validity of the technical accuracy and intelligibility of the JPA. Tasks performed incorrectly should be corrected and revalidated.

JPAs that are going to be used to facilitate a procedure, but are not replacing the procedure, have to be developed in a similar manner. JPAs of this type have to be validated to determine that information, whether it is a mimic, functional flow diagram, or decision aid, is both technically correct and usable by the intended users. In addition, if these JPAs are integrated into various procedures, the validation, verification, and updating processes have to consider these JPAs.

CONSTRAINTS

Technically, only the JPAs that are going to be computer-based are constrained. The constraint is more in the form of room for the diplays rather than capabilities. In the control room, available room and location for the display terminals may limit what can be considered for computer-based JPAs. Within the plant itself, other constraints have to be considered.

Access and environmental constraints (heat, dust, noise, etc.) may prevent portable display terminals being taken to a job. Using a remote controlled device, e.g., a hand-held radio transmitter/receiver and scanner, may rely heavily on signals that could become degraded when used in remote parts of the plant.

A bigger constraint is the acceptance of job performance aids by management and the users. Hard copy-based JPAs require more paper than the abbreviated procedure format. Utilities may be reluctant to expend the extra time and effort required to produce documents that are going to consume more volume. As mentioned, to produce good JPAs requires additional commitment by management to validate, verify, and continually update the JPAs. This commitment will mean an information management system to continually track each JPA. Of course, when JPAs are incorporated with the procedures, the additional effort, if any, will be minimal.

The acceptance by the users of JPAs is not a problem when JPA development is part of the system approach. As modifications to existing practices are developed, however, lack of user acceptance can eliminate any potential gain that was anticipated.

In hard-copy-based aids, multilevels of detailed format are rare. Therefore, the level of detail required for the anticipated user is a prime concern. Too much detail causes the user to feel he is being seen as less intelligent than he is.

Too little detail leaves the user with the responsibility of not understanding the aid, or worse, missing important steps. Thus, the development process should incorporate user comments and reviews to facilitate user acceptance.

Technical content is very important. If the user discovers or perceives that information contained in a JPA or procedure is wrong, he will not use it. Technical content has to be validated. Verification by the user will rarely occur. The JPA has to be part of the orientation and training processes. The user must:

a. understand why a particular JPA is useful,
b. accept the validity of the data source used to generate the JPA, and
c. understand the logic used to produce the JPA.

Some JPAs require training or an explanation before they can be used. This training or explanation has to be considered as part of the JPA and integrated accordingly.

Acceptance of computer-based aids demands the same considerations. In addition, ease of use is important. If the user has to interact with the JPA through a keyboard-type interface, then access to the information must not be a complicated process. Hard-copy JPAs and procedures usually have tables of content, indices, and cross-indices that offer the user various ways to assess the information. Computer-based JPAs should provide similar devices. However, if the interface requires the learning of a new skill to gain access to the information, the user may refuse to learn that skill and thus obviate the JPA.

Finally, the NRC attitude may be a constraint for the incorporation of JPAs. The NRC has to recognize the JPAs are part of the overall systems approach. Perception of JPAs as an add-on will foster an attitude whereby the JPAs are considered nice to have, both hard-copy and computer-based, but not necessary. Attitudes are hard to change, and if the regulatory agency does not take a positive approach, the utilities will also be resistant to change.

PRESENT STATUS

Most of the work that is being done in the area of job performance aids incorporates a computer. Some of the short-term improvements, however, that have been added to control rooms since Three Mile Island include items such as static mimic displays and functional flow lines to group controls.

The changing of the emergency operating procedures from event-based to symptom-based represents an innovation in the hard-copy-based JPAs. Other efforts include EPRI's ongoing feasibility study of taking examples of different procedure types and organizing them into a standard JPA format -- discrete steps supported by graphics. In a study by Roddis et al. (27), it was recommended that flow charts be used to reduce the textual bulk of procedures and to aid in the understanding of the procedures.

Oak Ridge National Laboratory has an ongoing effort directed at determining the role of operational aids. Within this effort ORNL has proposed the operational aids listed below for the specific behavioral action:

Behavior	Operator Aid
detection of problems	alarm filter system
information acquisition	flexible display and communication system
verification of information	on-line data verification system and actuator monitor system
determine plant status	system which evaluates essential data and classifies states, maintenance monitoring system
determine action plan	procedure recall system
determine effect of action	on-line, real time dynamic simulator

While this sets a perspective for the types of aids that are needed in the control room, it is not clear that the function needs of the users have been considered.

In a study that reviewed at maintenance aids, photos, slides, films, and video tapes used to preview a task helped to reduce radiation exposure time. Audio aids were another type of aid that proved useful in facilitating task performance.

Philadelphia Electric (122) is developing a computer monitoring system for plant components. The utility is attempting to label all the plant components with a bar code for identification. Using a hand held scanner and audio transmitter, technicians can verify plant locations and correct status of components. As a management system, it can also be used to keep track of components that are tagged-out to prevent inadvertent violations of the Technical Specifications.

As already mentioned, the research focus is on the use of computer displays in the control room. In review of advanced control room displays, Seminara and Eckert (37) found two organizational schemes for accessing information. One organized the information by level of detail. Using tree structure, information parameters were monitored, and when more information was required, branching could be done to obtain either a control display or a diagnostic display. The second organized information by plant state or operating mode. A predetermined parameter set was displayed to satisfy the operator's information requirements. Upon request, branching into specific systems could be accomplished. Seminara and Eckert found that neither alone was adequate in all circumstances. The study listed four factors that have to be considered in computer-based displays.

a. Access - should be natural to the operator, difficult access could present problems during a stressful period

b. Feedback - to tell the operator the display is functioning the way it should

c. Response time - should be short, when the operator activates the interface device something should happen

d. Automatic monitoring - of set parameters that alert the operator to the display would be helpful

In the aftermath of TMI, the NRC formulated requirements for more control room instrumentation that would aid in the analysis of abnormal and accident conditions within a nuclear power plant. Initially, the NRC and the industry probably envisioned some combination of instrumentation and processing capabilities that would act as an all-knowing sooth-sayer. In a more practical approach, the NRC's interest focused on two somewhat related safety systems: the safety parameter display (SPDS) and the disturbance analysis and surveillance system (DASS).

In NUREG-0585, a lesson-learned recommendation concerned the man-machine interface. Part of that recommendation set forth a requirement for an SPDS. An SPDS (also called a safety state vector monitor) that defines the safety status of the nuclear power plant had to be available to control room operators. This SPDS should comprise a concise set of easily assessable information to ascertain the safety status of the nuclear power generation process. The safety status had to be a formation of the various barriers against radioactivity. The development of a more advanced system for disturbance analysis was also recommended.

NUREG-0696 was the first attempt to define and describe the SPDS and its relation to DASS. The SPDS is to help control room operators make quick assessments of plant safety. The formation of SPDS is to evaluate the safety status of the plant by providing continuous indications of parameters or derived variables that represent plant safety. While the SPDS is defined as only a monitor, it has to operate in both normal and off-normal conditions, and display information for both steady-state and transients. SPDS may also serve as an information source to other systems.

The SPDS parameters or derived variables can be individual plant parameters, or composed of a number of plant parameters, or derived variables that provide an overall system status. While a complete set of functions to be monitored is not defined, a minimum set includes:

1). reactivity control
2). reactor core cooling
3). reactor coolant system integrity
4). disactivity containment
5). containment integrity

Finally, SPDS should be flexible enough to allow for future incorporation of advanced diagnostic concepts and evaluation techniques, such as DASS.

The NRC's action plan (NUREG-0660) defines the objective for control room design as the improvement in order to prevent accidents or to cope with accidents, if they occur. Specifically I.D.2 and I.D.3 call for a plant safety parameter display console or monitoring safety system status. Improved control room instrumentation research (DASS) beyond the SPDS requirements are encouraged in Item I.D.5.

Before it was called SPDS, it was a safety state vector, and the NRC tried to determine the technical parameters to make the concept feasible. One approach selected important accident sequences based on the relative amount of public risk. Physical phenomena associated with the sequence in terms of a unique set of measurable parameters were defined. Then, events involving operator action were identified with an event tree. The tree was expanded and additional physical phenomena were defined. Finally, the necessary and sufficient parameters to describe plant status and system availability of potential components to terminate the sequences were determined.

DASS goes beyond SPDS and as such may prove to be a valuable aid to the operator. DASS will be used to monitor plant parameters to determine if any are outside of or approaching tolerance limits for a particular mode of operation. DASS would provide the information needed by the operator to return the plant to normal or mitigate the disturbance. Controls for reacting to the disturbance could be included. The general requirements for DASS are:

a. Assist operator in diagnosing the primary cause of disturbance and/or direct attention to the area of disturbance.
b. Provide timely disturbance recognition and provide information for corrective action.

c. Perform signal and data verification.
d. Assign priorities to the information
e. Provide information during normal, off-normal conditions and during accidents.
f. Interact with operator.
g. Adapt to a changed plant configuration when equipment is out of service.
h. Be generic, modular, and expandable.
i. Be reliable.

EPRI (40) did a feasibility study on DASS and described DASS as being able:

a. To identify the nature of the disturbance and possible corrective actions.
b. To enhance content of the displayed information.
c. To predict future propagation of disturbance if uncorrected.
d. To prevent disturbances by anticipation, i.e., provide constant monitoring.
e. To assist operator in taking correct action.

In comparing DASS and SPDS, Disalvo (154) noted that DASS could be used to satisfy the requirements of SPDS as long as the SPDS design criteria are met. DASS is viewed more as an operator's aid with dependence on it increasing over time until DASS is used as the primary source of information. This assumes that the information will be reliable and that the operators will accept DASS as the primary source of data.

Another area in operator aid research is the augmented operator capabilities program at the Idaho National Engineering Laboratory. As part of this program, the Operator Diagnostic and Display System (ODDS) is a computer-based graphics display system that will aid the operator in diagnosing particular plant disturbances. Within the graphics display, two display types were possible - process schematics and status and trend plots (46). The process schematics are simplified schematics with parameter values and component status information (e.g., valve position). Symbols and colors are used to establish representative conventions. Status and trend plots are used to:

a. Present status of one or more crucial plant parameters,
b. Review recent past history of these parameters, and
c. Derive operating limits of these parameters appropriate for the mode of operation for which the display was intended.

A vendor developed system that approximates the DASS and satisfies the display requirements of SPDS is the critical function monitoring system (CFMS), by Combustion Engineering (65). The CFMS displays provide integrative plant state information to the operator in mitigating the consequences of off-normal events. They provide real-time feedback information to illustrate to the operator the consequences of a performed action. The CFMS has a hierarchal display of these levels: Level one presents overall status, with branching to Level 2 for function status, and Level 3 for subfunction diagnostics.

Perhaps the most comprehensive undertaking in computer-based operator aids is the STAR system (23). Akin to DASS, the principal objectives of STAR are:

a. To determine the course of the disturbance.
b. To provide plant status information.
c. To present the best corrective action to mitigate the disturbance.
d. To recognize the significance of events and alarms compared to present plant conditions and preceding events and compared to the plant's generating mode.
e. To enhance the information of alarms during disturbances and filter alarms to reduce the number of extraneous alarms.
f. To predict the future propagation of disturbances.
g. To assist in a post-trip analysis.
h. To consider human performance capabilities of the operator to prevent confusion because of information overload during the disturbance.

As can be seen, most of the operator aid effort is in the development of computer-based aids. Given the technology, significant improvements are possible. Costs may be very high for both hardware and validated software, but can be offset if a few unnecessary shutdowns are eliminated. Unfortunately, only a small portion of the JPA effort is directed at determining what the needs of the user are and tailoring the design of the JPA to those needs.

PLANNED ACTIVITIES

In FY82, the Office of Nuclear Regulatory Research will continue its operational aids element. Under it, RES will (1) evaluate engineered safety features for level of automation, man-machine interaction, and operator decision points, (2) apply developed function allocation criteria to a selected engineered safety feature, and (3) determine the impact of automation on operator selection, training, qualifications, and performance requirements.

The NRC's long range research plan (NUREG-0740) has planned for two categories of activities. Under plant status monitoring, an effort scheduled for FY83, is the identification of accident signatures to develop diagnostic and corrective action aids. Under operational aids, work is scheduled to continue through FY86 on DASS. This DASS effort would emphasize the identification of recommended operator actions to be taken in the event of a disturbance. Specific areas of research to be addressed include:

a. Identification of functional requirements and criteria for computerized aids.

b. A generic task analysis for the reactor operator to provide a data base for human factors requirements.

c. Development of data and design criteria for computerized systems.

d. Identification of human performance data using CRT displays.

On the industry side, the major effort in the future is EPRI's continuation of the development of DASS. By the end of 1982, DASS should have incorporated alarm filtering. In addition, EPRI's effort will (1) determine integration of information requirements, (2) consider prevention as well as detection, and (3) develop a descriptive model characterizing factors influencing operator decision making. By 1983, DASS should be able to detect disturbances, verify control and safety actions, monitor technical specifications, provide subsystem surveillance, and derive integrated parameters. The 1984-1985 time frame should have a DASS that can provide simple predictions about disturbances, assist the operator with the performance of procedures, and verify system performance by comparing different analytical techniques.

As can be seen, the types of planned research activities are not that different from the present status. For the future, the development of job performance aids that are not computer-based displays is nonexistent.

MISSING ELEMENTS

As a general category, more emphasis should be placed on the development of hard-copy-based JPAs. There are many procedures, e.g., initial start-up, that could be improved if they were reformed into a JPA. JPAs oriented to specific tagging and blocking would go a long way in improving safety.

The one area that would benefit the most from JPAs is maintenance. The maintenance procedures are poorly written with heavy reliance on off-the-shelf vendors' manuals that may be in as many formats as there are vendors. Computer-based aids, like the one under development at Philadelphia Electric, can solve many of the location and identification problems that can easily happen in a plant with numerous similar components.

Another area in which the computer can function as an aid is the storage of many of the procedures for display. With this, the large amount of voluminous material could be reduced substantially. If nothing else, the computer can act as a page turner and fast indexing device.

TECHNICAL FEASIBILITY AND PROBLEMS

Job performance aids have been used in the military and other industries for over 15 years. While the technology, as with any technology, can be improved upon, there is enough research evidence and guidance to develop usable JPAs. The analyses that have to be performed for the upgrading of procedures and the control room review could also serve as the source material for operator JPAs.

JPAs evolved because as systems became more complex, the maintenance of the systems also became more complex. The same methodology used to develop JPAs that have been used to maintain military systems in the areas of preventive and corrective maintenance, and troubleshooting, can be used to produce the same types of JPAs to maintain a nuclear power plant.

Utilities may question the cost-effectiveness of developing hard-copy-based JPAs. In a military system, a single JPA development effort may benefit several hundred copies of the system. In a nuclear power plant, the system interface is usually unique to that nuclear power plant.

It must be remembered, however, that procedures have to be developed regardless. The additional cost of developing JPAs has to be compared to reduced probability of a procedural error contributing to lost operating time.

The use of a computer-based display to present procedural information is totally within the capability of modern technology. While some format programs will have to be resolved, the real problem will be the NRC's regulations. Can a utility place its procedures in digital storage for display and retrieval without having a paper-based backup in the control room? The tasks that are performed in the more remote areas of the plant, particularly maintenance, would probably still require hard-copy JPAs and procedures for performance.

The continued research on computerized operator aids, such as DASS and STAR, should provide the operator with a fairly sophisticated device to assist in control room operation. The potential problem with this rapidly growing technology is that the devices will become too sophisticated. In other words, the user's needs have to be considered and the display device has to be designed accordingly. Devices that are hard to use and intimidate the user will sit in the control room unused. Improper design of the interface and display components can negate much of the potential benefits of a device. While the technology may exist to permit full automation, the human may refuse to accept it.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The major interaction of JPA development is with the development of the instructional system and procedures development. If a trade-off analysis is conducted, then an economically efficient package of training and job performance aids can be developed. Because different user populations exist in a nuclear power plant, the system approach has to consider the training/JPA trade-off for each unique group. The trade-off exercise will eliminate redundancy while ensuring that all identified functions and tasks are covered. JPAs will become part of the technical data base, and as such will have to be integrated with training.

In procedures, a similar trade-off should be conducted. While it would be possible to develop JPAs for all the procedures, it would be uneconomical to do so. Many of the procedures are done on a frequent enough basis, and when presented in a clear, understandable format would not benefit from a JPA, except on first use. Therefore, the procedures indentified in the function and task analysis will have to be reviewed to determine which ones should be developed as JPAs. At the same time, procedures can be identified that would benefit from a JPA being incorporated as part of that procedure.

The development of computer-based job performance aids will impact the training requirements and the training itself because the JPAs will have to be integrated with training. Hardware and software development will also be impacted because the computer-based JPA will require hardware for display and software to support the makeup of that display. A computer used to display procedures may require slight format changes in the procedure development. Finally, the design of the computerized display will be impacted by the functions it was allotted and the personnel capable of operating the device.

RECOMMENDATIONS

In conjuncton with NRC's and INPO's job/task analysis efforts and implementation of NUREG-0700, and based upon user needs, the NRC should establish requirements for hard-copy, electronic, and computer-based JPAs. These efforts should include determination of the fundamental user requirements for the more sophisticated multiple JPAs, e.g., SPDS and DASS. Specifically, all system requirements have to be identified and functions allocated to ensure that the JPAs are designed and developed to satisfy those requirements and functions. Both operator and maintenance JPAs should be considered.

Importance: High

Schedule: Urgency - begin in 3-5 years

Duration - complete in 2 years

Resources: Level of effort cannot be determined until the requirements have been established for the various JPA types.

Implementation: Personnel skills will require career human factors professionals, system analysts, and subject matter experts.

Dependencies: Completion of NRC and INPO job/task analyses, implementation of NUREG-0700, and coordination with ongoing EPRI research. This effort should interact with the ISD process regarding the training requirements and the personnel selection process.

3.4 Format for Procedures and Job Performance Aids

REQUIREMENT

Given that the decision has been made to develop procedures and job performance aids as part of the technical data, it then becomes important to determine the best way to present that information. An optimal format requires decisions concerning the mix of text and graphics, the level of detail of both text and graphics, the appropriateness of graphics, page layout, typeface, and style. The overall objective of the format is to provide for the presentation of information in an unambiguous form.

In general, the format is considered a series of frames. A frame can consist of a single page or two facing pages. Each frame is an integrated verbal text and graphic illustration field. Each frame is organized such that only the illustrations needed, if any, to support the verbal text are included. For a computer-based format, the frame would be the usable portion of the screen that presents undistorted text and graphics.

Many formats are possible, but the four most accepted are:

a. Division of the frame, vertically, where text is presented on the left and graphics are given on the right. Another version of this type divides the frame horizontally and places the textual material on the top portion.

b. Recognizing that illustrations, depending on use, vary in size, the illustrations are placed as required to support specific parts of the step.

c. In the first two formats the illustrations support the text. In the third type of format, the text supports the illustrations. This format is mostly illustrations and as such contributes a lot of bulk to a task that has a JPA in this format.

d. Complete text where illustrations are rarely used. Within this format task statements are written as discrete steps. Another version of this is a columnar presentation that presents action steps on the left and expected results (feedback) from the action and supplemental information on the right.

In the nuclear power plant environment, the first two formats and the last would probably be the most acceptable. For straight presentation of procedures, the last format seems to be the preferred one, although a modified form of the second would be better.

When considering the format for the text, requirements have to be established for:

a. layout and size
b. typeface and size
c. borders
d. page numbering scheme
e. indexing
f. method of tracking change pages
g. placement of warnings, cautions, and notes
h. paper stock
i. binding method

For illustrations, requirements have to be established for:

a. quality
b. level of detail
c. angle of view
d. locator illustrations
e. item enlargement
f. exploded views
g. callouts

CONSTRAINTS

Technically, there are no constraints for the hard-copy-based formats. The assumption here is that the appropriate guidance of the formats has been determined and distributed. With computer-based formats, the constraints are the limits of the display device. These constraints could affect presentation requirements if principles that were established for hard-copy formats are violated, e.g., use of upper case only to present textual information. Research may be required to determine if the hard copy format principles are appropriate.

As with job performance aids, the bigger problem concerns acceptance. More than any other area of concern, the level of detail of the format for either JPAs or procedures will be the deciding factor in user acceptance. The level of detail for a format with no assumptions could have very specific descriptions about certain tools and their use, whereas a level of detail for a format that assumes a "master" operator can leave large gaps in the technical content, e.g., "verify XYZ" as a stand alone statement may require considerable control manipulations and display readings. As long as the "master" operator always does that task there may not be a problem. Thus, two extremes are possible. The extremely detailed format may be rejected by the users as being too simple. On the other hand, the minimal level of detail format can lead to errors of omission.

As with other areas of concern, if the system approach is taken, a lot of potential constraints and problems are alleviated. Information from the task analysis and definition of the personnel requirements will provide the input needed to arrive at a level of detail that will be appropriate for the user population. Since different types of procedures and JPAs will be associated with different user populations, level of detail will vary across user populations.

PRESENT STATUS

The NRC's major contribution to format is NUREG-0799 . In an earlier effort (NUREG/CR-1580) a weak attempt was made to present some format information. Unfortunately, it was too broad to be applied to nuclear power plants in particular.

NUREG-0799, however, is a useful foundation document. In it, information and guidance for hard copy-based emergency procedures are presented. It contains the following format topic areas:

a. cover page
b. identifying information
c. page layout
d. placement of warnings, cautions, and notes
e. sign-off provisions
f. divisions
g. emphasizing
h. letter size
i. letter style
j. line size
k. figures and tables
l. style
m. vocabulary

n. use of abbreviations and acronyms
o. use of logic terms
p. use of symbols, units, and numbers

Being directed at emergency procedures, NUREG-0799 provides a consensus that utilities can use to develop their own guidelines in order to upgrade their emergency procedures. Unfortunately, only minimal guidance is provided on the use of illustrations to present technical information.

The technical literature describes format principles based on the law of good form. These are:

a. Perception of differences. Differences are perceived if material contains contrasting elements, e.g., use of all upper case for control label and action draws attention to that part of the textual material.

b. Perception of similarity. Consistency is important, e.g., using the same abbreviations all the time.

c. Perception of nearness in space and time. Figures should be integrated with the text.

d. Facilitating understanding by organization of the perceptual field. Formats that emphasize the important points facilitate performance.

At least one vendor has performed some preliminary investigation of alternative formats for hard-copy-based procedures. By developing alternatives to the standard narrative format, improvements in presentation of information were compared. The vendor's concern is noteworthy, even though a large body of presentation techniques have already been documented and evaluated in the literature.

While most of the above have similar implications for computer-based displays, there are also unique format problems. To date, most of the technology has centered on using a CRT as the display device.

Danchak (27) proposed several coding schemes that would be effective for CRT displays, These include:

a. Numeric for digital strings
b. Alphanumeric for words
c. Shapes for symbols
d. Color and blink for redundancy and to reinforce the information.

Danchak goes on to provide some guidance on density of the displayed information. Density combined with format, coding, and rates of change are discussed for alphanumeric displays, graphic displays, and representational displays. Because character density is limited, Danchak recommends abbreviating words with known abbreviations, masking, or vowel deletion. Use of this technique without standardized guidance could lead to a plethora of abbreviations that may place too high a mental load for interpretation during periods of stress.

For CRT displays, several characteristics have to be considered when developing the format for a particular display (44). These include:

a. raster scan process - the method used to initiate an image on the screen;
b. refresh rate - the method used to hold an image on the screen;
c. refresh flicker - when an image is held on the screen it is not continuously illuminated, rather the image intensity will vary as a function of the refresh rate;
d. resolution - a physical quality of CRTs that is related to the number of lines possible, the number of characters possible, the number of pixels (matrix) that make up the character, and the number of the three color groups of data (triads);
e. character graphics - well-encoded information; and
f. bit-map graphics - pixel-encoded information (provides greater flexibility).

In a continuing program at Idaho National Engineering Laboratory, distinctive characteristics for CRTs have been identified. Incorporating the findings of Ramsey and Atwood's (126) extensive review of the human factors literature for computer system, Banks provided guidelines concerning the following format characteristics:

a. highlighting - use it for emphasis by increasing intensity, flashing, underlining, varying character size or font, using pointers, reversing image or boxing it
b. data presentation - use line drawings to supplement tasks, break alphanumeric strings into chunks of 3 - 4 characters, and use complete words
c. screen layout and structuring - use perceptual organizations, i.e., reserve parts for specific types of information, but tell user how it is organized. Avoid chopping the screen. Avoid overcrowding and run-on text. Always place directions first.

d. feedback to user - indicate selection user makes, provide explicit exit instructions, and avoid providing extraneous data
e. messages - present information in usable format, avoid requirements for transposing or mental translation, and avoid humor
f. interframe considerations - if scrolling, provide locational information, use lower and upper case for text, be consistent in meaning and context, as well as labeling

In a related study on color, Banks and Clark found that colors can be identified more accurately than sizes, brightness, familiar geometric shapes, and other shapes or form parameters. Alphanumeric symbols, however, are more accurately identified than colors. Given specific characteristics, color is a good way of coding information, but caution is advised. (See Section 2, this Volume.)

Hol, Ohra, and Netland (45) provided a review of the research appropriate to the design of illustrations for CRTs and the use of colors and symbols for CRTs. Colors can be used in process control to denote various conditions or states (on/off), various control modes (manual/automatic), and status. Subdued colors should be used. The optimal number of colors displayed at any one time should be limited to eight.

For CRT illustration design, information content must track with information philosophy, e.g., is the display going to be used to monitor task performance, or is it going to be used for safety monitoring? Blink and flash can be used as an attention-getter to denote urgency. Color can denote information types. Character size and font can be used as information carriers. Display complexity is hard to determine and is dependent on information density, arrangement of the information, frequency of use, viewer's training, and viewer's mental state (relaxed vs. strained, rested vs. fatigued). Due to the highly variable factors, care will have to be exercised to prevent displays that are overly packed with information.

Hol et al. provide the following guidelines for format design of alphanumerics, trends, bar graphs, and circuit diagrams:

a. alphanumeric format - use columns and rows; use colors consistently, e.g., if particular colors are used to denote alarm priority the same colors should not be used for any other information; otherwise, habituation may result
b. trend format - no more than four diagrams should be displayed at any one time; the color used for a curve or value axis should track with the color code used for alarms; if curves

go off scale, let them; otherwise, deviations from the norm may be missed

c. bar graph format - use horizontal bars; the number of bars should be a function of the information type; bars may have common scales, but individual scales are better; color should match scale

d. circuit diagram format - avoid skewed lines.

Hol et al. summarize their effort by noting that criteria should be established before the generation of CRT displays. The criteria should consider the role of the operator and the degree of automation required.

Danchak (26) developed a methodology for graphic displays. First, determine the data types. Data can be described by the number of independent dimensions (unidimensional, duodimensional, multidimensional), the number of variables (univariate, limited multivariate - five or less, multivariate - greater than five), and the number of samples (discrete, limited series - 25 or less, series - greater than 15). Second, determine the techniques that are available for the various combinations of data types. Tables are provided. Third and last, determine primary use. Use can be quantitative, qualitative (approximate value), check information (deviation from normal, normal, within range), status and warning, prediction, and pattern recognition.

PLANNED ACTIVITIES

The Office of Nuclear Reactor Regulations has identified specific research needs for human factors safety that are expected to be funded. One of these needs, plant procedures, has within it a subelement on format research. Specifically, the HFSD wants alternative ways of presenting procedures to operator and other plant personnel to optimize comprehension and response explored and tested.

In accomplishing these needs, the NRC expects to review and summarize contractor efforts for I & E and NRR related procedure development, and monitor the Halden work on its computerized display system for presenting procedures. In addition, INPO, working under contract to DOE, expects to generate criteria and guidelines for format and content of emergency procedures.

It appears that the research need requests a comprehensive effort to address specific format questions. When compared to the expected response, however, many of the questions will go unanswered.

The Office of Nuclear Regulatory Research intends to continue its effort at the INEL on augmented operator capability. For FY82 INEL will (1) complete a literature search on the effects of changes in control room modifications on operator performance,

(2) investigate the applicability of advanced graphic display concepts for assisting operators during accidents, and (3) define the functional requirements of a research facility to evaluate information presentation devices. A related INEL effort will evaluate four potential formats for SPDS - deviation bar diagram, circular plot or star diagram, clustered meters, and trend plots.

MISSING ELEMENTS

There is no question that a lot has been done in the use of format, both for hard-copy-based JPAs and procedures and for computer-based JPAs. The missing element is user acceptance. Many of the format techniques presently under development must have as their ultimate criterion acceptance by the user. Thus, all the on-going efforts should insure that the resultant innovation in the presentation of information is accepted by the user. With the user acceptance criterion satisfied, guidelines for various format alternatives can be established.

Another effort that should be investigated if a computer is going to present procedural information, is the presentation of information in a multitrack level of detail that would appeal to a wide range of users. Through the various task analytic techniques required for the development of proceduralized job information, multilevel detail tracks can be developed. Preliminary work in the Air Force and Navy have demonstrated that, from a single source of job information, multitracks can be developed that allow the user to pick the level of detail required to perform a particular task. The features of this type of computer-based display system are:

a. Completely validated and verified data source.

   Validated means the procedures have been performed on the equipment or a simulator to ensure that the technical content is complete and accurate.

b. From the data source, develop three tracks, i.e., three different levels of detail that cover an extremely wide range of users (three is picked as an illustrative example).

   (1) The most detailed track would be a mixed text/graphics format.

      o The text would be a complete step-by-step procedure.
      o The graphics would be keyed to the text with callouts.

- o The graphics would supplement the text by providing the references for the procedural step.
- o Zoom and pan capabilities would be incorporated into the graphics display.

(2) The next level of detail would assume more experienced users.

- o The textural information content would be greatly reduced, e.g., a valve alignment procedure written for the more detailed track would list the valves by name with specific locational information and content, whereas, for this second level of detail, the task step may just state to align the identified valves with the assumption being that the valve locations are known.
- o The use of illustrations would be greatly reduced.

(3) The third level of detail would abbreviate the procedure to a checklist format.

- o The test would be greatly condensed and summarized with the assumption that the user would know what and how to do all the required substeps, when a topical task identifier is used.
- o Illustrations would probably not be used.

c. In addition to the procedural information, additional data pools would be available to the user, such as P & IDs, functional schematics, illustrated parts breakdown, etc.

- o The user would easily be able to call up this supplemental information if he required it.

d. The three tracks themselves would be interactive so that the user, who may start out using a less detailed track, would be able to request more information about a step or task at any time.

e. The user would determine the level of detail he requires before starting a particular procedure. In other words, he would tailor the level of detail of the procedure to his skills and experience.

f. Access time to procedural information would be greatly reduced.

g. The interactive nature of the system should be reinforcing to allow for better use of the available technical information.

TECHNICAL FEASIBILITY AND PROBLEMS

The technical limitations will occur in the area of computer technology and display characteristics. Enough research has been done to provide the nuclear power industry with good formats for the presentation of information. The problems arise when trying to transfer the hard-copy-based techniques to a computer-based display. The presentation device, whether paper or computer, has to be readable, understandable, and acceptable by the user.

The commitment of large financial resources for reorganizing the formats of procedures may be a deterrent. Before changing the format for every procedure, however, a plan should be developed to determine a cost-effective way to apply the various format techniques.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Format has to be correlated with level of detail which, in turn, has to correlate with the personnel requirements, i.e, the user description. Assumptions about the skills and knowledge of the user population will have an impact on the level of detail required for the development of procedures and job performance aids.

Of course, format will have its greatest impact on the procedure development and JPA development requirements. Because the procedures serve as source documents, a wider range of formats is available for the job performance aids.

RECOMMENDATIONS

Review and evaluate existing formats for applicability. Determine (1) alternative choices of information presentation techniques, (2) format limitations of CRT and of computer-based displays, and (3) feasibility of using CRTs for the presentation of procedures. Develop guidelines describing acceptable JPA formats and delineating validation/verification and implementation procedures.

Importance: High

Schedule: Urgency - begin in 1-2 years

Duration - complete in 3 years

Resources: Research will require 10-12 person-years with access to computer driven CRTs, part- and whole-test simulators, and math models of reactor system. Guidelines can be developed as an NRC staff function.

Implementation: Personnel skills will require career human factors professionals, system analysts, programmers, technical criteria, graphics illustrators, and subject matter experts.

Dependencies: Coordinate with on-going efforts by NRC, INPO, and EPRI.

3.5 Implementation and Revision Procedures

REQUIREMENT

The application of the systems approach to procedure development will provide validated technical information developed from task analytic data. Incorporation of a good format for clear, unambiguous presentation of concise information will satisfy the need of the user population. But, the development effort will have been useless if a mechanism is not established for implementation.

Implementation requires plant administrative controls on the use and application of procedures. Plant administrative procedures have to be developed that provide guidance on how procedures are tracked after they are developed.

In addition to the operation and maintenance procedures that are developed internally, the nuclear power plant has many vendors who supply manuals for the various mechanical and electrical components that require maintenance. With so many vendors, the level of detail and format of the manuals vary from vendor to vendor. Thus, the utility is faced with a problem -- how to incorporate these manuals as procedures, i.e., should the utility use the off-the-shelf manuals or rewrite the manuals to a single format and consistent level of detail?

If no attempt is made to integrate the various maintenance manuals into some type of control scheme, the manuals will probably be filed away and never used. If used initially, the dissimilarities across vendors may be so great that the users perceive the manuals as useless and refuse to use them. At a minimum, these disparate manuals have to be cataloged and indexed so they can be tracked for updates and corrections.

In procedure implementation there is also the problem that administrative controls are not adequate for the location and use of the operating procedures in the control room. Procedures are not always located in a centralized and easily accessible place. In some dual unit control rooms, for example, only a single set of procedures is provided and placed in a location that is not easily accessible to operators from either unit. In-use procedures, e.g., a start-up, are sometimes read by the shift supervisor to the operators performing the task. Unfortunately, this practice can lead to the supervisor abbreviating the written statements, and/or the operators grouping instructions out of sequence if the supervisor does not wait for confirmation of each action before proceeding (e.g., for convenience, since two instructions may refer to the same panel, while the one in between directs the operator to another panel, the operator or the supervisor may group the first and third instructions together). With a definitive set of administrative procedures supported by a strict quality assurance program, the above problems can be solved.

A procedure or operator aid that is the product of a system approach and presents the information in an unambiguous, concise format will continue to be used only if needed revisions to the technical content are added. Even procedures and operator aids that were not submitted to a rigorous development process can be revised and updated while currently in use. Revisions and updates not only ensure the continued quality of the technical data, but also improve the acceptance of the product by the users.

Revisions and updates are not necessarily incorporated as part of the procedure development and implementation process in a systematic manner. Many procedures are revised and errors are corrected, but a large number of procedures contain errors that either go undetected or are not submitted for revision.

While it appears that the procedures in the control room are fairly well controlled by centrally locating the procedures, the maintenance procedures are a different story. Since maintenance personnel tend to specialize, some develop their personal technical data base. These "personal" copies are usually embellished by the technician to tailor the procedure to his own needs. The problems with "personal" copies is that when major revisions or updates are made, the revisions are not incorporated into the "personal" copies.

The revision process has to be able to determine:

a. when there are changes to equipment and specification limits,
b. when equipment is replaced by modified versions,

c. when performance verification in the field identifies an error in a procedure or job performance aid,
d. when procedures and job performance aids are obsolete, and
e. when procedures and job performance aids, in particular drawings, become worn and faded, and details are hard to detect.

CONSTRAINTS

Technically, the development of administrative controls for the implementation of procedures is not difficult; the continued enforcement of these controls is. In the control room alone, the emergency operating, the normal operating, the testing, the calibration, and the surveillance procedures occupy 5-10 feet of shelf space. Thus, enforcement of administrative controls is a monumental task. Quality assurance has to determine that new procedures are correctly incorporated into the existing documents and incorrect or outdated procedures are removed.

In maintenance, the enforcement problem is compounded. The technical manuals will be as numerous as they are varied. Even if the utility determines that changing the format of the vendors' manuals is too costly, the manuals can be cataloged and stored in a central location. Then before initiating a task, the maintenance technician can, with the aid of P & IDs and plant schematics, determine what component is going to be worked on and review the appropriate vendor's manual prior to performing the task. Administrative controls that do not provide for the establishment of a well-organized central storage facility, and that weakly enforce the requirement for preview before task initiation, will affect overall plant availability because simple procedural errors will be made.

A sign-off requirement incorporated into each procedure may contribute to the procedure being used. The requirement can be single sign-off at the end of a sign-off at each step. Such a requirement, however, will probably be rejected by the users as an unnecessary, burdensome task. Periodic quality assurance checks during normal operations and maintenance might facilitate use. Of course, the best way to ensure use is to provide the user with technically correct and clear, unambiguous procedures.

Procedures that are not controlled by means of a centrally located documentation center will be hard to track for revisions and updates. To be effective, a central control facility has to be able to track all issued procedures and know how the procedure and job performance aids are interrelated. For example, if there is a control room hardware modification that results in the deletion of some components, a good tracking capability would result in the identification of all affected procedures.

The administrative control procedures have to have an effective method by which operators and technicians can provide feedback to the development process for the purpose of revision. The method has to be viewed as simple and straightforward by the users. If it involves a lot of form-filling, few requests for revisions or updates will be submitted. On the other hand, processing of requests have to be completed in a timely manner. If the management review cycle is overly long, the users will view the process as ineffective.

PRESENT STATUS

One effort has looked at the updates and revisions process German nuclear plants, and found updates to be a problem because when a plant component is modified, changes to the documentation are sometimes left for "another time," i.e., after the modification has been completed, more urgent tasks are awaiting. Thus, revising the procedures is left until more time is available.

It is important for plant management to take the lead and establish updates as an important goal. Management needs to demonstrate and discuss the consequences of undesired behavior for a given procedure. The advantages of desired behavior should be stressed.

The NRC has concentrated on feedback from operating experience, in general, and how this feedback should be reflected in procedure revisions. The NRC's action plan (NUREG-0660) identified item I.C. 5 as relating to this process. The purpose of this feedback is not solely for upgrading procedures, but also should be incorporated into training and retraining programs.

NUREG-0737 clarifies the action item and states that each license applicant is required to prepare procedures and continually supply this information to operators and other plant personnel by:

a. Identifying organizational responsbilities for review of feedback.
b. Identifying the process to review and incorporate the information into plant procedures, etc.
c. Identifying recipients of the various information categories.
d. Providing a means to ensure affected personnel are aware of the information applicable to them.
e. Assuring that extraneous information does not obscure the pertinent information.
f. Providing checks that conflicting or contradictory information is not supplied to plant personnel.

g. Providing periodic audits to assure effectiveness of feedback processing process.

There are many sources for the feedback data. These include LERs, NRC bulletins, circulars, notices, and industrial assessments. One source that appears to have specific input for procedure updating is the analysis of selected incidents by NSAC and INPO. For example, the analysis of incomplete rod insertion at Browns Ferry 3 (56) found that the procedures were not broad enough to cover the actions required to recover from this incident. Similarly, the analysis of the Crystal River Unit 3 incident (55) identified specific procedures that needed revision.

PLANNED ACTIVITIES

The Licensee Qualification Branch is trying to work with INPO and NSAC in allowing INPO/NSAC to screen all the data that are generated as possible feedback to determine what is applicable to a particular task. LQB will then notify the utilities of the INPO/NSAC services and to what extent the screening process satisfies I.D.5. Finally, criteria for the minimum acceptable utility program to totally complement I.C.5 will be developed, and will be incorporated into the standard review plan.

MISSING ELEMENTS

Procedure implementation requires plant administrative controls. It also requires review of the existing guidance (e.g., ANS 3.2) to determine if the administrative procedures governing implementation, tracking, and revision of procedures are adequate. Without a standardized set of guidelines, compliance will be difficult to determine.

Administrative controls have to address storage and use. Each plant has to devise storage techniques for both the operators and maintainers that are convenient and do not interfere with normal operations. When procedures are used, e.g., plant start-up, the administrative controls should state a specific scenario to follow. If the use of the procedure is a team effort, the specific roles, e.g., reader, performer, checker, etc., have to be delineated. This incorporation of a reader for the procedure would eliminate the problem of finding a place to put the procedure when performing the activity. Determining compliance with the above requirements will also have to be investigated.

Similar administrative controls for maintenance also have to be evaluated. In maintenance, the applicable job performance aids and procedures can be included as part of the tools and equipment needed for a task. A maintenance management system that defines all the preventive and corrective maintenance tasks, and identifies as part of each task the tools and documentation needed to perform the task, would help assure procedure implementation in the maintenance area.

TECHNICAL FEASIBILITY AND PROBLEMS

The enforcement of administrative controls for procedure implementation and revision will be a problem. Unless the users, both operators and maintainers, can be made aware of the importance of using procedures and providing feedback when errors are encountered, they will not use the procedures and job aids as intended.

The NRC has the capabilities to evaluate existing plant administrative procedures relating to implementation and revision. The constant intrusion of the NRC, however, may have the utilities developing administrative controls just to satisfy NRC regulations and requirements. The NRC's role should be one of approval for guidelines that will aid the utilities in developing effective implementation and revision procedures. Once implemented, the NRC's role should be one of audit to determine compliance.

The development of an automated updating system may be a difficult undertaking. Software would have to be defined that would describe the relationship within and between procedures. If the procedures are not included as part of the data base, an elaborate indexing scheme would be needed to ensure that all the references to other procedures and branching paths are accounted for.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Requirements for administrative procedures will be identified during the systems analysis and subsequent task analysis for administration. Since the administrative controls are what guide the procedure development process, the administrative controls will have to be developed in parallel with the operation and maintenance procedures.

RECOMMENDATIONS

The NRC should determine an effective process for implementating and revising plant operational and maintenance procedures. Develop requirements for an information management system that will (1) index and cross-index all plant procedures to ensure that all changes are incorporated into the affected procedures and (2) track procedures (a) to ensure that all procedures are distributed and recalled, if necessary, in a timely manner and (b) to ensure operational feedback data from within and outside the plant are incorporated into procedure revision, when necessary.

<u>Importance</u>: Low, but desirable

<u>Schedule</u>: Urgency - begin in 6-10 years

Duration - complete in 1 year

Resources: Research should required 1-½ - 2 person-years

Implementation: Personnel skills will require a management information specialist and a career human factors professional.

Dependencies: None

3.6 Performance Verification

REQUIREMENT

Performance verification is necessary to ensure that critical tasks and safety procedures are performed correctly. Because of the size of the nuclear power plant, the similarities within the plant, and the numerous components and systems, verifying correct location becomes an important safety aspect of the procedure. Incorrect location and/or identification of the component could lead to an inadvertent violation of the Technical Specification and to personnel injury if the technician performs a task on an operating component. The process for verifying that equipment is properly tagged-out before starting a maintenance task also has to be delineated.

CONSTRAINTS

Independent performance verification for all tasks is impractical, e.g., performance verifying reseating of a valve would require another strip-down of the valve and reseating. Where possible, in particular the preliminary procedures of tag-out and location verification and the independent performance verification process should be established.

PRESENT STATUS

One recommendation of the NRC's lessons learned task force (NUREG-0585) was the verification of correct performance of operating activities. Specifically, all normal, maintenance, task, and surveillance activities should be reviewed to assure adequate description and documentation. Once reviewed, the personnel responsible for the verification and the method of documentation have to be specified.

Item I.C.6 in the NRC's action plan (NUREG-0660) described the NRC's role. The Office of Nuclear Reactor Regulation is responsible for ensuring that licensees' procedures be reviewed and revised, where necessary, to include performance verification requirements. Performance verification can be either automatic system status monitoring, or independent human verification. The overall objective of this action item is the reduction of human error and the improvement of the quality of normal operations.

Item I.C.6 was clarified in NUREG-0737. The clarification states that a revision to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" will endorse ANSI Standard N18.7-1972 (ANS 3.2), "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Additional provisions of Regulatory Guide 1.33 state:

a. Control measures will include surveillance testing along with maintenance.
b. Release authority will be extended to an on-shift SRO.
c. Tag-out procedures will be verified by qualified personnel. (Deleted in cases of significant radiation exposure).
d. Control operators will be informed of equipment status changes and effects of changes.
e. System alignment will be verified by a second qualified operator for return-to-service unless functional testing does not compromise plant safety and will verify alignment.

Human verification can be reduced if automatic safety system status monitoring is added to the plant (Item I.D.3).

The only ongoing activity in industry on performance verification is Philadelphia Electric's Critical Equipment Monitoring System (CEMS). This system uses hand held terminals that are radio linked to a central computer system. A voice channel is also available for communications. An attached optical scanner reads permanently attached bar codes for equipment identification. Once read, the equipment is checked against last known status, plant mode of operation, percent power, etc. The central computer contains a data base in the form of data flow diagrams and system descriptions. Among the capabilities of CEMS are the following:

a. It will help assure correct identification of equipment components by using portable bar code readers, i.e., the information will be transmitted via radio to the computer for verification.
b. The procedures will be cross-indexed with the components so that all information that pertains to a particular equipment component will be readily available.
c. Tag-out procedures will be easily verified.
d. Quality assurance checks of system alignment will be enhanced.
e. Performance verification will be automatic.

PLANNED ACTIVITIES

NRR's technical assistance contract for FY82 with Battelle Pacific Northwest Laboratory requires the development of techniques that will ensure that license applicants meet the acceptance criteria established for task action item I.C.6. The requirement for task action item I.D.3 will also be reviewed for coordination with I.C.6. The Philadelphia Electric effort is also continuing.

MISSING ELEMENTS

For performance verification, the feasibility of applying a computer-based system like Philadelphia Electric's CEMS should be investigated. Administrative controls should include a method for determining which tasks or parts of tasks have to be verified. The requirements for performance verification should be incorporated as an integral part of the procedure or operator aid.

TECHNICAL FEASIBILITY AND PROBLEMS

Performance verification will be viewed by some plant personnel as "checking up," which will certainly have an effect on morale. A problem arises when the verification process finds an error, i.e., should action be taken against the individual who performed the procedure.

Finally, performance verification will only verify the performance related to the procedures at hand. If, by chance, a valve is turned that is not related to the procedure, performance verification will not find it.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Performance verification is an integral component of the administrative controls for quality assurance.

RECOMMENDATIONS

The NRC should conduct a study for development of a reliable automatic system status monitoring device that will provide information on (1) valve and switch positioning upon completion of surveillance, test, calibration, and standard maintenance tasks, (2) completeness of tag-out procedures including removal upon task completion, and (3) inadvertent violations of the technical specification.

Importance: Medium

Schedule: Begin in 6-10 years. An interim system could be developed within 1 year with the effort continuing as long as reliability can be improved.

Resources: Interim effort will require 5 person-years with access to state-of-the-art computerized monitoring devices.

Implementation: Personnel skills will require career human factors professionals, system analysts, programmers, nuclear engineers, and knowledgable plant personnel as subject matter experts.

Dependencies: None

3.7 Change-of-Shift Procedures

REQUIREMENT

During a shift at a normal operating power plant and, in particular, during a refueling outage, many tasks are performed. For the operations staff, surveillance, tests, and calibration procedures are completed. For maintenance, both preventive and corrective procedures are completed. In addition, many tasks require time periods that are longer than an 8-hour shift.

For the control room operator who is trying to produce electrical power from a nuclear source, it is important to know the status of the plant. To operate the plant effectively and safely the operator must know the status of all systems within the plant. This requirement imposes an extensive information load on the operator.

Since the nuclear power plant is operated on a 24-hour basis, it is imperative that the operating crew be fully aware of the status of the plant. Given a single crew, continuity of plant information could be maintained. Unfortunately, a single crew cannot operate the plant around the clock, i.e., shifts, usually 8 hours long, are used to staff the control room. Thus, corporate memory cannot last longer than the shift duration unless some device is used to transfer the information from the out-going shift to the in-coming shift.

Usually during a shift change information is exchanged, regardless of type of work. The exchanged information can range from a simple exchange of greetings to a complete rundown of what occurred on the previous shift. In critical industries, such as the operation of a nuclear power plant, the information exchange has to be systematic, precise, and detailed. A shift turnover procedure that is integrated with plant status modes would help prevent incidents such as the One in which operators at TMI were not aware that both emergency feedwater block valves were closed. Well-defined shift turnover procedures are not new. Most medical institutions are maintained on a 24-hour basis and always require a 20-30 minute overlap of shifts to allow the outgoing shift to report the critical information to the incoming shift. The safe operation of a nuclear power plant is no less critical.

During any shift, the operating and maintenance crew needs to document all actions that may have an effect on the status of the plant. The oncoming shift needs to review all these actions and resolve any questions before the outgoing shift departs. A detailed checklist that is based on the plant's technical specification and, in particular, the engineered safety functions, would facilitate this exchange of information. A walk-through of the plant controls would also help.

CONSTRAINTS

There should be no problem in defining a set of shift relief and turnover procedures. For example, most medical institutions have 8-hour shifts with ½ hour for lunch, for a total of 8½ hours. Thus, there is a 30-minute overlap during which the outgoing shift briefs the incoming shift. In the nuclear power plant, however, the union and management strongly influence administrative practices. In addition, unlike their medical counterparts, the control room operators are usually required to work during their lunch break.

In a plant where the union is strong, the overlap period is construed as mandatory overtime, an aspect management does not tolerate too well. On the other hand, a non-union plant may not elect to use the ½-hour overlap for the same reason. Also, the ½-hour overlap example may create morale problems for the shift workers that do not require the overlap of time to exchange critical information.

Another constraint to good shift relief and turnover procedures is that the process may become completely perfunctory if the procedures were not systematically developed. An excess of checklist items that are always in the same status during a particular mode will lead to items being overlooked even if a sign-off procedure is incorporated.

PRESENT STATUS

Among the early recommendations from the TMI accident is the improvement of administrative procedures. Specifically, NUREG-0578 recommended that plant procedures be reviewed and revised to assure that a shift turnover checklist is developed and used during the change of shifts. Supplementary checklists and shift logs were also recommended for non-control room personnel.

The NRC recognized that the version of Regulatory Guide 1.33 available at that time did not provide any guidance as to the content for a shift turnover checklist. To correct this deficiency, NUREG-0578 required the utilities to develop (a) a checklist for the off-going and on-coming control room operators, and signed by the on-coming shift supervisor, (b) supplementary checklists or shift logs for off-going and on-coming auxiliary operators and technicians, and (c) a system to evaluate the effectiveness of the procedure. Implementation of these requirements was performed through a series of letters to the utilities (September 13, 1979; September 27, 1979; October 10, 1979; October 30, 1979; and November 9, 1979). In addition, plant visits were conducted to review the required changes to the administrative procedures and quality assurance NUREG-0694 reiterated the requirements for new operating licenses.

With the issuance of NUREG-0660, the NRC formalized specific task action items. Item I.C.2 addresses shift relief and turnover procedures. Since revised shift turnover procedures were required by the short-term lessons learned from TMI, NUREG-0660 recognized the item as complete.

The July 1981 revision of the Standard Review Plan (NUREG-0880) required specific procedures for shift relief and turnover. The acceptance criteria used to evaluate the license applicant's administrative procedures include ANS 3.2 as augmented by Regulatory Guide 1.33 and the task action plan item I.C.2 of NUREG-0694. In essence, the Licensee Qualifications Branch evaluates the utility on its commitment to produce specific shift relief and turnover procedures and IE evaluates the procedures after they have been implemented.

Although change of shift procedures have always been required, it was not until TMI that serious consideration was given as to the content and structure of those procedures. Shift relief and turnover checklists are common and contain three key items:

(a) Critical plant parameters and their allowable limits.
(b) Status of systems essential to the prevention and mitigation of accidents, including auxiliary feedwater system and high pressure injection system.
(c) Identification of systems and components that are in a degraded mode of operation.

TMI may have highlighted the need to correct inadequate change of shift procedures, but the identification of the problem is not new. A review of Swedish nuclear power plants conducted from 1977 to 1979. Recommended a change of shift checklist to include a log of all the system components with completed periodic tests, on-going work, deviations from normal values, permanent changes, problem areas, and general notes.

Change in shift procedures have been improved. Unfortunately, the lack of a standard or specification detailing what is required, and the specific format and content of checklists, have led to inconsistency across level of detail. One utility requires journal entries throughout the shift and requires the outgoing shift to review all significant entires. The oncoming personnel have to review all journal entries back to the time of their last shift. Three checklists are provided, one for transfer of responsibility on which significant activities have been noted, another requiring complete verification of all listed comments by the oncoming shift supervisor (list contains bulb and alarm checks as well as extensive verification of systems, valves, and components), and finally, a checklist to verify critical valve alignments. On the other hand, some utilities require a walk-through by the oncoming personnel, but the provided checklist, completed after the walk-through, appears to be nothing more than a formality.

PLANNED ACTIVITIES

The NRC stated that the recommendations of task action plan item I.C.2 has been completed and there are no known planned research efforts in this area. Revisions to ANS 3.2 and Regulatory Guide 1.33, however, are scheduled to be published. Together, these require checklists, shift overlap to permit a walk-through by the oncoming shift, and a plant tour of ongoing maintenance and surveillance testing during the shift.

MISSING ELEMENTS

The pieces are all there. It is the compilation and the consistent application of them that is lacking. The NRC has taken the lead, (Regulatory Guide 1.33, Revision 3, 1981), and has defined what the change of shift procedures are. It needs to complement this by detailing the content of required checklists.

The NRC could provide good examples and alternatives, if appropriate, as a supplement to Regulatory Guide 1.33.

In the health field, the shift turnover procedure that is followed is:

a. Shifts always overlap. Shifts are 8½ hours long - 8 hours work, ½ hour for lunch. Each shift overlaps the previous one by ½ hour.

b. Outgoing shift: personnel record on tape all the pertinent informaton and happenings of the shift. The recording is made just prior to the end of the shift but before the new shift reports.

c. Oncoming shift: personnel listen to the tape upon first reporting and ask any questions while the previous shift is still on duty.

TECHNICAL FEASIBILITY AND PROBLEMS

Development of shift relief and turnover procedure guidelines are well within the present capabilities of the NRC.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Procedures for shift relief and turnover are developed as part of the administrative procedures. And, since the change of shift is a specific task, the requirements are identified during the function allocation and task analysis.

RECOMMENDATIONS

The NRC should establish criteria for effective shift relief and turnover, and develop requirements for checklists and procedures, such as walk-throughs and log reviews for both operational and maintenance personnel.

Importance: High

Schedule: Urgency - begin in 1-2 years

Duration - complete in 1 year

Resources: NRC staff function

Implementation: Required personnel skills include a career human factors professional and nuclear engineer.

Dependencies: None

4.0 Personnel and Staffing Problems

INTRODUCTION

The fundamental requirement for personnel and staffing is to ensure initial and continuing quality control of all categories of plant personnel. Personnel and staffing demands are driven by the requirements for operational quality assurance, maintenance effectiveness, safety standards, and effective management for both normal and emergency conditions. In addition to extensive interaction with plant design and training, the personnel and staffing area is impacted by selection criteria, operator qualification and requalification standards, examining procedures, shift duration and rotation practices, performance assessment and feedback practices, and a variety of factors that constitute the reward system. Each of these system characteristics has received considerable attention by the NRC and industry since Three Mile Island.

4.1 Personnel Selection - Practices and Standards

SYSTEM REQUIREMENTS

The purpose of personnel selection is to generate adequate numbers of operator, maintenance, and other plant specialists to ensure that all staffing requirements can be met with personnel who have (1) appropriate aptitudes for efficiently learning and properly performing the job and (2) appropriate temperamental characteristics, including emotional stability, for coping with both the tedium of routine nuclear power plant operations and the stress associated with occasional accident events. A corollary of these requirements is that the selection procedures should identify personnel who will develop a career interest in nuclear power plant operations, in the interest of minimizing personnel turnover.

CONSTRAINTS

There are at least three basic constraints associated with personnel selection practices in the nuclear power industry:

1. There is, at least in some geographic areas, an apparent shortage of personnel who are capable of meeting the licensing criteria for ROs and SROs. This constraint would probably be more severe if it were not for the availability of significant numbers of Navy-trained nuclear power technicians to the industry.

2. Selection procedures must, in view of current law, have predictive validity for actual job performance. Yet, in the case of nuclear powerplant personnel, this is difficult to demonstrate because objective, quantified measures of on-the-job performance against which to validate selection instruments and procedures are lacking.

3. In some cases, labor-management contracts incorporate mandatory consideration of such factors as seniority in establishing eligibility for candidacy as an RO; this may limit the effectiveness of performance-relevant selection procedures.

PRESENT STATUS IN MEETING REQUIREMENTS

NUREG/CR-1750 points out that the selection of candidates for RO training is a two-step process in the nuclear power industry. A variety of selection criteria are employed for qualifying a candidate for initial employment. Additional criteria must be satisfied when the potential operator is selected to enter the RO training program. Various combinations of the following procedures are used during initial screening: medical examinations, interviews, background check, aptitude and achievement testing, personality inventories, and technical examinations.

In addition, where candidates are drawn from unlicensed personnel pools, "performance evaluations" and seniority are used as selection criteria in various combinations.

Perhaps the two most important observations that can be made with respect to currently used initial screening procedures are (1) the practices vary widely among different utilities, and (2) the validity of the procedures for predicting operational performance remains uncertain.

The "American National Standard for Selection and Training of Nuclear Powerplant Personnel" (ANSI/ANS-3.1-1978) stipulates the following criteria for selection of operators who are to be licensed.

1. High school diploma or equivalent.

2. Two years of powerplant experience, one of which must be in a nuclear powerplant. Six months of the nuclear experience shall be at the plant for which the operator seeks a license or on a similar unit. If prior technical training or experience warrant, six months of the experience requirement may be credited for that training or experience.

3. High degree of manual dexterity.

4. High degree of mature judgment.

A proposed revision to this standard would increase experience requirements to three years of power plant experience of which one must be at the nuclear power plant for which the operator will hold a license and, in addition, require six months of duties as a nonlicensed operator.

The NRC Office of Standards Development has published (May 1980) a proposed revision to 10 CFR, Part 55, "Operators Licenses." This proposed standard requires that the license applicant hold a high school diploma or General Education Development Program Certificate and endorses the proposed revision to ANSI/ANS-3.1-1978 (see above).

From this, it may be concluded that the NRC's criterion for acceptance of personnel into candidacy as operators lies less with initial selection procedures and more with the extent and type of employment experience prior to application for licensing. Indeed, research on initial selection criteria and procedures appears to have been the exclusive province of industry.

Industry has clearly benefited from and recognizes the value of selecting operator personnel through prior service in the nuclear Navy. One plant visited made a point of selecting their entire initial cadre of operators from ex-Navy nuclear propulsion plant personnel. Since such personnel have been prescreened, both in terms of aptitudes and job experience, there is little doubt that this has been an effective selection procedure for the industry. However, the available pool of ex-Navy personnel has not been adequate to meet industry's need in many parts of the country and the requirement for effective, more general conventional selection procedures is well recognized.

By far the most comprehensive research and development effort aimed at improving the selection process has been that performed by the Personnel Decisions Research Institute (PDRI) under contract to the Edison Electric Institute. This study addressed personnel selection for a variety of plant operating personnel, including control room operators in hydroelectric plants, fossil plants, and nuclear plants.

PDRI reviewed a substantial number of earlier industry efforts to develop valid selection procedures and concluded that most of these prior attempts were technically deficient (47). PDRI used established job description techniques and an information processing model of operator requisite knowledges, skills, and abilities to develop candidate selection tests hypothesized to be related to several dimensions of operator effectiveness. In addition, a total of 600 to 700 "critical incidents" were collected from industry, reflecting both good and poor operator performance. Some of these incidents involved aberrent job behavior, reflecting what was judged to be a lack of emotional stability or suitability for the job. The identified job performance categories include:

Systems comprehension

Response to critical or emergency situations

Maintaining standard operations (monitoring, inspecting, testing)

Administrative record keeping (maintenance requests, logs, etc.)

Informing others of needed information (peers, superiors)

Relationships with co-workers (cooperation, getting along)

Coping with job circumstances (acceptance of regulations, authority)

A wide variety of candidate selection tests was administered to job incumbents which reflected the investigators' hypothesis concerning the cognitive, motivational, and personalogic requirements for successful performance on these dimensions (Table 5).

The validation criteria were carefully collected supervisory ratings on nine-point rating scales, reflecting the performance dimensions listed earlier. Several composite criterion scores were formulated, including Emotional Stability, Operations Competence, Problem Solving Ability, and Overall Performance.

Despite the evident methodological thoroughness of this study, the general finding was that, for nuclear power plant operators, the predictive validity of the experimental selection tests for the several performance criteria was, at best, very modest. No predictive validity was found for the emotional stability criterion. Most other validities were quite low (circa .20) with the exception of the measure of Problem Solving Ability, for which the best predictors had correlations of about .30.

The predictive power of the test battery could, of course, be increased by using several tests in combination assuming that the results hold up under cross-validation. However, several questions arise as a result of the quite modest predictive relationships found:

TABLE 5

Qualities Measured by Tests and Inventories Chosen For Inclusion In The EEI Plant Operator Experimental Battery (29)

1. Numerical aptitude
2. Spatial visualization (three dimensions)
3. Speed of perception and accuracy (detail oriented)
4. Reasoning ability (inductive reasoning and deductive reasoning)
5. Knowledge of mechanical principles
6. Fluency of ideas for problem solving
7. Verbal ability
8. Attentional selectivity (field independence)
9. Spatial memory (visual screening)
10. Reading comprehension
11. System comprehension
12. Care and accuracy in following directions
13. Sociability
14. Leadership orientation
15. Freedom from anxiety
16. Playfulness
17. Self control
18. Acceptance of routine
19. Adjustment to shift work
20. Willingness to accept authority
21. Defensiveness
22. Psychopathy
23. Impulsiveness
24. Dependability/conscientiousness
25. Sleep/wakefulness physiology
26. Habits of forgetfulness
27. Absorption
28. Risk taking orientation
29. Emotional maturity
30. Hard work/accomplishment
31. Confidence/self esteem
32. Interest in things/ideas (e.g., practical, scientific, artistic interests)
33. Changes in life circumstances
34. Check scales to detect inattention in completing tests, effort to "look good," and deliberate random responding

1. Would the same selection tests be identified as candidate predictors if detailed task analysis data (as opposed to the more general job descriptive data used by these investigators) were used to define the job dimensions?

2. Were the relationships generally attenuated because of restriction in range due to other selection processes? (There is some evidence that this may be true. For example, PDRI reports (personal communication) that powerplant operators possess superior mechanical comprehension and are clearly above the average high school graduate on the measures of verbal and abstract reasoning and numerical ability.)

3. Would the results be appreciably different if objective job performance measures were used as criteria instead of, or in addition to, the supervisory ratings?

While the validity of supervisory ratings, properly collected, is probably substantial for some important dimensions of job effectiveness, research in other contexts has shown the correlation between supervisory ratings and objective measures of technical performance to be uniformly low. This probably means that the two types of measures are best considered complementary aspects of some more comprehensive criterion of job performance. Further work is clearly needed before firm conclusions can be reached concerning these issues.

NUREG/CR-2075 is concerned with the development of standards for the assessment of emotional instability in applicants for a nuclear facility position. Originally directed only toward security personnel at nuclear power plants, the findings of this study were later generalized to recommendations for all nuclear power plant personnel. It is concluded that emotional instability is a multidimensional concept and that no single instrument is, by itself, capable of measuring emotonal instability. It is further noted that few studies have been conducted in a nuclear setting aimed at determining the predictive validity of various selection instruments for emotional stability.

ANSI N546-1976 outlines health requirements and disqualifying conditions applicable to nuclear facility personnel requiring operator licenses. This standard is intended to apply to requirements for both initial selection and the continued monitoring of licensed operators. The provisions of this standard state that an established history or clinical diagnosis of any of the following conditions constitutes disqualification of the applicant or employee:

1. Any psychological condition which could result in impaired alertness, judgement, or motor ability
2. A personality disorder severe enough to have been displayed by overt actions
3. A past suicide attempt
4. A history of psychosis
5. Alcoholism
6. Drug dependence
7. Presence or history of any other clinically significant psychological disorder in which the condition or its treatment could hamper safe performance of all operator duties (NUREG/CR-2075)

10 CFR Part 55.11 states that an applicant for an operator position must not show evidence of any medical disorder which might cause inadequate performance of required job duties. The specific disorders related to emotional instability which would result in disqualification include "insanity or any other mental condition which might cause impaired judgment or motor coordination."

Following a detailed analysis of traditionally used psychological and psychiatric screening procedures, and an assessment of the reliability, content validity, construct validity and "criterion-oriented" validity of various psychological screening devices, NUREG/CR-2075 concludes that for positions of considerable on-the-job stress, the selection system should include use of the Minnesota Multiphasic Personality Inventory, the 16PF (personality factor) Questionnaire, a clinical interview and, in the case of some positions, situational simulations. These recommendations are made in the absence of any empirical evidence of the validity of these screening techniques in the nuclear power industry. The authors conclude, however, that criterion-oriented validity studies should be carried out to identify relationships between various predictors of emotional instability and behavioral indices of emotional instability on the job. It is recognized that it will be necessary to develop criterion measures of on-the-job performance emotional stability before this can be accomplished. It is noted that such validation studies have not been particularly successful for the classes of personnel who perform under high stress conditions (e.g., air traffic controllers, pilots, and law enforcement officers). It is concluded that researchers in these areas have not been able to develop instruments which are reliable and valid predictors of stability as it relates to on-the-job performance and, more importantly, that criterion measures of emotional instability on the job have not been identified.

To accomplish the needed validation studies, it is suggested that use should be made of situational simulations which would "approximate" the specific elements and conditions surrounding key positions which would contribute to the manifestation of emotionally unstable on-the-job behaviors. It is also recommended that effort be directed toward developing and implementing on-the-job behavioral observation programs to supplement information obtained during the hiring process.

In a study related to this latter recommendation, NUREG/CR-2076 was aimed at the development of standards for a behavioral observation program which could be used by NRC licensed nuclear facilities to detect indications of emotional instability in employees who have access to protected and vital areas. Emphasis was placed on identifying observable characteristics which could be assessed by supervisors or peers in the work environment. Referred to as the "behavioral reliability program," the basic requirement is for personnel in appropriate positions to watch subordinates for signs of unreliability, poor judgment, behavior change, or inability to cope with job stress. Thus, behavioral reliability programs are aimed at <u>detecting</u> aberrant behaviors or behavioral change within the context of the job environment.

Based on their own analyses as well as those appearing in NUREG/CR-2075, the authors of NUREG/CR-2076 identify five broad criteria of behavioral unreliability:

1. argumentative hostility toward authority,
2. irresponsibility,
3. defensive incompetence,
4. reaction to stress, and
5. emotional and personal adaptability.

A variety of illustrative examples of behaviors falling under each of these major headings is provided for guidance to personnel who would be responsible for a behavioral reliability program. This is supported by an analysis of 158 "critical incidents" gathered during job analysis interviews at power generating sites. In this respect, the study has a convincing empirical foundation.

It is reported that, as a result of the American National Standards Institute proposed revision (ANSI 18.17), some nuclear power generating facilities have begun pilot behavioral reliability programs. This revision calls for a continuing observation program to be administered by supervisory personnel who are instructed in methods of recognizing unusual behavior. In discussing these programs with representatives of industry, the authors of NUREG/CR-2076 reported a preference for program management by each individual facility rather than by a government regulatory agency. In evaluating industry programs, however, the authors noted a lack of inter-rater reliability due to an inadequate definition of the behaviors to be measured and lack of specificity about how often and under what circumstances observers should report their observations.

A panel of experts convened for the purpose agrees that the supervisors of nuclear power plant employees would find some cues of behavioral unreliability quite easy to detect. Included among these were energy level, hostility, anger, insubordination, frequent errors, and other indices of the quality of work performance. It was cautioned, however, that the supervisor must be convinced that his reporting of such behavior will be helpful, not harmful, to his subordinates. Further, he should serve as an observer and referral source only, not as a diagnostician or counselor. A further restriction on the approach is that it is aimed primarily toward individuals who are experiencing emotional instability because of personal adjustment problems or the stresses of the job. It was agreed that even the most comprehensive assessment and observation program would be hard pressed to detect the determined saboteur.

Despite the evident complexities of implementation, NUREG/CR-2076 concludes that a behavioral reliability program should be an integral part of safeguarding a nuclear facility. It is further concluded that no existing behavior reliability program in either the public or private sector can be "lifted" as is and installed in nuclear facilities. Further, there is no body of research that has demonstrated the effectiveness or ineffectiveness of such a program in the nuclear power industry.

NUREG-0660 calls for the development of regulations (1) to provide that applicants for RO and SRO licenses are psychologically fit (emphasis on "stress and malevolence"), and (2) to prohibit licensing of persons with histories of drug and alcohol abuse. To this end, the Office of Standards Development sponsored the previously referred to study of "Standards of Psychological Assessment of Nuclear Security Personnel" (NUREG/CR-2075) and a study of "Behavioral Reliability Programs for the Nuclear Industry" (NUREG/CR-2076). In addition, the Office of Research is sponsoring research to obtain "a clearer understanding of the operator's performance under. . .stress." General Physics Corporation is examining heart rate and EKG data in a search for physiological indices of stress response but, as yet, there are no definitive results (62).

PLANNED ACTIVITIES

A proposed rule is being formulated to establish a screening and behavior observation program for plant employees having unescorted access, including ROs and SROs. The program calls for (1) background investigation, (2) psychological assessment, based primarily on written psychological tests, and (3) "continual behavioral observation." This rule is being prepared by RES and NMSS. NRR (LQB) is tasked with the analysis of its impact and effectiveness.

The Edison Electric Institute is supporting Personnel Decisions Research Institute in a new study of criteria for the selection of maintenance personnel in electric power plants. This study is to be completed in 1983. The same general approach to test validation will be employed as was described earlier for selecting operating personnel. The objective is to develop predictors that will get at "particularly critical job components."

NUREG-0660 1.A.3.3 identifies plans for improved methods of screening ROs and SROs for "psychological fitness," and the NRR/LQB Safety Technology budget for FY83-84 provides for a moderate continuing effort to be completed in FY83.

A substantial, continuing effort under ORNL direction to collect and assess operator performance data ("Safety Related Operator Actions") is relevant to personnel selection issues since, potentially at least, this could lead to the development of objective on-the-job performance criteria for use in validation.

MISSING ELEMENTS

The most fundamental missing element in these and other industry studies is the absence of objective criteria whose relevance and scope, with respect to operator or maintainer performance, is assured. An attempt by PDRI to employ data recorded automatically in control room simulators as a criterion proved unsuccessful, apparently because the resulting metrics were not readily translatable into meaningful dimensions of operator performance. As noted earlier, supervisory ratings collected on carefully designed scales may reflect a portion of the variance in the ultimate criterion of operator (or maintainer) performance, but research in other contexts suggests that the overlap of these ratings with actual performance capability may be negligible.

There have been few industry attempts to validate selection procedures against training criteria, including assessments of performance in a simulator (47). Although these are not the ultimate criteria of interest, they are certainly of considerable practical and economic interest as intermediate criteria. Curiously, we know of no attempt to validate selection criteria against The probability of successfully becoming licensed.

As noted above, suitable criteria of psychological fitness and emotional stability on the job are also missing. The definitions in NUREG/CR-2076, being based on actual incidents involving power plant personnel, may represent a starting point for development in this area.

TECHNICAL FEASIBILITY AND PROBLEMS

While there are no unsolved theoretic problems associated with the development and validation of selection procedures for nuclear power plant personnel, the principle limitation, as noted, concerns the availability of criteria that reflect the various dimensions of operator and maintainer on-the-job performance. Work supported to date by EPRI on methods for automatically recording operator control actions in simulators suggests that this is technically feasible, but that there are problems to be solved in translating such data into all of the significant dimensions of performance effectiveness. In particular, the interpretive and evaluative responses of the operator are not captured (62).

There also may be technical limitations on the effectiveness of conventional standardized aptitude, interest, and personality tests for selecting nuclear power plant personnel. It is possible that conventional aptitude tests will prove more predictive of achievement in the classroom and performance on licensing examinations than they will of actual performance in the control room. Significantly improved prediction of control room performance may require innovative test construction including, for example, the application of computer- based technology aimed at the dynamics of cognitive abilities hypothesized to be relevant to plant state diagnostics and decision making. In this regard, there is a need for refinement of the process of translation between models of operator performance and corresponding aptitude and personality test variables.

With respect to the vigilance aspects of operator performance, and the related issues of work/rest cycles (described in more detail later), an unsolved technical problem is the requirement for a method of tracking changes in operator monitoring performance over extended periods of time. This is fundamental to the development of methods for selecting operators who are resistant to the vigilance decrement. Some progress has been made in this area as reflected, for example, by the rather consistent finding that introverted people make better watchstanders than extroverted people. The level of predictability from available personality tests is modest, however, and most validations have employed laboratory tasks whose relevance to real-world operations may be questioned. Further advances in this area will require the development of a criterion of the vigilance decrement if it does in fact occur in nuclear power plants.

Finally, the assessment of emotional stability and performance under stress represents a considerable technical challenge. First, except in cases of very bizarre behavior, there are questions about the validity of emotional stability assessments made by nonprofessional personnel. Related to this is the lack of a suitable criterion of performance under stress. It is possible that the control room simulator can be used as a satisfactory test bed for developing genuine stress responses in operator personnel. If this should prove to be the case, the opportunity is opened up for the assessment of individual differences in resistance to stress (or ability to cope with it) and for relating these differences to the results of assessment procedures. Second, as noted in NUREG/CR-2075, there have been no validation studies of psychological testing and interviewing techniques in relation to "behavioral reliability" in nuclear power plants.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The personnel selection subsystem interacts strongly with each of the other major elements of the human factors system. The cognitive capabilities of operator and maintenance personnel obviously impact their trainability and, therefore, the design of the instructional system, the need for various kinds of performance aids, their "mental model" of various plant systems, and their ability to interpret information as it is presented through the various system displays. Clearly, a wide variety of cognitive abilities is associated with requirements for assessment of plant state, decision making, and prediction of action outcomes. There are critical personality characteristics associated with the ability to maintain vigilance under tedious operational conditions, and maintaining composure under stress conditions. These requirements also interact with information display requirements, decision-aid requirements, design of procedures, etc.

RECOMMENDATIONS

The following recommendations are made in the order of their potential priority. No distinction is made concerning whether the recommended action should be pursued by the NRC or by industry, although industry appears to have taken the lead with respect to the personnel selection problem. Further, NUREG/CR-1750 regards the assessment of applicant aptitudes and interests to be an inappropriate function of the regulatory process. In contrast, screening procedures which identify signs of unsuitable personality or inability to meet the physical demands of the job are regarded as having potential impacts on plant and public safety, and therefore should be regulated.

1. Research should be conducted leading to the validation of current and newly proposed selection procedures against comprehensive criteria of the job effectiveness of operator and maintenance personnel. The validation should reflect both technical performance and secondary criteria such as trainability and probability of meeting NRC qualifications for licensing. (See Section 1.8 a for discussion of criterion issues).

Importance: High

Schedule: Urgency - immediate start using currently available criterion measures.

Duration - completion indefinite - continuing effort is called for as criterion measures are refined and new selection technology becomes available.

Resources: 1 person-year per year.

Implementation: Professional psychologist with experience in industrial or military personnel selection. Access to data reflecting various criterion measures.

Dependencies: None, but coordinate with results of INPO and NRC task analyses.

2. Research should be conducted on individual differences in ability to cope with the stress generated by accident conditions and methods of screening individuals to ensure high probability of emotional stability on the job. The Shift Supervisor who, according to Worsham (164), is the most stressed individual in the plant should receive particular attention. The possibility of using the simulator as a platform for inducing job-relevant stress should be investigated, and various physiological indicators of stress response should be critically evaluated.

Importance: High

Schedule: Urgency - start in 1-2 year timeframe

Resources: 3-4 person-years per year for first two years; reevaluate after that in view of progress and promise.

Implementation: Research psychologist; stress psychologist; dedicated simulator time.

Dependencies: None

3. Monitor and critically evaluate behavioral reliability programs initiated by industry, including benefits, evidence of validity/payoff, and potential deficiencies/abuses.

Importance: Medium

Schedule: Urgency - Upon initation of behavioral reliability programs. A continuing effort until benefits of the program are clearly established.

Resources: NRC staff plus consultants - ½ person-year per year.

Implementation: Qualified industrial psychologist.

Dependencies: Implementation of behavioral reliability programs.

4. Research should be conducted on new technology testing procedures in an attempt to predict variance in personnel effectiveness criteria that is not well predicted by presently available aptitude and temperament tests.

Importance: Low

Schedule: Urgency - Start in 6-10 years.

Duration - A continuing effort as testing technology advances.

Resources: 2 person-years per year.

Implementation: Qualified cognitive/measurement psychologists; technology specialist in computerized testing.

Dependencies: Completion of item #1 above.

4.2 Operator Certification and Licensing

GENERAL SYSTEM REQUIREMENTS

The objectives of certification and licensing are to ensure that control room and other plant personnel, as appropriate, have the necessary technical knowledge and skill to ensure safe, competent plant operation and maintenance. (Maintenance personnel are not presently licensed or certified, although it has been argued that they should be.) The focus of attention since Three Mile Island has been on enhancing the qualifications of ROs, SROs, and Shift Supervisors. NUREG-0737, which is viewed as the most definitive NRC statement on operator qualifications (121), calls for the immediate upgrading of RO and SRO qualifications. It details experience and training requirements, and specifies certain control manipulations required of operators to assure their capability for controlling plant parameters. In general, the NRC emphasis has been on upgrading of formal education, more stringent examination cutoff scores, and increased involvement of management in certifying the technical competence of their personnel.

CONSTRAINTS

There are both technical and practical constraints associated with ensuring that candidates for licensing possess all necessary qualifications:

1. The practical impossibility of demonstrating, through testing, that candidates possess all required skills and knowledge and know how to apply them in a great variety of circumstances, not all of which can be specified in advance.

2. The extensive investment of examiner and simulator time necessary for comprehensive testing of operator capabilities.

3. The absence of objective performance criteria for validating licensing examination cutoff scores.

4. The inability to observe directly critical operator behaviors associated with assessment and prediction of plant state, problem diagnosis, and decision making.

5. The lack, in some locations, of easy accessibility to appropriate simulators and (if deemed necessary) coursework in appropriate higher level curricula.

PRESENT STATUS IN MEETING REQUIREMENTS

Reactor Operators. As part of the license examination application for an RO, the utility must certify to the NRC that the candidate has learned to operate controls "in a competent and safe manner" (NUREG/CR-1750). A proposed revision to ANSI/ANS-3.1-1978 would require that this process include a performance examination on the simulator, a prelicense examination, and certification of competence by high levels of corporate management. The NRC has been particularly emphatic about the last requirement. Perhaps the most significant requirement is the inclusion of the simulator examination which would require the candidate to operate at power, deal with malfunctions, and perform startups and shutdowns of the reactor. NUREG 0737 specifies that simulator examinations must be initiated by October 1, 1981.

There is considerable emphasis on increasing the amount of power plant work experience required as a prerequisite to RO licensing. Current requirements (SECY 79-330E, July 1979) stipulate that ROs should have two years of previous power plant experience or the "equivalent." It is proposed (SECY 81-84, February 1981) that the work experience requirement be increased to three years in the testing, operation, and maintenance of a power generating plant (not necessarily nuclear), one year of which must be at the site for which a license is sought. Further, the candidate must have had a minimum of six months experience in RO duties, or in service as an auxiliary operator.

There appears to be general industry agreement with this proposed requirement. ANS 3.1 (revised April 1981) specifies three years of power plant experience, including one at a nuclear facility, and six months as a nonlicensed operator, with a preference for one year. INPO has proposed (GPG-03, April 1981) that the RO candidate have three years of power plant experience, including one year at a nuclear facility, and six months at the same facility for which the license is sought.

It should be clear that the experience requirements for ROs or other positions can have little meaning if they are stipulated only in terms of months or years. Obviously, it is the scope and depth of experience in various activities and tasks involved in plant operation that should be the focus of concern. It is conceivable that a year of experience in a carefully designed apprentice program might be considerably superior to three years of nonuniform, unstructured, happen-stance "experience" on the job. Some NRC commissioners have suggested amendments to SECY 81-84 which would permit higher level engineering education (up to and including a BS degree) to substitute for up to two years of the on-the-job experience requirement. No objective basis exists for "trading off" operating plant experience against formal education and it is evident that there is a considerable diversity of opinion concerning the optimum mix. To determine what that mix should be, it will be necessary to evaluate experience on some basis other than simple longevity and to evaluate the value of higher level education to actual RO performance.

The proposed revision to 10 CFR, Part 55, May 1980, specifies that future RO candidates be required to serve for at least one year in some capacity at the plant for which a license is being sought and in the capacity of an AO for at least a half-year at the plant or similar plant. NUREG/CR-1750 proposes that the latter requirement be increased to one year. There seems little doubt that prior operational experience is relevant to the qualification of RO candidates for licensing, but there are varying opinions concerning the relative importance of prior experience and seniority in making advancement decisions. At some facilities seniority alone is used as the criterion for selection to candidacy, evidently due to labor-management agreements. NUREG/CR-1750 recommends that a variety of criteria be used, including demonstrated technical knowledge, rate of qualification progress, AO performance evaluations, training performance, and seniority.

In addition to whatever initial selection criteria are used, considerable screening of RO candidates is possible during the RO training program. A December 1979 proposed revision to ANSI/ANS-3.1 would require the administration of examinations, either written or oral, or operational tests where simulators are available, during the course of each phase of training. According to NUREG/CR-1750, there are currently no requirements for these procedures, and utility practices vary widely with respect to disenrollment criteria, timing of disenrollments in the training sequence, and provisions for remedial training in the event of disenrollment. This is in sharp contrast, for example, to Navy practices where graded watches, and oral examinations by a progress review board and by instructors, are routinely used.

The importance of appropriate examining procedures during various phases of instruction, as emphasized by NUREG/CR-1750, stems from the presumption that failure to acquire a particular needed skill or knowledge during one phase of training may mean that the opportunity to satisfactorily acquire that knowledge or skill is lost. In general, there is considerable building of skills and knowledges in each phase of training upon those acquired in previous phases. The position is taken in NUREG/CR-1750 that the NRC should require, as part of a licensee training program, that the utilities establish a formal method for certifying satisfactory knowledge and performance for each applicable phase of the training program. The importance of this recommendation is a function, of course, of how well the licensing examination samples all necessary operator skills and knowledges. In effect, the RO training program as conducted by the plant or a vendor is a screening procedure for certification of the RO candidate. In many instances this endeavor reportedly has been oriented simply toward helping the candidate pass the NRC examination; this, of course, places a heavy burden for operational qualification on the NRC examining process.

Federal Regulation 10 CFR, Part 55 requires that the facility provide evidence that the applicant has learned to operate the controls of the reactor in a "competent and safe" manner. As proof of this, however, the NRC previously has accepted a certification which includes details on courses of instruction, numbers of course hours, numbers of hours of training and nature of training received, and evidence that startup and shutdown experience was received. NUREG-0094, "NRC Operator Licensing Guide," reiterates these requirements, and, in addition, requires that the applicant must have manipulated the controls of the reactor through at least two reactor startups and have participated as a member of the control room in several plant transients. Although it appears that these requirements are moving in the desirable direction of an objective demonstration of operating skills, the development of objective performance standards and methods of performance measurement are problems that remain to be solved. There seems to be an assumption that the requirement for corporate management to certify a candidate's readiness for licensing will somehow ensure that the necessary skills have been acquired. It may well be that the task of certifying the candidate's technical competence should rest with the utility; it is not clear that the utilities have the type of guidance they need concerning performance assessment techniques to ensure that appropriate standards of performance are being uniformly applied through the industry. Indeed, NUREG/CR-1750 concludes that the techniques used by the utilities to certify technical competence focus primarily on the candidate's ability to pass the NRC examination. The newly implemented requirement that all RO and SRO candidates take an examination in the plant simulator may or may not reduce this problem.

Taken at face value, the NRC's licensing requirements are aimed at ensuring that the candidate has sufficient knowledge of system theory and sufficient operational proficiency to perform competently in the control room under normal, off-normal, and emergency conditions. However, whether the requirements achieve these objectives is clearly dependent on the relevance of each requirement to operating safety and effectiveness, and whether the full scope of critical skills and knowledge is sampled.

The least controversial of the NRC's requirements appears to be the amount and type of prior power plant experience, and the most controversial appears to be the amount of formal education required. There are, however, other fundamental questions regarding the the content validity of the licensing examinations, the reliability of the examination procedure, and the operational meaning of the various cutoff scores employed. Although the considerations differ for ROs, SROs, and Shift Supervisors, there are certain fundamental questions that apply to all three groups and that extend, in fact, to nonlicensed plant personnel as well. These include:

1. How should prior operational experience (including experience in Navy nuclear power plants) be evaluated in the selection/certification process? What weight should on-the-job experience in various capacities receive? (Labor unions sometimes argue that longevity alone is sufficient.)

2. What should be the criterion for written test and performance (walk-through or simulator) examination cutoff scores, assuming that all relevant areas of critical job knowledge are properly sampled?

3. Is advanced education (or a college degree) relevant to job performance and, if so, what aspects of that education are relevant to what aspects of job performance? What formal courses of instruction are necessary or desirable for effective overall performance on the job and safety of operations?

Because no objective performance criteria exist against which to perform validation studies of licensing requirements, the NRC has used its best subjective judgment in formulating a reasonable set of qualification criteria. The fundamental problem, of course, is that the operational validity of each of the qualification requirements is uncertain and objective criteria by which to determine their validity are not presently available.

Industry objects to NRC qualification requirements in various respects but has not offered clearly defensible alternatives. For example, members of industry have criticized the NRC for failure to identify fully the technical expertise requirements of the control room operator's job and view the NRC's imposition of the Shift Technical Advisor as an excuse for this failure. Similarly, the SPDS is viewed by some as a naive attempt to display a few parameters in such a way that any "dumbbell" can immediately perceive what is wrong.

The examinations presently employed for the qualification and requalification of ROs and SROs reflect broad assumptions concerning the kind and degree of system theoretic knowledge necessary for competent and safe operations. In other job contexts, scores on written examinations have shown surprisingly low correlations (often zero) with objective measures of operational performance. This does not negate the possibility, of course, that the examinations measure knowledge that is necessary, but not sufficient, for effective performance. Regardless, perhaps the most important generalization that can be made about current examining procedures is that their operational relevance is yet to be shown. There are reasons to question (1) whether the examinations have sufficient validity and scope; (2) the degree of subjectivity involved in the examining process and the methods of scoring; (3) the degree to which the verbal skills of the candidate (both written and oral) influence the obtained scores, and whether this is job relevant; and (4) the operational meaningfulness of the cutoff scores employed.

Industry has not appeared particularly sensitive to the examination of validation issues and has, in many cases, appeared to adopt an "if you can't lick 'em, join 'em" attitude. That is, it has been common practice for industry to pattern its own qualification examinations after the format and content of the NRC examinations in the apparent hope that training to pass the examination will maximize the probability of the candidate's being licensed. There is also an assumption on the part of some in industry that passing the NRC examination means the operator must be fully competent. This viewpoint is defensible, of course, to the extent that the examinations do represent a valid, comprehensive measure of all necessary operational and safety knowledge. As noted, this is more assumed than demonstrated. Industry (INPO) has also recognized that the training base should be much broader than the NRC examinations, which are primarily safety oriented. In addition, there is considerable unease with the content of the examinations on other counts; many in industry feel that the skills required to pass the examinations are different from the skills required to operate. In this connection, some members of industry feel that the walk-through portions of the examination and use of the simulator as a test platform represent steps in the right direction, given that the available simulator appropriately reflects their control room specifics.

Some industry representatives assert that the examination questions often appear to have been written by people "who never ran a power plant." While agreeing that some theoretical knowledge is necessary, there was concern over why they had never seen practical questions, for example, "on proper feed pump operation." Industry cites numerous examples of operators with several years of experience, previously licensed, who have failed the new examinations. It is asserted that there is a negative psychological impact of these failures which will be reflected in the loss of already scarce competent personnel. In one case it was suggested that the NRC has adopted the Navy's model of examination and re-examination. This is regarded as "negative management," which reportedly was resented by most Navy personnel who could, of course, do nothing about it when they were in the Navy. In civilian life it is viewed as a different matter.

It is also argued by some that industry is in a better position to perform thorough examinations of their operator personnel than is the NRC. This extends to the use of simulators where they feel that any competent operator could "snow" the examiner since the candidate knows so much more about the plant than the examiner does. These contentions are, of course, just that, but they are unlikely to be put to rest without appropriate validation studies of the entire examination process.

One related issue is of considerable interest with respect to test theory. This has to do with the operator's "mental" model of how the plant functions and the recent emphasis on the importance of the operator's ability to respond to "symptoms" as opposed to "events." Since the full range of "events" cannot possibly be foreseen, the operator must be capable of relating "symptoms" to system functioning; but, it has been claimed that the operators rarely think in terms of system functioning. If this viewpoint is valid, it obviously should impact the content of examinations as well as the qualification process.

Guidance concerning the content of SRO (and RO) licensing examinations is given by NUREG-0094 and 10 CFR Part 55. According to NUREG/CR-1750, this guidance is organized around facility features, system characteristics, and theory, rather than around what the operator is required to do in performing his job. The operational significance of the arbitrary passing scores (80% overall and 70% on each part) remains to be determined. NUREG/CR-1750 takes a stronger position and suggests that the written examinations lack sufficient content validity to ensure that the applicant has the required knowledge to function as an RO or SRO. According to that source, RO and SRO skills and knowledges cannot be evaluated using current OLB licensing practices with respect to a candidate's ability to:

1. Coordinate actions of two or more procedures.
2. Carry out actions of abnormal, off-normal, and alarm procedures in proper sequence through reference to procedures.
3. Recall plant personnel.
4. Use decision rules
5. Maintain good judgment and problem solving performance under stressful or physically hazardous conditions.
6. Identify cues as indicative of an emergency condition.
7. Determine that cues are not completely addressed by a single procedure.
8. Determine whether multiple casualties have occurred.
9. Identify cues as indicative of an abnormal, off-normal, or alarm condition.
10. Receive advice from the Shift Technical Advisor or other technical personnel.
11. Coordinate actions of all shift personnel.

In addition, NUREG-0094 (NRC "Operator Licensing Guide") has been criticized because it makes no provision for a quantitative assessment of the applicant's performance (i.e., only a pass/fail criterion is employed); evaluations currently are judged only as "satisfactory," "marginal," or "unsatisfactory" in each subject area; and there is no objective or consistent procedure to ensure that each examiner has similar criteria for making these asessments or even that a given examiner uses the same acceptance criteria from one examination to the next.

A study by Analysis and Technology, Inc. (NUREG/CR-1750) presents what limited evidence there appears to be on the correlation between licensing examination scores and criteria of on-the-job performance. In that study, the performance criterion was supervisory ratings of ROs and SROs into three categories: "below average," "average," or "above average." Average licensing examination scores were found to be insignificantly different for operator personnel assigned to these three categories. Further, scores on individual examination sections failed to discriminate reliably among these groups. Unfortunately, no data were presented concerning the reliability of the performance criterion or its relevance to actual task performance. It is quite possible that unreliable and fallible predictors were correlated with unreliable and fallible criteria. On the other hand, research in other contexts has shown that correlations are modest, at best, between measures of academic achievement (reflected by examination scores) and measures of on-the-job performance including supervisory ratings.

The SRO. Many of the previous observations concerning the qualification of ROs, particularly the examination practices involved, also apply to the SRO. There are, however, additional considerations related to the qualification of SROs. A two-track path to the position of SRO is identified in which this position can be reached either by coming up through the operator ranks or by selection from a pool of degree-holding engineers. Substantial additional formal education requirements are called out for nondegreed SRO candidates. From the perspective of the NRC, qualification is again defined primarily by formal education and job experience, as opposed to screening on the basis of aptitudes or demonstrated technical competence.

With respect to education, it has simply been required, up to the present time, that SROs be high school graduates or "the equivalent." However, SECY 81-84 (February 1981) would require the SRO to have a BS degree, including 60 hours in technical subjects. Several alternative revisions to this requirement have been proposed by various commissioners and NRC the staff. The least stringent proposal by the NRC staff would upgrade the SRO's formal education to include 30 semester hours of technical education at the college level. This is in agreement with the recommendation in NUREG/CR-1750, as well as with the April 1981 revision of ANS 3.1. In contrast, the INPO proposal requirement (GPG-03, April 1981) leaves the formal education requirement unchanged; i.e., graduation from high school or "equivalent."

The relevance of college level coursework to effective performance as an SRO is assumed, not demonstrated. There is concern within industry that substantial increases in educational requirements may have an adverse impact on perceived career opportunities which will result in the loss of potentially effective SROs and Shift Supervisors. Further, there is no evidence that degreed engineers necessarily made good ROs or SROs.

There are also wide differences of opinion concerning the amount of prior work experience necessary for qualifying SROs. The current requirement (SECY 79-330E, July 1979) specifies four years of responsible power plant experience, including one year at a nuclear power plant. The proposed requirements (SECY 81-84, February 1981) would reduce the total number of years of power plant experience by requiring just two years of experience at a nuclear facility, including one year of experience as a licensed RO and six months of experience in the facility for which the license is sought. There seems to be general agreement within the NRC concerning the desirability of this change. However, NUREG/CR-1280 recommends four years of power plant experience, including three years as RO at the plant, or similar plant, for which a license is desired and NUREG/CR-1750 would also require four years of power plant experience, including two years at a nuclear power plant and six months at the same facility for which the license is sought.

The April 1981 revision of ANS 3.1 would require three years of power plant experience, including two years at the nuclear facility. In addition, the candidate must have participated in RO duties, including six weeks of operation while above 20% power. INPO's recommendation (GPG 03, April 1981) is for four years of power plant experience, including two years of nuclear experience, and one year of control room operation, of which six months must be at the same facility for which the SRO license is sought. Thus, in general, industry appears to feel a need for more power plant experience for SROs than that called for by the NRC.

It is of interest that none of these proposals would require more than one year of experience as a licensed nuclear power plant operator (RO). The rather extensive experience requirements reflected in the industry proposals again raises questions concerning the depth and diversity of experience and its relevance to a well rounded set of qualifications for senior operator personnel. It seems likely that, without some regulatory guidance, "experience" could be acquired in a variety of ways and that much of it might not be necessary to effective development of SRO qualifications.

As in the case of the ROs, current and proposed qualification requirements for SROs also involve trade-offs between prior operational experience and amount of formal education. SECY 79-330E (July 1979) provides for the substitution of additional academic or related technical training up to a maximum of two years of power plant experience on a one-for-one basis. A proposed revision to SECY 81-84 by the NRC staff would permit the substitution of a BS degree for experience as an RO. That is, a BS degree plus two years of more general nuclear power plant experience would suffice. NUREG/CR-1280 suggests that the BS degree in engineering or physical science might substitute for a maximum of three years of the four-year power plant experience requirement suggested in that document. The proposed revision to ANS 3.1 (April 1981) also suggests substitution of a BS degree for power plant experience but on a case-by-case basis.

In general, these proposals reflect an assumption of considerable relevance of the engineering degree to the performance of SROs. As in the case of the RO, the merits of this argument remain to be demonstrated as does the optimum balance between prior operational experience and a relevant higher education curriculum. The December 1979 proposed revision to ANS 3.1 included a suggested curriculum of college level instruction in mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, electrical theory, and reactor control theory. As we will see later, there is far from uniform opinion concerning the nature of a desirable curriculum either for SROs or for Shift Supervisors.

REQUALIFICATION OF OPERATOR PERSONNEL

The May 1980 proposed revision to 10 CFR Part 55 places increased emphasis on the examination procedures by requiring that a license may be revoked or suspended for failure to satisfactorily complete annual requalification examinations. It introduced a requirement for the use of a simulator for abnormal, infrequent, and emergency training of ROs and SROs as a part of the requalification program; and a change of purpose of the annual examination from the determination of areas in which retraining is needed to verification that an operator can operate the controls or supervise operations of the controls in a safe and competent manner. An oral examination and simulator test are called for, as well as a written examination. Enclosure 1 to SECY-79-330E (July 1979) recommends periodic (annual) retraining and recertification on a "full-scope" simulator. NUREG-0737 specifies that certain control room manipulations will be performed on an annual basis.

From a performance-oriented viewpoint, some of these changes in emphasis are clearly desirable. However, in the interest of meaningful qualification and requalification, sooner or later it will be necessary to establish a proper balance between tested performance on a broad sampling of operational and emergency tasks, and tested knowledge of system functioning that is necessary for assessing plant state, diagnosing off-normal conditions, and making appropriate emergency decisions. Concern with present examination practices for licensing ROs and SROs extends to the requalification program since the same examining procedures and criteria are used.

Appendix A to 10 CFR Part 55 establishes federal requirements for the requalification of licensed operators. The purpose of the requalification program is to demonstrate competence for license renewal. There are three basic requirements (NUREG-0084):

1. A satisfactory medical examination.

2. A finding of extensive and active engagement as an operator or senior operator under the

existing license, competent prior operation and the capability of continuing to do so, and successful participation in the facility requalification program, or successful completion of a prescribed reexamination. To be considered actively engaged under his license, the individual's duties must be performed at the facility, normally, on a day-to-day basis.

3. Continued need for a license.

ANS 3.1, Standard for Qualification and Training of Personnel for Nuclear Power Plants (draft revision, December 1979), describes the requirements of annual retraining programs. These are to include pre-planned lectures, on-the-job training, and operational evaluation on a regular and continuing basis. Of particular interest are the required control manipulations (listed below). Those items that are starred are to be performed on an annual basis and all others are to be performed on a two-year cycle. Control manipulations that are not performed at the plant are to be performed on a simulator.

1.* Plant or reactor startups to include a range such that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.
2. Plant shutdown.
3.* Manual control of steam generators and/or feed water during startup and shutdown.
4. Boration and/or dilution during power operation.
5.* Any significant (greater than 10%) power changes in manual rod control or recirculation flow.
6. Any reactor power change of 10% or greater where load change is performed with load limit control or where flux, temperature, or speed control is on manual (HTGR).
7* Loss of coolant.
   a. including significant PWR steam generator leaks
   b. inside and outside primary containment
   c. large and small, including leak rate determination
   d. saturated reactor coolant response (PWR)
8. Loss of instrument error.
9. Loss of electrical power (and/or degraded power sources).
10.* Loss of core coolant flow/natural circulation.
11. Loss of condenser vacuum.
12. Loss of service water if required for safety.
13. Loss of shutdown cooling.

14. Loss of component cooling system or cooling to an individual component.
15. Loss of normal feedwater or normal feedwater system failure.
16.* Loss of all feedwater (normal and emergency).
17. Loss of protective system channel.
18. Mispositioned control rod or rods (or rod drops).
19. Inability to drive control rods.
20. Conditions requiring use of emergency boration or standby liquid control system.
21. Fuel cladding failure or high activity in reactor coolant or off-gas.
22. Turbine or generator trip.
23. Malfunction of automatic control systems which affect reactivity.
24. Malfunction of reactor coolant pressure/volume control system.
25. Reactor trip.
26. Main steamline break (inside or outside containment).
27. Nuclear instrumentation failures.

The March 1980 NRR letter to power reactor applicants and licensees requires that the six starred items be performed annually by in-plant walk-through drills, or drills in a control room simulator. The same letter modifies the criterion for participation in accelerated requalification so that it will be consistent with the new NRC written examination passing grades of 80% overall and 70% on each category.

The May 1980 proposed revision to 10 CFR Part 55 would:

1. Require enrollment in the requalification program as a condition of an operator's license.
2. Reinforce the importance of completing annual examinations by indicating that a license may be revoked or suspended for failure to satisfactorily complete these examinations.
3. Require the use of a simulator for abnormal, infrequent, and emergency training for ROs and SROs as a part of the requalification program.
4. Change the purpose of the annual examination from the determination of areas in which retraining is needed to verify that an operator can operate the controls, or supervise operations of the controls (SRO) in a safe and competent manner.
5. Include an oral examination and simulator test as well as a written test in the annual examination.

6. Require that the NRC administer annual examinations (however, as stated in the proposed revision, the NRC may permit these examinations to be given by the facility) (NUREG/CR-1750).

Based on a review of the programs at nine reactor sites, the authors of NUREG/CR-1750 concluded that industry requalification programs are fairly "consistent" compared to the wide spectrum of practices found in initial license training programs. In general, they were in accord with ANS 3.1 (1978) and 10 CFR Part 55. With respect to requalification programs conducted in the simulator, most of these were found to concentrate on the more advanced aspects of plant operation such as:

1. Operator knowledge of major equipment and instrumentation failures commonly known throughout the industry and their effects on plant operation.
2. Major accident diagnosis and corrective action.
3. Recognition of multiple failures and their effects.
4. Training in infrequently used procedures.

Requalification programs are jointly audited by IE and OLB. OLB audits the annual examinations to ensure that they are comparable to the NRC examinations in depth and content, and that the grading is also comparable. SECY-79-330E states that continuous evaluation of on-the-job performance is also required. It is, therefore, expected that training and evaluation will be continuous, rather than a once-a-year phenomenon, although it is not clear on what basis the evaluations will be made. However, it is recognized in SECY-79-330E that there should be more explicit requirements regarding exercises to be included in simulator training programs. The need for a broad spectrum of normal and abnormal operations in response to transients and emergencies, including multiple failures, compound abnormalities, and imperfect initialization, is recognized.

NUREG/CR-1750 reports that the requirements for periodic observation and evaluation of operators by industry are met by:

1. Evaluation by shift supervisory personnel during actual plant operations and abnormal and emergency conditions.
2. Evaluaton during simulator training.
3. Evaluation during walk-through drills conducted in the plant.

If this is done on a comprehensive and systematic basis, there could be clear benefits. It is reported, however, that operator performance is simply graded as "satisfactory" or "unsatisfactory" by a designated supervisor. Thus, the criteria and standards for these important activities represent a significant uncertainty. NUREG/CR-1750 concludes that both the utilities and the NRC inappropriately rely on the results of the annual written examinations as the basis for judging operator competency. According to that study, most of the utilities surveyed did not have an effective system for comprehensive evaluation of technical peformance. Use of an annual written examination, of scope and depth comparable to the NRC licensing examination, fosters the development of requalification programs designed around passing those examinations, and can have a negative effect on operator motivation. The annual examination is, by itself, an ineffective tool for evaluating many aspects of operator competence (NUREG/CR-1750, p. 2-217). This report makes major recommendations for changes in the proposed requalification program, including (1) an annual operating test using a simulator for both individual evaluation and team evaluation, supplemented by oral examinations if weaknesses are noted and (2) the administration once every <u>five</u> years of a comprehensive written and oral examination similar to that used for initial licensing.

The NRC has been tasked to administer all of the requalification examinations as a check on requalification program effectiveness (November 27, 1979 memorandum to SECY-79-330E/330F). In view of the considerable resources required for this task, it seems likely that test procedures will continue to emphasize formal knowledge as opposed to demonstrated operational proficiency.

<u>Shift Supervisor</u>. NUREG-0585 calls for the upgrading of the qualifications of Senior Reactor Operators and Shift Supervisors over a five-year period starting in 1979. This publication states that Shift Supervisors should have at least a Bachelor of Science degree or "equivalent" training and experience in engineering or the related physical sciences. He should also hold an SRO license and have served as a Reactor Operator for six months. In establishing "equivalence" with a Bachelor of Science degree, consideration is to be given not only to formal courses in engineering and related sciences, but also to education in the liberal arts. However, it is recommended that the use of "equivalence" to the bachelor's degree be exercised to only a limited extent and that most Shift Supervisors should hold degrees.

Since the position of Shift Supervisor is not currently licensed, there is no minimum standard regarding the amount of formal education. There is general agreement within the NRC, however, concerning the need for education beyond the high school diploma. The recommendations range from a minimum of 60 semester hours of college level technical subjects to a complete BS degree. The proposed revision of ANS 3.1 (April 1981) concurs with the requirement of 60 semester hours of college education in technical subjects beyond the high school diploma, whereas INPO (GPG-03, April 1981) is satisfied with the high school diploma or "equivalent."

Closely related to the educational requirements for Shift Supervisors are several proposals for the amount of work experience these personnel must have. Several proposals from the NRC staff and commissioners agree on the desirability of five years of responsible nuclear power plant experience, including two years as a licensed SRO and one year as an SRO at the facility for which the supervisory position is sought. The viewpoints expressed in ANS 3.1 (revised April 1981) are only slightly more conservative. These recommendations agree on the need for four years of power plant experience, including two years at a nuclear facility. ANS 3.1 requires participation in RO duties and calls for six weeks of experience in certain specific operational evolutions (above 20% power, start-up, shutdown, etc.). The INPO proposal includes one year of experience in SRO duties, six months of experience at the facility for which the supervisory position is sought, and participation as an SRO for four months in various specified plant evolutions.

INPO (September 1981) has sponsored a survey of 40 Shift Supervisors to identify the tasks that they perform, the knowledges required, and the knowledges "offered" by a number of associate of science and bachelor of science degree programs throughout the country. Then, the knowledges "offered" were compared with those required for the job. It was concluded that no universally applicable academic curricula meets the knowledge requirements of Shift Supervisors. An examination of curricula in the physical sciences suggested that the shift supervisor needs to be more familiar with the application of concepts than with the theory of those concepts. In most cases, the knowledge required of the Shift Supervisor did not exceed selected topics in lower-division college courses.

As in the case of the recommended upgrading of qualifications for ROs and SROs, it is evident that there is no objective basis for formulating the balance of formal education and operational experience requirements in establishing the qualifications for Shift Supervisors. It was noted earlier that the position of Shift Technical Advisor has been established for the purpose of plant accident assessment during transients and other circumstances as needed. This is an interim position which may be discontinued after Shift Supervisor and SRO qualifications have been upgraded. It is of interest, however, that the qualifications of the STA, as outlined in ANS 3.1 (draft revision, December 1979), are no greater than the educational requirements recommended for the personnel he is supposed to assist. The recommended formal education requirement is a bachelor's degree in engineering or related sciences, or a high school diploma and 60 semester hours of college level education in mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, and electrical and reactor control theory. Since the STA is required to have only one year of "professional level" nuclear power plant experience, of which six months shall be on-site, it is perhaps not surprising that the more highly experienced Shift Supervisors and SROs are often reluctant to consult the STA. Anecdotal reports suggest that the success of the STA approach has been limited because of a lack of confidence on the part of the SROs and Shift Supervisors that the STA has sufficiently detailed knowledge about plant operations to provide them the technical support they might require.

Proposed Revision 2 to Regulatory Guide 1.8 on Personnel Qualification and Training (September 1980) states that the Advisory Committee on Reactor Safeguards (ACRS) holds the view that "although a broader technical background should be required of Shift Supervisors, it may be neither necessary nor practical to require that all Shift Supervisors have a Bachelor of Science degree." The committee recommends that the NRC define its criteria for "equivalent training and experience in engineering or the related physical sciences." The ACRS believes that an educational program tailored to the requirements of reactor operation, possibly of less than four years duration, may provide a practical alternative to a formal degree program.

NUREG/CR-1280 discusses the differences in NRC, commercial, and Naval practice for Shift Supervisors. This report recommends that a new position entitled "Shift Engineer" be created. This individual would be a degreed engineer who would normally function within the technical organization, but is assigned to the Operations Manager to provide shift engineering coverage. In this concept, the "Shift Engineer" would have the power and responsibility to direct the Shift Supervisor in the event of an emergency. NUREG/CR-1656 on Utility Management and Technical Resources also endorses the notion that the Shift Supervisor should have a bachelor's degree in engineering.

Other groups have reservations about this recommendation. The Atomic Industrial Forum (AIF) has suggested that the requirement for a BS degree for Shift Supervisors could have adverse effects on plant safety in that it would possibly result in a higher turnover rate, thus reducing experience in this position at most plants industry-wide. Turnover is to be expected, it is reasoned, because the supervisor is likely to consider himself over-qualified for usual daily operations and "would not be gaining professional satisfaction" from the job. The AIF did endorse requiring additional professional education, but not requiring a degree.

The December 1979 Draft Standard ANS-3.1, "Standards for Qualification and Training of Personnel for Nuclear Powerplants," recommends that the education requirements for Shift Supervisor should include a high school diploma plus the equivalent of 60 semester hours of college level education in specified technical topics.

The proposed revision to Regulatory Guide 1.8 (September 1980) takes the following positions:

> Limiting education requirements to a high school diploma is unacceptable. The technical complexity of supervising the operation of a nuclear powerplant requires an education exceeding that demonstrated by a high school diploma.
>
> The high school diploma plus completion of a specified number of college level technical courses is also unacceptable. This does not provide a broadbased education in nontechnical subjects such as management, leadership, and written communication that is necessary to deal with many of the nontechnical responsibilities of the Shift Supervisor.
>
> A Bachelor of Science degree in Engineering or related physical science (by itself) is also considered unacceptable. This is because there are some BS degree programs that do not meet the educational requirements for a Shift Supervisor. Having a degree in engineering does not ensure knowledge of such specific areas as fluid mechanics and reactor control theory that is necessary for the Shift Supervisor.
>
> The preferred alternative is a Bachelor of Science degree in Engineering or related physical science that includes a specified number of courses in technical subjects, as well as courses in humanities and social studies such as written and oral communication, applied psychology, political science, and economics.
>
> It was concluded that:
>
> "A degree is based on a well-thought-out curriculum with required courses integrated and dovetailed to complement one another in a consistent manner in order to equip a person to consider problems and make decisions in a constructive way in a particular field. Without such planning, 60 hours or any other required number of courses is meaningless - it could be a hopeless hodgepodge of unintegrated and only vaguely related information."

From the above, it is evident that there is a considerable diversity of opinion concerning the general and specialized education requirements for Shift Supervisors and particularly the need for an engineering or science degree. The emphasis of the NRC appears to suggest that a college degree, at this level of supervision, is essential to safe powerplant operations. If this requirement is implemented, it seems probable that many ROs and SROs will consider their career opportunities to be seriously limited. The long-range impact of this on industry effectiveness and safety is, of course, speculative. It is unfortunate that no objective data are available concerning the formal education of supervisors who have and have not been effective in the industry. Since there does not appear to be a suitable criterion of Shift Supervisor effectiveness available, the debate involves a great deal of subjective judgment on the relevance of a diversity of educational requirements to a very complex performance criterion. However, the recent INPO study appears to be a significant beginning in solving this problem.

Nonlicensed Personnel. Nonlicensed personnel include a variety of operating, maintenance, and technical support personnel. They include auxiliary operators, maintenance technicians, radiation protection technicians, engineers and technical support personnel, chemistry technicians, instrumentation and control technicians, quality assurance and quality control inspectors, shift technical advisors, independent review personnel, and management personnel.

According to NUREG/CR-1750, the NRC endorses the American National Standard for selection and training of nuclear power plant personnel (ANSI/ANS-3.1-1978) with respect to the organizational structure and qualifications of nonlicensed personnel. The qualification requirements for nonlicensed personnel are stated as follows:

> "Nuclear powerplant personnel shall have a combination of education, experience, health, and skills commensurate with their functional level of responsibility which provides reasonable assurance that decisions and actions during normal and abnormal conditions will be such that the plant is operated in a safe and efficient manner."
>
> "Technicians in responsible positions shall have a minimum of two years of working experience in their specialty. These personnel should have a minimium of one year of related technical training in addition to their experience."
>
> "Repairmen in responsible positions shall have a minimum of three years in one or more crafts. They should possess a high degree of manual dexterity and ability and should be capable of learning and applying basic skills to maintenance operations."

With respect to training, it is indicated that technicians and repairmen shall be trained on the job by participation in initial calibration, testing, and equipment acceptance programs, or by related technical training.

It is evident that all of these qualification criteria are too general to ensure the competence of nonlicensed personnel. NUREG/CR-1750 reports that there is a wide range of utility practices employed in the training, qualification, and certification of these personnel. The variations reportedly are wider for nonlicensed personnel than they are for licensed operators. IE has audit responsibility for nonlicensed plant personnel, and this is divided between on-site inspectors and regional IE offices. Regional IE inspections are required to verify annually that overall training activities for nonlicensed employees are in accordance with technical specifications and quality assurance program requirements.

NUREG/CR-1750 notes that all nonlicensed plant personnel have some safety related tasks and responsibilitites. It is concluded that the very general NRC requirements concerning nonlicensed personnel training and qualifications do not provide the necessary assurance that these personnel can adequately perform their safety related tasks. The requirements are viewed as so general that they have little operational value and will not provide the basis for adequate job performance unless the NRC or industry organizations take the lead in further defining these requirements. Apparently (NUREG/CR-1750), the immediate supervisors of nonlicensed personnel at utilities with informal on-the-job training programs have indicated that they, too, feel that these training and qualification programs are inadequate. However, it was suggested that this problem may not be recognized at higher levels of management. NUREG/CR-1750 concludes that in many cases there is no utility commitment to conduct a formal training or qualification program for nonlicensed personnel, and recommends that the NRC require the utilities to certify formally the qualifications of nonlicensed plant personnel, and develop industry wide criteria for this certification.

Anecdotal information suggests that at least some of the operating problems that control room operators experience are the direct result of improper performance by maintenance personnel. NUREG/CR-1750 includes a very general task analysis of the jobs performed by these nonlicensed personnel. More detailed task analyses will be necessary before objective performance standards can be established that will minimize this potential source of safety problems.

NUREG/CR-1280 compares the staffing and qualifications of personnel at nuclear power plants with those employed in the U.S. Navy nuclear program. In many respects, the comparisons are unfavorable to industry. The report notes that there are no eligibility requirements issued by the NRC for maintenance personnel other than minimum periods of prior working experience in their specialty. NUREG/CR-1280 emphasizes that there are wide differences in industry practices in handling the selection, training, and qualification of maintenance personnel. It is stated that maintenance procedures do not usually specify that the work to be performed must be by a person having a particular qualification. "There are no posted lists which specify which craftsman has a particular qualification to do specific work on the plant." These decisions are generally left up to the Foreman or the Maintenance Group Manager.

While each utility company has a training program for its maintenance personnel, it is asserted that the scope, depth, and duration of these programs varies widely among the different companies and among various crafts at a site. The principal training responsibility typically rests with the maintenance trade shops where on-the-job training is concentrated. NUREG/CR-1280 contrasts this situation with the Navy nuclear system in which all personnel who perform maintenance are nuclear trained in addition to their maintenance responsibilities, and have responsibility for operating the systems they maintain. It is felt that by virtue of this training, these personnel have a background in the fundamentals of reactor design, operation, and safety, in addition to the craft training they have received. They are examined on this and other aspects of their duties every year.

NUREG/CR-1280 recommends that personnel who conduct maintenance on any reactor system should be qualified and licensed by the NRC. The license would be based on an affidavit submitted by the utility company attesting to the fact that the individual had successfully completed a classroom course of instruction on:

Basic principles of reactor operation

Basic principles of reactor safety

Reactor systems

Steam systems

Electrical systems

Quality assurance

Radiation protection

Site emergency systems

Industrial safety

It would also be required that the candidate demonstrate that he possessed the necessary trade skill to perform the intended work, and that he have at least three years of experience in the trade skill involved.

PLANNED ACTIVITIES

NUREG-0660 (I.A.3.1) established the objective of upgrading the requirements and procedures for nuclear power plant operator and supervisor licensing to assure that safe and competent operators and senior operators are in charge of the day-to-day operation of nuclear power plants. As a part of this effort, the scope and criteria of licensing examinations are to be revised. Changes that have been made to date include an increase in examination scope to include thermodynamics and other subjects, the imposition of higher passing grades and the requirement for oral examinations for SROs. Simulator examinations were included as a part of the licensing examination beginning in October 1981. Also called for is a continuing program to determine the feasibility of validation and to proceed with validation if practicable.

The NRC's research plan for human factors safety (RR-NRR-81-2 and RR-NRR-81-5) identifies a number of significant research tasks related to improvement of the operator qualification process.

A Study of Operator Examinations was ranked 5th in a list of 16 research tasks. The objectives of this task are to evaluate present RO and SRO examination methods, frequency, and content; to define alternative examination techniques; and to correlate both present and alternative examination techniques with on-the-job performance. This task is scheduled for support pending the availability of FY82-83 funds. It is indicated that industry (INPO and EPRI) have no planned research in this area.

The DHFS on September 4, 81 issued a technical assistance work order to the Oak Ridge National Laboratory for "preliminary evaluation of licensing validity." This effort is to include a review of past and current efforts to validate examinations including non-nuclear fields; a preliminary investigation of methods for relating examination results to operator performance, including methods for measuring operator performance; and a "statistical sampling" of operators to compare examination performance with operations performance.

ORNL is to provide the following technical assistance to NRC in support of these objectives: (1) analyze the way in which the examination is prepared, administered, and scored; (2) statistically analyze a sample of examination scores for the oral and written parts and the written subparts; (3) content analyze the present oral, written, and simulation examination; (4) make interim recommendations for examination process improvements; (5) develop some preliminary approaches to performance measurement in terms of examination validity; and (6) analyze for statistical validity a sample of examination scores against the preliminary performance measures.

A related task also assigned to Oak Ridge National Laboratory is to develop and validate a new licensing examination for ROs and SROs which is "an accurate predictor of operating performance." The various tasks involved in this effort include (1) definition of what constitutes a capable reactor operator; (2) determination of an appropriate examination validation methodology; (3) development and validation of a new examination; (4) development of guidelines for the NRC acceptance and accreditation of training programs and centers; (5) a determination of whether NRC should set trainee selection standards or guidelines; (6) evaluation of utility requalifying examinations; and (7) establishment of a mechanism for periodic program evaluation. This effort is expected to answer such fundamental questions as: (a) how does the examination relate to operator performance; (b) what skills and knowledges should the examination test; (c) what skills and knowledges are best tested by oral, written, or simulator examinations; and (d) how does the examination process affect examination results. This program also extends to questions concerning what the selection criteria for operators should be, and what role the NRC should play in the selection process. An attempt will be made to identify not only what skills potential can be identified by preselection techniques, but what personality factors influence operator performance as well.

NUREG-0660 (I.A.2.2) addressed the training and qualifications of operations personnel. The objective is to increase the education, experience, and training requirements for operators, senior operators, supervisors, and other personnel in the operations organization to substantially improve their capabilities to perform their duties. Though the primary focus of this effort appears to be on training, the suggested methodology has implications for issues related to personnel selection and qualification as well. It is stated that the preferred method for fulfilling this requirement is "position task analysis" in which the tasks performed by personnel in each position are defined, and the skills and knowledge necessary to perform the duties are identified. In addition, it is expected that these analyses will aid in identifying performance-shaping factors that influence the task performance capability of operations personnel.

<u>Validation of Current Educational and Training Requirements</u> for Operators and Senior Operators was given a priority of 6 in a list of 16 items. The objective of this research element is to perform controlled studies to validate the education, training, initial examination and requalification requirements that are currently imposed on Operators and Senior Operators. Both inadequacies or possible excessive requirements are to be identified, as are measures to optimize selection and training requirements. The INPO-sponsored task analysis is seen as relevant to this research objective but it does not provide for a validation of education and training requirements. The NRC's own planned task analysis (NUREG-0660) also requires that

licensees of operating reactors and applicants for licenses must conduct training program reviews and task analyses directed at identifying needed training modifications. The results of these analyses will be used to develop criteria for qualification requirements and training programs for all licensed and nonlicensed operations personnel. The training and qualifications of personnel are to be reviewed and upgraded as necessary to ensure that all operations personnel have a combination of education, experience, and skills commensurate with their functional level of responsibility for safe plant operations.

It should be noted that the NRC's planned activities in this area are directed at determining the best <u>combination</u> of education, experience, and skills commensurate <u>with</u>. . .safe plant operations. The stated program goal is to upgrade the qualifications and training of operations personnel for incorporation as guidance in a revision to Regulatory Guide 1.8, "Personnel Qualifications and Training."

An assignment has been made to Battelle Pacific Northwest Laboratory to conduct research to establish appropriate educational, training, and experience requirements for licensed operators. Apparently because of wide differences in viewpoint concerning the most appropriate mix of prior power plant experience and formal education, a three-pronged approach to the problem has been formulated: (1) establish and coordinate a five- to eight-member peer review panel; (2) plan, conduct, and report on a workshop of 15 to 20 individuals representing various affected groups; and (3) provide an assessment of the state of the art in related fields regarding operator qualification requirements, and the recommendation of the workshop and peer review panel. The workshop activities will include discussions of past proposals, effects on operators of various proposals, effects on the industry, possible alternatives, and methods of implementation. This work began mid November 1981.

<u>Education and Training Requirements</u> for Shift Supervisors, Shift Technical Advisors, and upgrading of Senior Operators is a research task with a priority of 7 in a list of 16 identified by NRR. This task is relevant to the qualification issue since its objective is to develop a consistent set of training and education requirements for the positions noted. It is specifically to address the question of the amount, type, and quality of formal education that is required. Thus, though this task is more fundamentally related to the training area, it will nevertheless have important implications for qualification criteria. The task also requires the development of a definition of "equivalency" requirements for persons who do not hold engineering degress. The information will be used to conform or revise the requirements in 10 CFR Part 55 and Regulatory Guide 1.8. This work is slated for an FY82 start; there are no comparable industry efforts.

NUREG-0660 calls for a complete review of STA training and qualifications for operating plants, an evaluation of the usefulness of the STA concept and experience among the utilities using STAs to date, and an examination of alternative long-range requirements for STAs in view of the possibility (SECY 81-84) that the upgraded education requirements for SROs and Shift Supervisors will eliminate the need for STAs. Battelle Pacific Northwest Laboratory has been tasked to (1) determine current utility practices and experience with STAs through selected utility visits and feedback from the operator workshop; (2) define foreign practices and experience with related programs; (3) assess the effectiveness of the various STA approaches based on these experiences and practices; and (4) recommend near-term guidelines for the STA position.

Plant Maintenance is a research task with a priority of 15 in NRR's list of 16 research tasks. This requirement involves the identification of jobs and tasks in maintenance, test, and calibration that could most directly affect public health and safety. Task analyses of selected job positions and job requirements will be performed. The project addresses, among other considerations, the personnel qualifications necessary to assure that maintenance is performed with minimal adverse impact on safety. The anticipated outcome is the development of regulatory guidance and criteria for training and qualification of maintenance staff, as well as guidelines for effective management control of maintenance activities.

There are relevant industry efforts in this area, including the EPRI-sponsored study of human factors issues in plant maintainability conducted by Lockheed, and a Kinton study of some maintenance tasks in relation to the development of job performance aids. EPRI is contemplating a more thorough maintenance task analysis for support in FY82.

In addition, a technical assistance effort has been contracted for with Battelle Pacific Northwest Laboratories, which is entitled "Maintenance, Human Factors and Procedures Guidelines." This task reflects NRR sensitivity to human errors that have contributed to equipment failures and nonavailability that may adversely affect safety. It reflects the large number of maintenance concerns derived from Licensee Event Reports. The effort will serve as the basis for initiating and/or resolving numerous technical issues related to the training, staffing, and selection of maintenance personnel. Surveys will be conducted to identify conditions, practices, and design and procedural shortcomings that result in mistakes by maintenance personnel. It is intended that the investigation lead to recommendations for the development of guidelines which incorporate human factors concerns.

Included in the study will be a review and assessment of administrative procedures related to equipment replacement and/or modification and an assessment of their contribution to failure rates, nonavailability, and miscalibration. The processes by which the licensee's maintenance procedures were prepared are to be studied, and the adequacy of those procedures and the qualifications of those who developed them are to be assessed. Administrative controls over all maintenance-related activities are to be documented. Particular attention is to be paid to an evaluation of the likelihood that maintenance personnel may induce common mode failures.

Of particular interest with respect to the personnel system, this work is also to address how maintenance personnel are selected and how their proficiency is measured. In addition, there will be a study of factors affecting the motivation of these personnel for doing an effective job.

Requirements for Operator Fitness. NUREG-0660 (I.A.3.3) calls for the development of regulations to provide assurance that applicants for operator and senior operator licenses are psychologically fit (emphasis on stress and malevolence), and to prohibit licensing of persons with histories of drug and alcohol abuse. Previously completed work in the area on "Standards for Psychological Assessment of Nuclear Security Personnel" (NUREG/CR-2075) and on a "Behavioral Reliability Program for the Nuclear Industry" (NUREG/CR-2076) were described earlier. In addition, NUREG-0660 states that the Office of Research is presently sponsoring studies to obtain a clearer understanding of the operator's performance under various conditions such as "stress."

A proposed rule is being prepared jointly by RES and NMSS to establish a screening and continuing behavior observation program for nuclear plant personnel who have unescorted access, including ROs and SROs. The rule consists of three major components:

1. Background investigation.
2. Psychological assessment which includes as a minimum a written psychological test and, for any applicant indicated by the test to have possible emotional problems, an interview with a psychologist.
3. Continual behavioral observation.

Future programs plans in this area are awaiting responses to this proposed rule as well as another one designed to control licensing of operators with histories of drug and alcohol abuse.

MISSING ELEMENTS

All efforts to objectively evaluate and validate current or proposed qualification procedures depend upon the availability of criterion measures that reflect all important dimensions of job performance. As noted elsewhere, objective criterion measures have not been developed on a broad scope for either operator or maintainer personnel in nuclear power plants. There does not appear to be among the NRC's planned activities a direct attack on this problem, although the operator task analyses represent a significant beginning within industry. The EPRI-sponsored research by General Physics Corporation to develop automatic measures of operator performance is clearly relevant and promising, as is ORNL's work on the collection and assessment of performance data associated with safety related operator actions and performance shaping factors. A significant attack on the maintenance problem has been initiated but, once again, a criterion of maintenance personnel performance will need to be developed if clear associations are to be made with selection, training, and qualification variables.

In the short-term, it is evident that performance assesesments of RO, SRO, and Shift Supervisor personnel will continue to depend on subjective evaluations, (i,e., supervisory ratings). The scope and reliability of these procedures is generally undocumented despite their importance to many decisions.

The meaning of operating "experience" is in need of definition since this is a critical variable in all RO, SRO, and Shift Supervisor qualification criteria. Clearly, it cannot be adequately defined in terms of length of service alone.

TECHNICAL FEASIBILITY AND PROBLEMS

The technical problems involved in the validation of operator qualification procedures stem from the need to sample appropriately and record objectively the full domain of performance associated with routine, off-normal, and emergency procedures. As noted, EPRI-sponsored research by General Physics Corporation has moved in the direction of automatically recording operator control room inputs, but this technique is limited in that only the operator's overt responses (i.e., his control manipulations) are recorded. Clearly, much of the behavior of interest lies in the areas of the operator's diagnosis of plant state, his predictions about future states, his fault diagnosis processes, and the decisions that precede his behavioral output. The technical problems associated with capturing operator responses that reflect these complex processes have not been solved. It appears unlikely that the required data can be obtained without somehow recording intervening processes between the input information and output responses of the operators. It may not be possible to do this in real time and it probably will be impossible to do except in a suitably configured control room simulator.

Similar technical challenges are to be faced in developing methods whereby the behavior of maintenance personnel can be tied directly to the maintenance status of the plant. This effort should extend to routine maintenance quality, not just to clear-cut deficiencies or errors. An index of maintenance quality control is needed.

The establishing of an objective basis for trading off increased formal education against operational plant experience is an particularly thorny issue that requires comprehensive performance data for its resolution.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Personnel certification, licensing, and continuing qualification procedures interact strongly with selection criteria, which fundamentally determine the size of the available manpower pool. Because of heavy dependence upon examination procedures, there is a strong interaction with the training subsystem which has been heavily oriented toward the objective of passing those examinations. Of course, all elements of control room design that impact the operator's ability to understand system operation and assess plant state will strongly interact with qualification and training requirements.

RECOMMENDATIONS

1. The NRC should conduct research aimed at the development of objective performance standards for operator and maintenance personnel. Develop methods for routinely evaluating all major dimensions of the job performance of ROs, SROs, and Shift Supervisors. Perform studies necessary to identify and define those dimensions.

Importance: High

Schedule: Urgency - start in 1-2 years

Duration - completion - 3 years

Resources: 3 person-years per year

Implementation: Career human factors specialist; unrestricted use of simulators; access to operating plants.

Dependencies: NRC task analysis; refinement of performance monitoring system.

2. The NRC should conduct research with the objective of developing more specific qualification requirements for non-licensed personnel who are in a position to directly or indirectly impact plant safety.

Importance: High

Schedule: Urgency - start in 1-2 years

Duration - completion - 2 years

Resources: 5 person-years per year; access to operating plants and personnel.

Implementation: Career human factors professional; engineers familiar with plant design and maintenance requirements.

Dependencies: Advantage should be taken of any related work by EPRI or INPO.

3. Research should be conducted leading to the development of methods for assessing and tracking progress through in-plant training programs, with the obective of improving the certification process for licensee candidates.

Importance: Low

Schedule: Urgency - start in 1-2 years

Duration - completion - 2 years

Resources: 1 person-year

Implementation: Training specialist; computer programmer.

Dependencies: None

4. Research should be conducted to define objectively the scope and length of "experience" required prior to qualification of ROs, SROs, and Shift Supervisors, and to provide a defensible basis for trading of formal education against "experience."

Importance: Medium

Schedule: Urgency - start in 1-2 years

Duration - completion in 2 years

Resources: 2½ person-years per year

Implementation: Career human factors specialist; power plant subject matter experts.

Dependencies: Task analyses of control room and auxiliary operator jobs; objective evaluation criteria (see Section 1.8).

4.3 Staffing and Organizational Characteristics

GENERAL SYSTEM REQUIREMENTS

The system requirement is to provide the personnel staffing levels and technical expertise necessary for the utility to properly support nuclear power plant operations and all required plant maintenance. In addition, both management and technical resources must be provided for accident mitigation, including long-term efforts required to return the plant to normal conditions in the event of accident.

CONSTRAINTS

The principal constraint lies in the availability of technical and engineering manpower to meet the staffing requirements. Personnel who are capable of meeting current qualification requirements as ROs and SROs are in short supply in some geographic areas. The off-site technical resources include a substantial number of degreed engineers with two to five years of nuclear power plant experience and five to ten years of overall experience. Given the competition for experienced personnel in nuclear engineering, electrical engineering, computer sciences, mechanical engineering, chemical engineering, and materials engineering, meeting the qualification guidelines of NUREG-0731 may be difficult for some utilities.

PRESENT STATUS IN MEETING REQUIREMENTS

NUREG-0731, issued for interim use and comment in September 1980, provides guidelines for nuclear plant staffing in accordance with various recommendations of studies following Three Mile Island. It describes an acceptable organizational structure and competence levels for nuclear power plant operations. "Competence" is defined in terms of level of formal education and years of relevant experience in the power industry. The guidelines address both on-site and off-site resources and the minimum shift staffing considered essential for short-term and long-term accident response.

NUREG-0731 divides accident mitigation into three time periods: short-term, from accident initiation until approximately one hour after declaration of an emergency; near-term, from approximately one hour after declaration of an emergency until approximately 16 hours; long-term, from 16 hours until plant conditions no longer pose a significant threat to public health and safety.

It is admitted that the formulation of the organizational structure and technical resources that must be available to a plant for both routine and emergency operations has been made on a largely subjective basis. NUREG-0731 attempts, however, to identify desirable goals with respect to these requirements.

These include, for example, independence from operating pressures and reporting in such functional areas as radiation protection, quality assurance, and training; clear lines of authority to the plant manager; clear definition of responsibility for all activities important to the safe operation of the facility; separate supervision or management of distinct functional areas; and sufficient managerial depth to provide qualified backup in the event of the absence of the incumbent. The guidelines cover both routine operations and accident conditions.

Minimum shift staffing is defined in relation to the number of control rooms and operating units. The simplist configuration (i.e., one operating unit and one control room) calls for one Shift Supervisor who is also an SRO, one SRO, two ROs, and two AOs. These staffing requirements are not based on detailed task analyses, but it is likely that they reasonably reflect operating requirements.

Thus far, there do not appear to be pressing research issues or major differences in viewpoint between industry and the NRC with regard to these organizational and staffing guidelines. A notable exception, however, is the NRC's requirement for a Shift Technical Advisor (STA). There are at least two fundamental considerations in the industry response to the STA requirement which appear to be largely negative: (1) creation of the STA position is viewed by operator personnel as a vote of "no confidence"; and (2) the operators themselves lack confidence in the knowledge of the STA about the details of plant operations and, therefore, do not expect the STA to be helpful. Consequently, the STA is seen as occupying a "do nothing," overpaid position. The most charitable description of the STA, coming from industry, was that he or she could be useful as a "knowledge base" as opposed to helping with specific actions. It was felt by some that when operators were working on a problem situation, the STA could be helpful in minimizing the difficulty maintaining an overview of plant condition.

One utility expects to develop STAs from non-degreed Shift Supervisors, using special technical courses developed by a nearby university to meet the educational requirements. It was strongly felt at this utility that unless STAs were developed from the pool of Shift Supervisors, the operators would vote "no confidence" in the STA. The reverse approach is being used by other utilities where attempts are being made to get degreed engineers qualified as SROs. The basic problem with the STA concept is, of course, that there is no objective evidence that the STA's presence will enhance performance. If the STA is to become a viable component of operational effectiveness, it is clear that the STA must be accepted as a contributor. This is not likely to happen unless industry recognizes the need, and in the absence of objective criteria, once again, the question of need is simply a matter of opinion.

NUREG-0731 also identifies the required off-site technical resources for emergency support and provides qualification guidelines in terms of formal education, total experience, and nuclear power plant experience. All of these positions require a bachelor's degree in engineering or related sciences, and from two to five years of nuclear power plant experience. The areas of expertise include transient analysis and system interactions; nuclear engineering; fuel management; core physics; control theory; process computers; thermal hydraulics; plant structural and containment design, etc.

Qualification guidelines are also provided for Site Support Staff personnel whose responsibilities include such areas as fire protection, chemical engineering, radiochemistry, radioactive waste management, decontamination of equipment, radiation control, plant operations, and plant maintenance. With the exception of fire protection and decontamination of equipment, most of these staff positions require a BS degree, and from two to four years of experience in nuclear power plants. There appear to be no urgent research issues with respect to the necessity of these general personnel qualifications.

Each management's off-site staff is to be capable of full functioning within four hours after an accident. For plant operations, an engineering degree is not specified but the individual in charge must hold, or have held, an SRO position in a plant by the same vendor as the one to whom assigned, and must have had five years of nuclear power plant experience. The Plant Maintenance Engineer must have a BS degree in engineering, five years of nuclear power plant experience, and ten years overall experience.

NUREG/CR-1656 reports an analysis of licensee submittals in response to NRC inquiries concerning management and technical short-term and long-term resources for reacting to TMI-2 type accidents. The study was directed at acceptance criteria for minimum management and technical resource needs, and evaluated the adequacy of licensee resources, both off-site and on-site, for coping with nuclear power plant events similar to the TMI-2 accident. This report concluded that resources at the various utilities reflected a general responsiveness to H. R. Denton's letter of June 29, 1979 concerning utility resources for handling accidents, the NRC lessons-learned task force reports, and the AIF report, "Nuclear Power Plant Emergency Response Plan." The industry viewpoint toward required off-site resources and plans for the use of multiple off-site resources when dealing with a TMI-2 type accident were regarded as favorable, or even impressive. With respect to emergencies, it was found that both the procedures and numbers of personnel available were acceptable. However, there was some uncertainty concerning whether the personnel had the skills needed to understand and implement all procedures properly.

NUREG/CR-1656 found the following weaknesses to be common to most of the utility resources:

> The failure to discuss in any depth the shift supervisor position or those filling this position. This position was viewed by the authors as of as much importance to the safe operation of the facility as the plant manager. The omission of data in this area was interpreted to mean that perhaps some utilities did not understand the importance of the shift supervisor.
>
> Insufficient advanced planning and procedures. It was felt that the submissions from the utilities showed little progress in this area.
>
> A lack of depth in certain required areas of expertise. It was noted that these weaknesses may require reassessment once the NRC has established acceptance criteria. It was considered possible that the numbers of personnel are adequate, and the areas of expertise/skills covered by these personnel, but this depends on criteria to be established. (As noted above, NUREG-0731, which was issued for interim use and comment, specified these criteria in terms of required formal education, amount of total industry experience and amount of nuclear power plant experience, or various key resources in the organization.)

In NUREG/CR-1280, the qualifications of the senior on-site Manager, or Plant Superintendent, are contrasted with his Navy counterpart who is viewed as the ship's Commanding Officer. It is stated that the major difference between the utility's on-site Manager and the Commanding Officer in Navy ships is that the latter is required to exercise sole responsibility for the safe operation of the plant, 24 hours a day, 7 days a week. There are certain decisions relating to the reactor that he and only he can make -- they cannot be delegated. He is always just moments away from the control room and within easy, direct communication with the others involved. However, while the Commanding Officer generally has full authority on his ship, he does not have authority, except in emergency situations, to deviate or change any officially transmitted requirement or procedure relating to the operation or maintenance of the reactor plant. It is noted that this is not the case in civilian nuclear plants. Rather, the senior on-site Manager has the authority to change the plant design or an operating procedure if he himself is satisfied that it is technically correct. The authors of NUREG/CR-1280 feel that this practice is wrong, and that the question of who in the utility organization is authorized to approve changes to the design or procedures needs to be clarified and justified. It is felt that, except under emergency conditions, "appropriate" technical review and approval, external to the senior on-site Manager should be required.

PLANNED ACTIVITIES

NUREG-0660 (I.B.1.1) calls for the development of criteria for on-site and off-site organizations, both management and technical, including the Radiological Protection Organization, that will assure the safe operation of the plant during normal and abnormal conditions and the capability necessary to respond to accident situations. It identifies the need to specify the qualifications and experience of management, technical staff, and safety review groups, both on-site and off-site, including the interactions of these groups to assure effectiveness and avoid duplication of effort. The scope of I.B.1.1 includes defining:

1. The duties and responsibilities of key personnel (except for shift supervisor responsibilities which are covered under I.A.1.2 and I.C.3).

2. The size of off-site staff, types of expertise needed, and the degree of their involvement in plant operations.

3. Pooling of resources among utilities to provide the operation's staff with a means to acquire prompt expert advise from off-site sources.

4. Organization arrangements for both normal and accident situations.

5. Implementation of pre-established plans for using available resources in the event of unusual situations.

A revision to NUREG-0731 is in process, reflecting the experience with this document to date and industry comments. It was planned that the revision will be submitted to the Commission by the end of FY81. If so indicated, a revised NUREG-0731 was to be issued to industry for action during the first quarter of FY82, and implementation was scheduled by the last quarter of FY82.

A contract has been established with Battelle Pacific Northwest Laboratory for research leading to the development and recommendation of guidelines for licensing actions regarding manpower and staffing of nuclear power plants. This includes consideration of both on-site and off-site staff requirements with emphasis on numbers of people by type of job, staff qualifications, work schedules, and overtime requirements. Among the tasks to be performed are a review and evaluation of current literature regarding manpower and staffing requirements; an examination of current industry staffing practices both in the U.S. and abroad; an assessment of the effectiveness of current staffing practices with emphasis on the advantages and disadvantages of different approaches that are used; and the definition of a program plan for developing guidelines to address priority manpower issues. The research will define what job functions are required for safe plants, how many people and what types of jobs are required; what staffing configurations are effective, including structure for work flow, communication, and control; work schedule alternatives; manpower needs and labor supply; career paths for the plant staff; and how shift staffing affects the role of the STA.

RR-NRR-81-2 (H. R. Denton) notes that present methods of assessing the capability of a utility organization to effectively and safely manage a nuclear power plant are quite subjective. It is recognized that there is a need for guidelines and methods for making such assessments in a valid manner. There are few data dealing with the attitudes of nuclear power plant management toward safety, and a need is felt for systematic study of those elements of management and those indicators of effectiveness which are important for assessing utility management qualifications from the standpoint of assuring safe operations.

Planned activities include the development of measures reflecting safety attitudes and behaviors of the plant staff and a standard information collection system for management review, including interview guides, observation, and information-recording forms. The review system would be pilot tested on selected utilities to obtain indications of its feasibility and effectiveness. The methodology would include individual and group interviews at utilities, NRC ratings of management elements, and a retrospective look at construction safety in new plants and operational safety in older plants managed by the utility. In the longer term, it is hoped that it will be possible to relate effective and ineffective management behaviors to safety criteria through an appropriate model.

NUREG-0660 (I.A.3.4) calls for a determination of which plant personnel, other than ROs and SROs, may need to be licensed by NRC. The personnel to be considered include managers, engineers, auxiliary operators, maintenance personnel, technicians, and STAs.

Section 307(B) of Public Law 96-295 directed NRC to study the feasibility and value of licensing plant managers and other senior licensee officers. A technical assistance contract has been placed with ORNL to study this issue. Information will be gathered from individuals having expertise in, or experience with, executive assessment techniques from utility management itself, and from the NRC and other appropriate federal agency staff. The inquiry will focus on such questions as what are the technical and managerial/administrative task/job elements of targeted personnel; what criteria can be imposed to evaluate performance on these job elements; is such licensing needed, practical, effective; and which of the targeted positions should be licensed. Both industry and NRC viewpoints are to be examined. Some key observations to date include the fact that a certification program is preferable to licensing in industry's view; the managerial/technical dichotomy may be misleading and there should be an increasing emphasis on management knowledge, skill, and ability as one moves vertically in the organization; having held an SRO or RO license should not be a requirement for nuclear power plant managers, but they should have undergone the training program; necessary management attributes may be difficult to define and measure (e.g., planning skills, communication skills, public relation skills, etc.); written tests of technical or managerial knowledge should be avoided; current NRC assessment procedures applied through the management review process are probably adequate to identify management problems; the assessment process possibly should include oral board's review, peer review panels, record and background reviews, etc.; periodic recertification should not be required; management assessment procedures should be linked to a thorough task analysis of the specific positions; the NRC should consider certifying the management development programs employed, rather than the individual managers; the licensing/certification programs should be industry developed, administered and policed, preferably by INPO; and the NRC should assist in the development of program criteria and have an audit function.

The program plan in this area calls for the completion of the feasibility study for licensing managers and senior licensee officers during FY81. A licensing program, if required, will be developed during FY82-FY85. In the meantime, additional study of the feasibility and value of licensing other targeted personnel is scheduled for completion by the end of FY83.

A part of the license application process involves an assessment of management and organizational resources based on NUREG-0731 by LQB. A contract has been let to Battelle Pacific Northwest Laboratory to develop and recommend guidelines for NRC licensing actions relating to utility management and organization for nuclear power plant construction and operation. The near-term objective is to recommend improvements in NUREG-0731. The work will involve the development of preliminary guidelines and their pilot testing during one or more plant reviews, revision of the guidelines, and further pilot testing if necessary. The effort includes a review and assessment of existing criteria for management and organization, and the development of revised criteria as necessary. Key issues in the assessment process will be addressed, including its reliability and validity (whether the judgments by different evaluators are the same, whether they use similar evaluative processes, whether they interpret the guidelines in the same way, etc.). Also to be addressed is the question of whether the utilities would be capable of effectively adjusting to revised criteria if they call for substantial changes in present organization and management practices.

MISSING ELEMENTS

There are no suitable intermediate criteria by which to judge management attitudes toward safety or the impact of organizational variables upon safety.

TECHNICAL FEASIBILITY AND PROBLEMS

A significant technical problem associated with the proposed studies of management attitudes, management practices, and organizational variables is that of validating proposed changes against safety criteria. Safe operations need to be operationally defined in terms of appropriate behavioral indices at all levels in the organization. Since accidents are still rare events, it is likely that intermediate criteria of operational safety will have to be developed before the effects of various organizational variables can be related to safety. Improved methods of LER reporting (see Section 1.5, this Volume) may be one avenue of approach.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Staffing and organizational characteristics clearly interact with manpower supply, training requirements, qualification standards, normal and emergency procedures, shift duration, design of the control room, design of the overall plant for maintainability, and design of facilities for handling emergencies.

RECOMMENDATIONS

1. The NRC should conduct research aimed at the development of criteria whereby the effects of staffing and organizational variables can be objectively assessed. Consider not only performance measures (see Section 1.8) but secondary criteria that may reflect safety-related management attitudes, (e.g., procedures for maintenance quality control, various operating rules, work schedules and amount of overtime, provisions for operation relief from boredom, methods used for feedback of industry operating experience, level of understanding of human-related safety issues, and so forth).

2. Once suitable behavioral indices of safety-related attitudes have been agreed upon, conduct research to identify the extent to which plant management practices differ on these indices and determine methods for generating desirable changes.

Importance: Medium

Schedule: Urgency - start in 3-5 years

Duration - 1 year

Resources: 2 person-years

Implementation: Career human factors professionals and/or industrial/organizational psychologist

Dependencies: Progress on development of evaluative criteria (see Section 1.8).

4.4 Shift Duration and Rotation

GENERAL SYSTEM REQUIREMENTS

The system requirement is to provide full staffing 24 hours a day, 7 days a week with operating personnel who are not only competent technically but who are fully alert to indications of changing conditions that may be symptomatic of preaccident states.

CONSTRAINTS

The shortage of qualified operator personnel in some geographic areas makes it impossible for the utility to meet the requirements for continuous operations without longer than normal industry work shifts. It is reported that the more stringent examination cut-off scores, which have produced a higher failure rate among operators up for requalification, have contributed to this problem.

There are inescapable periods of tedium associated with routine control room operations. The requirement for round-the-clock operations makes it unavoidable, as long as rotating shifts are used, that operational personnel will periodically be exposed to circadian depressions in the level of central nervous system arousal.

PRESENT STATUS IN MEETING REQUIREMENTS

The power industry is one of the few that requires full operational staffing 24 hours a day, 7 days a week. A natural consequence of the required high level of technical qualifications and current personnel shortages (in some areas) is pressure from industry for shifts of comparatively long duration. Although many plants utilize conventional 8-hour shifts, a significant number employ 12-hour shifts, and under certain circumstances personnel may work considerably longer than 12 hours before being relieved. There are three fundamental safety related questions to be answered:

1. How long can plant operators continuously perform without suffering some degradation in performance? Is the degradation serious?

2. What is the optimal work/rest pattern for a given shift?

3. What is the optimal shift rotation pattern to avoid cumulative fatigue and minimize probable circadian effects on personnel?

Each of these issues is complex and relates not only to safety of operations but to the morale of the working force.

The NRC has provided guidelines with respect to permissible overtime in NUREG-0731 but has not as yet officially addressed performance problems that might occur during routine work days of accepted length. Generally speaking, these guidelines stipulate that individuals shall not be permitted to work more than 12 hours straight; that an individual shall not work more than 72 hours in any 7-day period; and that an individual shall not be required to work more than 14 consecutive days without having 2 consecutive days off. Under certain circumstances, deviations from these rules can be authorized by the Plant Manger.

To date, no confirmatory research has been conducted to indicate whether these recommendations are good, bad, or indifferent with respect to operational safety. Evidence from other industries would suggest that significant performance degradation can occur in considerably less than 12 hours of continuous operations, but there has been no demonstration that this occurs in nuclear power plants. The matter is complicated by the pattern of rest breaks and opportunities for diversion from plant monitoring by periodic assignments to other responsibilities. Satisfactory research on this issue, of course, requires a performance criterion that is sensitive to the effects of fatigue and boredom and, as noted elsewhere, suitable performance measures of continuous operations have not been developed.

There is considerable diversity of opinion expressed within industry concerning both shift durations and shift rotation patterns. Those plants that have experienced a shortage of qualified operating personnel are quick to rationalize the advantages of 12-hour shifts. These include the fact that the operators have grown used to the additional pay they receive for overtime work, that they do not suffer performance degradation because they can still "answer questions sensibly" when coming off duty after 12 hours, that there is one fewer plant turnover every 24 hours, and that the plant personnel are kept alert during the night shift by virtue of the amount of surveillance work that must be performed (i.e., the problems of boredom are averted). The potentially adverse effects of circadian rhythms are recognized by some elements of industry with the result that in at least one utility there is consideration of changing from 7-day shift rotations to 30-day shift rotations. It is anticipated that this plan will be acceptable to plant personnel.

Despite these assurances, it is evident that certain industry practice compounds the possible adverse effects of long shift durations and adverse circadian influences. Whether or not serious performance degradation occurs under these circumstances is, as noted, unknown and suitable research remains to be performed on this issue. It is notable, however, that some regional inspectors report that the only incidents of "true" operator error they had ever observed occurred during the midnight shift.

NUREG/CR-1764 addressed the question of the relative merits of an 8-hour rotating shift versus a 12-hour rotating shift with a shorter work week (4 days per week rather than 5). This study was based on a review of laboratory research and field studies from other industrial and military settings. The implications of these studies for nuclear power plant shift durations were viewed as difficult to establish with confidence, but it was concluded (page 35) that:

> "A conservative interpretation of the studies reviewed in this section would suggest that a change from an 8-hour rotating shift to a 12-hour rotating shift is not a particularly desirable option, and is likely to result in certain types of performance decrements and a decrease in 'performance reserves'. Performance on the night shift is likely to be especially affected by the combined effects of longer working hours and larger circadian rhythm disruptions. Whether these changes in performance have operational significance in the reactor control room environment is unknown."

The authors of NUREG/CR-1750, in noting the contrast between Navy watch schedules and shift durations in nuclear power plants, commented as follows:

> It is our experience that many Navy watches, particularly in the Navy nuclear submarine program, are 6 hours in length and it is not uncommon for personnel at some watch stations to be on a "two-section rotation" (i.e., 6 hours on watch and 6 hours off watch). While we do not find fault with 8-hour watches currently assigned to plant personnel at some utilities, we do not agree with the practice of some utilities that have personnel stand back-to-back 8-hour shifts when an operator from a

relieving shift does not report due to sickness, etc. Utilities should have firm requirements that personnel stand no more than 12-hour shifts, and that they have at least as much time off between shifts as the length of their shifts. (p. D-12)

The problem of shift duration is compounded by shift rotation practices that require personnel to work on irregular schedules with respect to the 24-hour clock. Reportedly, virtually all of the utilities follow this practice. A particular group of operating personnel will be on the day shift for some defined period (a few days to a week in most instances), then change to the evening shift for a similar period, and then to the midnight shift. After some period of time off, the cycle is repeated.

Ehret and Cahill (30) have made a survey of shift rotation practices in the industry and have concluded that those most commonly in use are those that combine slow rotation with phase advance. They conclude, from both a theoretical point of view as well as experimentation with animals using various rotation protocols, that these conditions assure bad "circadian chronohygiene" and associated poor performance with respect to visual acuity and cognitive requirements of operational tasks. Rotation schedules involving a shift phase change daily, or every other day, are considered to be particularly undesirable though they are widely used. Slow rotation with phase delay is considered better than slow rotation with phase advance.

The influence of natural circadian rhythms on the performance of personnel in a variety of industrial and military settings appears to be quite well established. The general finding is that people on night shifts who are not adapted to working during non-daylight hours are increasingly prone to losses of alertness and degradation of performance, particularly between midnight and 6 AM, with the effect becoming most severe around 4 AM for most people. The more routine and tedious the task, the more likely it is that degradation will occur. Adaptation to a new shift is possible, of course, but circadian effects are persistent and may require several days, or even weeks of work on the new shift before personnel are adapted to the new time period.

Certain nuclear industry practices would appear to compound the possible adverse effects of long shift durations and adverse circadian influences. For example, in one plant visited a four-day 12-on, 12-off watch cycle was employed with phase advance and shift rotation occurring after either one and a half or two and half days off. Using this scheme, the personnel are exposed to quite lengthy watches and the night shift, which begins at 7 PM and ends at 7 AM, combines lengthy watches with the worst period of circadian influence. Whether or not consequential performance degradation occurs under these circumstances is, as noted, unknown and we know of no suitable research that has been peformed in nuclear power plants to address the issue.

PLANNED ACTIVITIES

NUREG-0660 (I.A.1.3) addresses the problem of ensuring the necessary number and ready availability of personnel to staff nuclear power plant operations shifts. The program plan includes (a) an investigation of the need to develop requirements for the total shift complement, as opposed to solely licensed and unlicensed operators and (b) studies regarding shift length, shift rotational schemes, and the use of overtime. The latter are to reflect the results of RES studies and other information, and feedback from industry, to develop a revised staff position on shift staffing, if necessary, and issue guidance to industry. This work is to continue through FY83 with guidance to industry scheduled for mid-FY84.

It is recognized within the NRC that there is a need to establish a "more rigorous scientific basis" for deciding what is acceptable from the standpoint of shift length, shift rotational schemes, and the use of overtime in nuclear power plants (Denton memorandum, March 27, 81). A program is called out to (1) critically evaluate what is known about shift length, shift rotation, and overtime use on operator performance and (2) conduct basic research, as needed, with human subjects to determine optimum methods for minimizing the deleterious effects of shift rotation, length, and overtime on operator performance.

The NRR priority list showed this item as number 10 in a list of 16. Current work includes in-house analysis of LERs to determine their relationship to shift work variables. An earlier review of overtime data proved "inconclusive." LERs are being examined for potential correlations between error frequency and time of day. The RES staff has drafted an experimental design to examine effects of shift work and rotation schedule but, prior to its implementaton, they are awaiting additional evidence that working hours are "dominant performance-shaping factors." It is hoped that some of this evidence will be forthcoming from outside programs (such as NASA's study of pilot fatigue).

The Department of Energy is sponsoring work at Argonne National Laboratory on how to adapt most rapidly to a new shift with minimum disturbance. In particular, these investigators are studying dietary practices which are designed to accelerate circadian phase shifts. This work, which is being done on animals, investigates the influence of consuming meals on "days off" of a type and at times which anticipate the next shift schedule.

MISSING ELEMENTS

The key element in these planned activities is the need for development of a performance criterion measure that is sufficiently sensitive to detect changing levels of operator alertness as a function of time on the job and time of day. While it is certainly worthwhile to try to relate LERs to shift duration, overtime, and possible circadian effects, failure to detect these influences in the LERs may simply mean that the LER reporting system is not properly designed to detect them. Or, as in the case of accident research in other industries, it may be that only a very carefully defined subset of operator behavior is related to shift duration and rotation practices.

TECHNICAL FEASIBILITY AND PROBLEMS

It is a considerable technical challenge to develop criterion measures that are sensitive to momentary lapses of attention or progressive changes in levels of alertness. Ideally, the performance measure should be derivable from routine control room activities. Since the operators are not engaged in continuous control room activity, however, some form of secondary task may have to be introduced. There is a substantial literature on this technique, which generally indicates that considerable sophistication must be used in introducing a secondary task if the investigator is to avoid intrusion, i.e., changing the phenomena he wishes to observe.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The problems associated with shift duration and rotation and the associated problems of maintaining operator alertness interact with selection criteria, qualifications criteria which fundamentally limit the supply of operator manpower, and with the human engineering of control room displays, which can be designed to some extent to circumvent operator loss of alertness to changing plant conditions.

RECOMMENDATIONS

Loss of operator alertness and any adverse change in cognitive functioning must be viewed as one of the most consequential performance shaping factors in nuclear powerplant operations.[1]

1. Research should be conducted aimed at determining whether and under what conditions operator performance in the control room measurably deteriorates. Particular attention should be directed at identifying cognitive variables (information processing), performance measures, and physiological indices that are likely to be sensitive to loss of alertness and cumulative fatigue (see Section 1.8).

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1. A recent review of research on human vigilance (67) suggests that the answers needed by the NRC and the power industry on performance degradation as a function of watch schedules are not likely to be found in existing research literature. This is essentially the viewpoint also expressed in NUREG/CR-1764.

2. Work should be performed on the (necessary) redesign of LER reporting methods so that an appropriate data bank of events can be related to various independent variables logically associated with shift length, work/rest cycles, and shift duration.

3. Assuming that performance deterioration is documented, the NRC should conduct research to identify variables that influence its severity. These variables should include, but not be limited to, shift duration, shift rotation schemes, procedures for alleviating boredom, and so forth.

Importance: High

Schedule: Urgency - start within 1-2 years

Duration - 3 years

Resources: 10 person-years; access to control room and control room personnel

Implementation: Qualified human factors specialist or experimental psychologist; work physiologist

Dependencies: Development of unobtrusive sensitive measures of performance deterioration or long alertness. Cooperation of management, union, and the personnel themselves.

4.5 Factors Affecting Job Satisfaction

GENERAL SYSTEM REQUIREMENTS

The complexity and critical nature of nuclear power plant operations and maintenance requires not only a highly skilled work force, but also a highly motivated one. Staff stability is important because the loss of skill and plant knowledge associated with high turnover rates may adversely impact safety of operations. The requirement is for a stable, motivated staff that is willing to live with long hours, shift work, usually tedious (but sometimes highly stressful) conditions, and a certain degree of non-acceptance by the general public.

CONSTRAINTS

The requirement for 24-hour operations, 7 days a week necessitates shift work and, in some instances, long shifts and considerable overtime. Some plants are in locations that restrict the social or cultural activities of the family. There are limits to economic reward. Requirements for frequent requalification are viewed by some as placing their livelihood in jeopardy.

Career development may depend on educational credentials that are viewed as difficult to achieve.

PRESENT STATUS IN MEETING REQUIREMENTS

Certain management practices have reportedly led to a lack of job satisfaction and attrition of qualified personnel. Classifications of operator errors include those attributable to fatigue or loss of alertness. As discussed earlier, current shift durations and rotational practices do not preclude fatigue and vigilance problems as possible sources of operator error, and indeed might be viewed as possible contributors to operator "negligence" or "errors in judgment."

ANSI Standard N524-1967 (N18-20) describes a "nuclear plant reliability data collection and reporting system." This is a voluntary reporting system involving participation by industry, the Government, and the public. A proposed rule making action (January 30, 1980) would make this system a mandatory requirement for all facility licensees. Consideration is being given to the inclusion of man-machine interface data and to a reliability analysis which takes human factors into account.

NUREG/CR-1750 concludes that the NRC has not taken a strong leadership position with respect to operator error reporting. It is felt that the utilities have not been required to identify the "root causes" of personnel errors, and, therefore, the corrective actions taken have not necessarily been adequate. It is further concluded that no effective means has existed to disseminate the lessons learned to plant operator personnel.

PLANNED ACTIVITIES

Item number 13 in a priority list of 16 NRR research needs identified in RR-NRR-81-2 and RR-NRR-81-5 is directed at an assessment of the impact of post TMI requirements on operators. This is to include a survey of RO, SRO, and key personnel turnover from 1975 to date and anticipated turnover rates for the period 1983 to 1990. Likely causes of turnover, including those associated with NRC actions, and recommended methods for minimizing those causes are to be identified. This task also includes a feasibility study of "job stress effects" on performance.

A DOE industry manpower study, due to be completed in early 1982, will be examined for statistical trends in turnover. There is some doubt within the NRC that this survey will produce much reliable information on the causes for leaving. In addition, INPO and NRC/IE plant evaluations will be monitored for evidence of attitudes of utility personnel and management reflecting personnel problems. The RES staff has developed a plan for a broad industry survey, but there currently are no plans to implement this activity. It is preferred that the industry take the lead in this area.

NUREG-0660 (I.C.5) calls for the review and revision of licensee procedures to assure that important operating experience, originating both within and outside the organization, is continually provided to operators and other personnel, and is incorporated into training and retraining programs. This task was to have been implemented by operating plants by January 1, 1981, and applicants for new operating licenses must have appropriate procedures in place prior to fuel load.

It was noted in NUREG-0660 that a problem with this effort is the vast amount of data being generated (LERs, operating experiences, NRC bulletins). Thus, the screening of these data to determine what is applicable to a particular plant is a significant task. The task was originally viewed as one to be assigned to the Shift Technical Advisor but some utilities have used other personnel to perform this function. The NRC is working on an agreement which would allow INPO/NSAC to perform most of the necessary screening, thus allowing the utilities to concentrate their resources on the evaluation of "significant items." It is noted that it will be necessary to develop criteria for the minimum utility program needed to implement the feedback requirement.

MISSING ELEMENTS

From the data presented in NUREG/CR-1750 it is evident that there may be a number of sources of dissatisfaction among operator personnel. While some of these may be associated with the NRC's post-TMI requirements, there also may be a considerable number of other reasons associated with various management practices. Further, the severity of various problems leading to dissatisfaction may vary with the category of personnel, with plant location, and other variables. Valid data on motivation and feedback issues, known to be representative of the industry and different categories of personnel, are lacking.

TECHNICAL FEASIBILITY AND PROBLEMS

While it is technically feasible to develop measures of personnel attitudes and job satisfaction, the validity of such measures is a function of how skilfully the investigators deal with emotionally tinged issues, the representativeness of the respondents, and their willingness to be completely candid. Success of the methodology is obviously dependent upon sound procedures for sampling the appropriate populations of utility personnel in various job categories, and methods of inquiry that ensure the anonymity of the respondents.

With respect to the validation of job satisfaction data, the most relevant criterion, of course, is turnover rate. Obviously it would be desirable to develop intermediate measures that are predictive of this undesirable outcome so that appropriate and timely corrective actions can be initiated by management.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

Job satisfaction is fundamental to staff stability and performance quality control. It interacts with a variety of management practices associated with career development opportunities, reward systems, requalification requirements, overtime requirements, shift work requirements, and performance feedback.

RECOMMENDATIONS

If industry does not take the lead in research leading to the minimization of turnover and maximization of job satisfaction (particularly among ROs and SROs), it is recommended that:

1. Research be conducted by the NRC to establish recent turnover rates and rates that are predicted for the next 2-3 years throughout the nuclear power industry. (Attention should be directed to the distinction between personnel who actually leave the industry versus those that simply move to a new position within it.)

2. If those rates are judged to be excessive in relation to safety considerations, the NRC should perform reseach to identify causes and changes in industry or NRC practices that would be necessary to significantly reduce them. Identify the reward/feedback/professional growth structure necessary to minimize job dissatisfaction and maximize stability.

Importance: High

Schedule: Urgency - immediate

Duration - periodic assessment of trends is necessary

Resources: 2 person-years per year

Implementation: Career human factors specialist and/or industrial/organizational psychologist

Dependencies: None

5.0 PROBLEM AREAS IN TRAINING

The training system is bounded on one side by the skills and knowledges required for each plant job as determined by the plant design and operating parameters and procedures. On the other side, it is bounded by the licensing and other quality control systems that are employed to assure that those skills and knowledges are met and maintained. The focus of the NRC must be on these two aspects of the training system. The particular training methods and training equipment used for the training system itself should be the prerogative of industry. However, since the NRC cannot predict the adequacy of behavior, but can only assess it after its occurrence on the job or during examinations, the NRC should take some role in ensuring that training programs are comprehensive so that there will be a high likelihood of adequate job performance by plant personnel at all times.

5.1 Instructional Systems Development

REQUIREMENT

In much the same way that the man-machine interface should be developed using systematic analysis, so should the training components of the nuclear power system. By definition (from the Air Force Manual 50-2), the process of Instructional System Development (ISD) is "a deliberate and orderly process for planning and developing instructional programs which ensures that personnel are taught the knowledges, skills, and attitudes essential for successful job performance." For small systems requiring relatively few trainees per year, the formal ISD process is usually too expensive and time consuming to employ. Such is not the case, however, in the nuclear power industry which is characterized not only by high training and refresher throughput rates, but also by the high costs incurred by damage or disruption of the nuclear power plant.

Effort put into training development and implementation has payoff in reduced operating costs, maximal public safety, and potentially high levels of job satisfaction of the employees. ISD initiated at the time of plant design accrues the benefit of economies in utilization of the same data bases needed for other human factors efforts, access to subject matter experts knowledgeable in the intricacies of plant system interactions, and the acquisition of well-trained personnel as they are required.

ISD is the assembling of facts and assumptions in a manner that allows the iteration of decisions through a trade-off process. Although the formalized ISD process has many steps (cf.

NAVEDTRA 106A or MIL SPEC 29053B), at least a summary of the information and decision points that are useful to the training developer is necessary for this discussion. The process is also summarized in Figure 2. The description that follows is based on the Air Force Pamphlet 50-58. The initial steps differ from "traditional" ISD models in the increased emphasis on hardware and system-oriented data.

The major input is an analysis which differs from a human engineering task analysis only in the emphasis or inclusion of training related data, such as:

a) the functional as well as chronological sub-task sequence,
b) sub-task initiation and completion cues;
c) criticality;
d) difficulty;
e) probability;
f) personnel interactions;
g) performance limits;
h) most likely errors;
i) malfunction/contingency tasks;
j) affective skills; and
k) strategies.

This information, coupled with the functional descriptions of the plant hardware and software subsystems, provides the training developer with sufficient data to develop a set of job-oriented behavioral objectives. These objectives (as opposed to training-oriented objectives) are statements of what each category of trainee (RO, technician, etc.) is expected to do on the job in terms of behavior regarding the task to be performed, the conditions under which it is performed, and the acceptable performance limits. In addition to this information, which is taken almost directly from the task data (but may collapse across several similar tasks), one may add enabling objectives and ancillary objectives. Enabling objectives are the skills and knowledges needed to carry out the behavioral objective under the stated conditions. Ancillary objectives are the contingency actions to be performed to initiate responses to off-normal and emergency conditions.

The training objectives are the net result of modifying the behavioral objectives via the consideration of the trainee selection process (under ideal conditions, the latter is an iterative trade-off with the consequent training objectives) and the external influences and operational policies. The trainee selection process is based on NRC regulations for licensing, if any; union or other management policies; and the available sources of trainees and their related incoming skills and knowledges. These factors determine what may be expected to already be in the behavioral repertoire of the new trainee as a function of

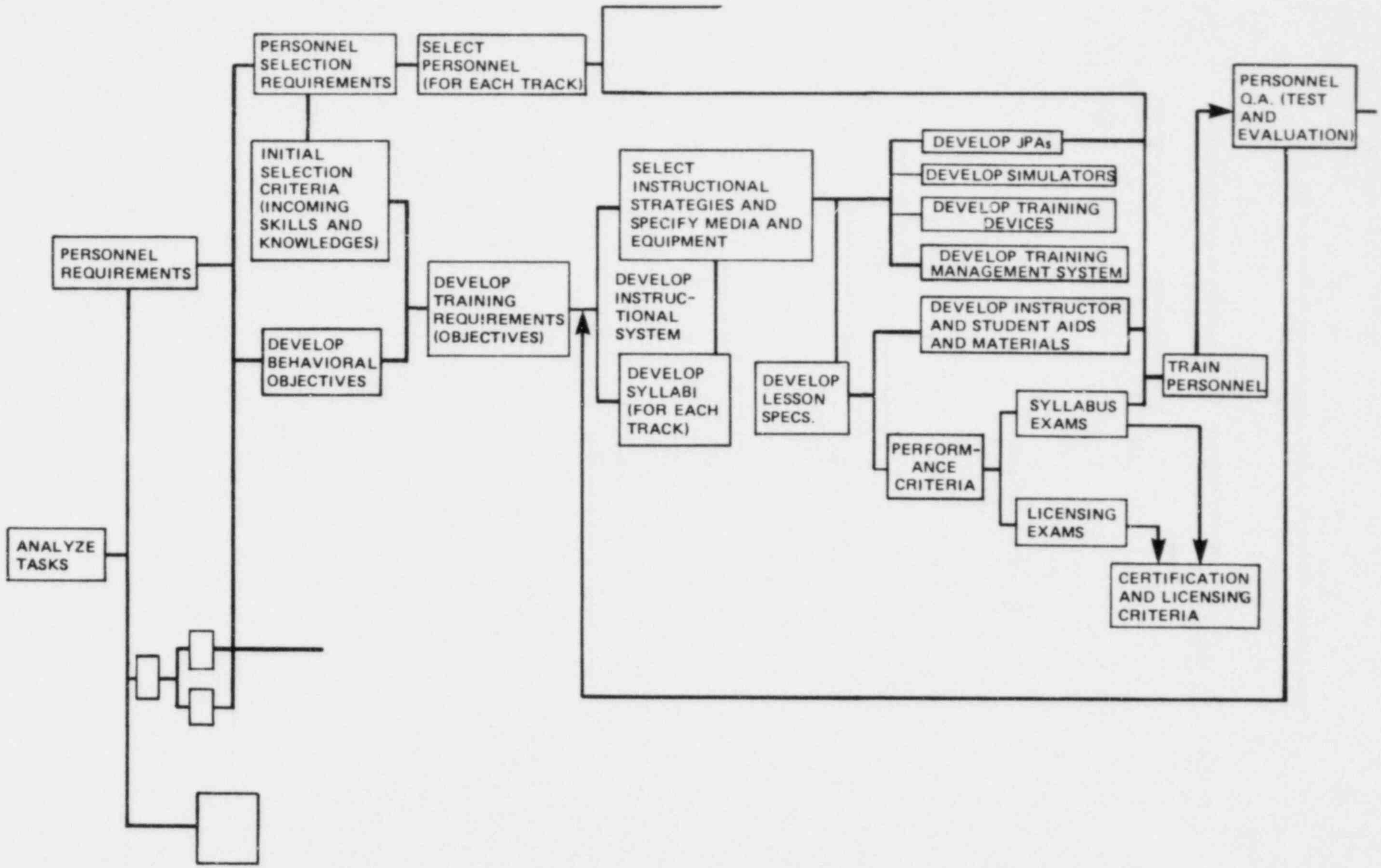

FIGURE 2. TRAINING SYSTEM FLOW DIAGRAM

his/her plant function (i.e., job). Obviously, if one can draw all ROs from nuclear Navy assets, then the training objectives will be few compared to entering trainees who come up the ranks through maintenance roles. The external influences modify the behavioral objectives by the application of a variety of constraints including personal safety around sources of radiation ("conditions" may then call for simulants or simulators) and economic factors such as the unavailability of operational plants for training. Hence, training objectives are "who, what, when, how well" statements in terms of what can be accomplished in a training situation.

Operational policies may further dictate the proficiency required at graduation (the remainder to be learned on the job), training locations, budgets, schedules, priorities, and other factors which may constrain the training. (It is, of course, fortunate that human learning is "robust" with respect to the techniques and plans used to teach -- a well-motivated trainee can adapt to the most marginal of training programs.) Training objectives may be characterized by the behavioral objective(s) that will be met, the training environments that are needed to support the learning process including the systems or stimuli that are required to elicit the desired responses, applicable teaching techniques (e.g., lectures, hands-on trainers, computer-aided instruction, demonstrations, etc.), applicable performance assessment techniques, phase of training priorities, and level of proficiency to be required at the end of each phase of training. Also determined are those objectives that, by virtue of their low criticality, difficulty, and/or frequency, may not be selected for training, but rather will be satisfied through the development of job performance aids.

One can see that even before the first lesson is written, a well-documented trail of decisions can be laid down that ties the training directly to the job, the trainee selection requirements, and the other system requirements. The construction of the syllabus for each course follows a similarly logical process to sequence the learning events in a manner that enhances motivation, optimizes training resources utilization, and facilitates efficient attainment of the objectives. As a parallel part of this process, the instructional strategies are selected (with implications for the skills required of the instructional staff), and the training support equipment and devices are specified.

This ISD process is applied to the development of curricula for each training track. Tracks are unique programs of instruction based on plant position (operator, technician, maintenance, etc.) and/or source of trainees (nuclear Navy, high school graduates, etc.) and phase of instruction (initial, transition, refresher, upgrade). Requirements for refresher training for routine reinforcement of skills and knowledges are undoubtedly among the most resistant to the application of ISD in the initial curricula development. That is because of the lack of specific incumbant job-holder feedback (and general lack of research) regarding the time course of forgetting of knowledges and extinction of infrequently used skills.

The production phase of training development includes the acquisition of the training devices and simulators and the writing of each lesson and subsequent production of the training aids (e.g., slides, video tapes, etc.), instructor manuals, student texts and hand-outs, and so forth. In parallel, the performance assessment instruments are prepared, based on the content of the behavioral and training objectives. These instruments include item pools for written pre- and post-tests; work sheets and scoring guides for simulator and laboratory lessons and plant drills; and guides for assessment of affective behavior (i.e., motivation, interpersonal interactions, ability to communicate, etc.).

A tool that is gaining rapid recognition in military training programs is a training management system, a computer-based system that provides wide-ranging logistical and quality control support to the managers of training. Potential capabilities of such systems are:

a) trainee record keeping;
b) diagnostic and prescriptive trainee evaluation;
c) resource management (availability of simulator time, etc.);
d) plant staffing demands (based on refueling schedules, etc.);
e) ISD data base management (impact of plant hardware changes, etc.);
f) item analysis of test items;
g) feedback for improvements to instructional modules;
h) operating system for simulator performance measurement system; and
i) operating system for computer-aided instruction.

With such a system in place, the final stages of the ISD process proceed much more efficiently. These stages are the evaluation and upgrading of the curricula and training materials and the processing of the trainees through the system.

Other requirements of the ISD process include establishment of training requirements and implementation of training programs for the ISD training analysts and managers, subject matter experts who provide the technical data and review of the ISD products, instructional staff who may be faced with new instruction techniques and equipment, and NRC staff who need to evaluate the quality of the ISD results.

CONSTRAINTS

Technical constraints are insignificant. The ISD process is well-defined, even though in some areas more reliance must be placed on expert judgment due to the lack of experimentally-based procedures.

Organizational constraints do exist, however. The adequacy of an ISD program heavily depends upon the team experience, team composition, and management support provided. The best experience for this type of effort comes from the major military ISD programs since those programs typically follow the process more closely than the "factory training" approach typical of most commercial training departments. The latter approach, which usually meets the established training requirements due to the generally high level of technical expertise and teaching experience of the staff, does not provide the comprehensiveness of the ISD approach, nor does it provide the opportunity to integrate all of the related areas of the nuclear power system. There are, in fact, very few individuals available who have had the exposure to a major ISD effort. Instructional technologists and educational psychologists are available, but in high demand, who are well versed in the ISD technology and who have practical experience on smaller systems. INPO seems to have such individuals who can effectively lead nuclear power training development programs.

The optimal mix of personnel on an ISD team includes instructional technologists and subject matter experts (SMEs). As pointed out above, well-trained instructional technologists are in demand, but the situation seems even worse for SMEs, especially qualified reactor operators. The available pool of such individuals is tapped by the utilities, NSSS vendors, training vendors, the NRC itself, and by other industries who can effectively attract these skilled workers into other endeavors in some geographic areas.

As alluded to earlier, utility management may not choose to invest in training program development in the absence of regulatory pressure. Even to meet the newer regulations, however, management may choose between committing only enough personnel resources to give sufficient appearance of complying and the much larger number of personnel to develop a high quality program and integrate it into the entire system.

NRC regulations covering training and licensing drive the construction of training programs, but since those regulation are not derived from an ISD process, or from any body of instructional technology literature, they typically cause training programs to fall short of professional standards. It is apparent that the training content of syllabi is selected not on the basis of the criticality/frequency/difficulty estimates associated with the task analyses and training objectives, but rather from the content of the licensing exams and requalification requirements.

In typical military ISD analyses, costs and resource availability are the most severe of the external drivers which affect the outcome of trade-off decisions. The nuclear power industry has unique characteristics which downplay those economic factors. In particular, any costs incurred by a utility to meet federal regulations can be passed on to their customers. Utilities buy or rent time on NRC-required sophisticated simulators which provide a very useful learning environment, but innovation and initiative are lacking to make the front-end investments in ISD-based curricula that in fact may lead to more efficient throughput of trainee assets and better retention of learned skills.

PRESENT STATUS

Current training requirements as set forth in NRC regulations, and guidelines do not call for the application of ISD to any personnel positions. Existing requirements are not related to objectively derived job criteria, but rather to expert judgments as to what learning experiences would make better qualified operators. Judgments such as these are a necessary part of the ISD process, but in the absence of job or criterion referencing, they do not satisfy the requirements for a system analytic approach. The NRC's Inspection and Enforcement personnel have developed courses used by their own and other NRC groups which follow the format of factory training courses. These courses are meant for a relatively small throughput of trainees and have the benefit of excellent training aids, but it must also be kept in mind that these courses are for the purpose of in-house (NRC) education, not to train proficient control room personnel.

NSSS and other vendors provide any training that the utilities wish to purchase. They have the capability to develop quality programs, and generally do so, but in the absence of regulatory pressures, do not go through the process of building a task analysis data base that can be used to integrate the training components of the personnel subsystem. These firms do develop training objectives, but these are generally directed toward training operators to pass the licensing and requalification exams. In non-regulated job categories (I&C, etc.), vendors create professional programs for training the skills that they have deemed necessary.

INPO has taken the lead (and, in fact, may be the only entrant) in ISD applications. The INPO Job/Task Analysis Model developed by T&E/TR&A is a comprehensive plan for "front-end" analysis, and is aimed at providing performance objectives and model programs to be tailored to individual utility needs. INPO's Training and Education Division also has in its statement of scope:

a) accredit utility training programs. (This will be along the same philosophy as that used for the accreditation of universities.)
b) develop methodology for instructor certification.

All in all, INPO is undertaking a program which is ambitious, but feasible. It is based on the principle that a training system can, and should be, self-regulating.

In addition to interim efforts to provide model programs to be adapted to each utility's own situation, they are currently engaged in a multi-year effort for DOE to carry out job and task analyses for all positions, from which training courses will be revised. The task analysis procedures are entirely adequate to produce a basic data base for their future efforts and should prove useful to the NRC in the NRC's efforts to establish or upgrade guidelines.

The INPO program to accredit industry training programs, including facilities, course content, teaching methods and equipment, instructor qualifications, and so forth, uses the basic concepts of peer review and self-improvement. There is a high likelihood that they will accomplish their goals of:

a) improving the general level of motivation by giving recognition to training programs that meet standards;

b) assuring that standards are met for both programs and staff on a continuing basis through annual reporting and periodic recertification; and

c) encouraging improvement via a heightened self-awareness required by the self-examination process.

It is encouraging that, as we understand it, the NRC is favorable to industry (through INPO) self-regulation with NRC verification.

The NRC is acting to upgrade its role in training, but to date has had a role which was little more than to conduct SRO examinations for instructors of certain courses at utilities, and to audit utility training programs to ensure that they are being conducted in accordance with the utility's application for licensing.

Secondary effects, of course, are precipitated whenever new licensing requirements are enacted. These include especially the motivation to structure training around the content of the examinations, and even the known examining inclinations of the examiners. These effects are counterproductive to good training practices, as is amply discussed elsewhere and fortunately recognized by many NRC personnel.

NRC plans for research activities are specific to particular areas of the nuclear power training system, and will, therefore, be discussed in their corresponding sections. Other activities, sponsored by DOE, INPO, and EPRI, are similarly deferred for later discussion. That is not to say that those topics (licensed personnel training, non-licensed personnel training, simulators) are independent of the ISD process, but rather that this is merely an arbitrary decision to limit the length of any particular section.

PLANNED ACTIVITIES

INPO's efforts are the primary programs that are addressing the general instructional development process. Specific task analysis efforts and simulator requirement efforts will be addressed in the next section. INPO plans to continue with further efforts as follows:

a) development of training guidelines;
b) job and task analysis program to provide training objectives, sequences of instruction, methods of instruction, and methods of evaluation for revisions to their model training programs;
c) workshops for Chief Executive Officers and universities;
d) development of specific courses;
e) scholarships;
f) participation in the regulatory process via comments; and
g) accreditation of training programs, facilities and equipment, and instructional staff.

These activities appear to be well directed toward assisting their member utilities in meeting existing and proposed regulatory requirements, as well as fostering professional practices.

MISSING ELEMENTS

Foremost among the missing elements is a single unifying force to ensure the coordination of all the efforts that directly or indirectly impact on training. Such efforts include:

a) task analysis program;
b) licensing examination development and validation;
c) simulator and trainer requirements;
d) licensing requirements; and

e) training program quality control (accreditation, upgrades, etc.).

It is our understanding that the NRC is favorable toward self-regulation with NRC verification, but the verification process will require well-experienced NRC personnel who are charged with and able to bring about the coordination of what are now separate activities both within and outside the NRC. Confirming the compatibility (i.e., cross-validation) of the separate data bases (from INPO, EPRI, NRC, utilities) being generated is alone a formidable job. Such an NRC group does not now exist.

To support the possible role of the NRC as a verifier of industry's self-regulation, a standard for the ISD process is needed but is not currently set forth in any NRC regulations or guidelines. Several standards exist, including the Florida State University model used by INPO.

Although the INPO effort has been discussed in very favorable terms in this section, one area to which INPO does not seem to have given sufficient emphasis is the application of modern training technology (e.g., computer-based instruction) and the functional specification of hands-on training equipment and their efficient incorporation into a syllabus. That is not to discount that the INPO Guidelines for Qualification Programs call for mixing of academics and hands-on training, but the manner in which their efficient integration should be brought about is not specified.

TECHNICAL FEASIBILITY AND PROBLEMS

In the preceding subsection, it was pointed out that a unifying body of individuals is needed to ensure that all of the activities that impact on training are coordinated. As time goes on, the coordination problems will get even worse: task analysis data bases will proliferate in differing formats with varying jargon and content; research tasks will go forward resulting in changes to licensing requirements and procedures which are already at odds with the process for the systematic development of instructional programs; and training practices will become even more engrained within each utility and more resistant to change. The difficulty in hiring well-trained instructional psychologists and technologists is a very serious one since such individuals are few in number and highly in demand.

The military experience provides sufficient reason to believe that the application of ISD to nuclear power industry personnel is both feasible and necessary. INPO has made a good start to bring this about. Several MIL SPECs for ISD are available, including the Florida State Model (adopted as the Tri-Services model and promulgated under separate titles by the Army and the Navy) and the more recently developed MIL-T-29053 produced by the Navy for aircraft crew and maintenance training programs, and adopted by some Air Force commands. Techniques for "computerizing" the data bases and for specifying the functional requirements for hands-on training equipment are available in the literature.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The development of training interacts with all other components of the system:

a) Licensing examinations should be the final criterion test for trainees and should be based on the same task analyses.
b) Personnel selection requirements determine the incoming skills, knowledges, and attitudes of trainees.
c) Equipment design determines which tasks are critical for training, which tasks are appropriate for job performance aids, and the content of each course of instruction.
d) Written documentation such as procedures and technical specifications become part of the training materials.
e) Availability of operational equipment and training simulators and devices constrain the selection of media.
f) Educational requirements interact with trainee population sources, training requirements, and career development requirements.
g) Quality of training is impacted by management philosophy, training personnel qualifications, and NRC regulatory emphases.

RECOMMENDATIONS

1. A point of contact should be established within the NRC to coordinate the training-related research and development efforts among the NRC groups with those of the utilities, INPO, and EPRI. This activity will also include ensuring the dissemination of training-related plant operating experience data from LERs and SALP (or 766 File), and from other observations of training-related deficiencies in plant personnel performance that should be routinely elicited from IE Resident Inspectors. Additionally, this activity should include the monitoring of the adequacy of training programs used for NRC personnel.

Importance: High

Urgency: Immediate

Implementation: Staff Educational Technologist

2. The NRC should publish a Regulatory Guide for Instructional System Development procedures which are suitable

for use by industry for the development of training for all plant personnel. For each plant personnel training program, the NRC should use that Reg Guide to evaluate the adequacy of the behavioral objectives and antecedent data. The NRC should monitor the procedures used by industry for the development of training to ensure that they are suitable for the development of comprehensive training programs and quality control.

Importance: High

Urgency: Immediately

Schedule: Issuance of Regulatory Guide - 1 year

Evaluation and Monitoring - continuous

Resources: Issuance of Regulatory Guide - 1 person-year; evaluation and monitoring - 0.1 to 0.5 person-year per plant position.

Implementation: Education Technologist; support from Subject Matter Experts. Schedule of availability of data from the utilities will constrain the evaluation and monitoring.

Dependency: Utilities will use as inputs the task analysis activiities of the NRC and INPO and the Model Training Programs under development by INPO. Coordination with the accreditation program of INPO will assist NRC efforts.

3. The NRC should sponsor research in retention of critical skills and knowledges for each plant job category. The results of this research should be used to develop a guide for determining refresher training requirements which will be implemented in the ISD Regulatory Guide.

Importance: High

Urgency: 3-5 years

Longevity: 3 years

Resources: 4 person-years per year; simulators or part-task trainers

Implementation: Career Human Factors Professional; support from training equipment programmers. Educational Technologist for Reg Guide update.

Dependency: Research designs will need inputs from ISD-based training programs and from the results of Evaluation Criteria research (Section 1.8). Results of this program will impact on the requalification requirements for licensed personnel (Section 4.2).

5.2 Licensed Personnel Training

REQUIREMENTS

When the activities of a job holder have an unusual potential for impacting the health and welfare of other individuals, government licensing of those job holders is common. Licensing requirements may include: initial training and experience before a license may be granted; a written, oral, and/or performance examination to qualify for the license; periodic in-service training; and peer, or licensing board review, in the event of malpractice. In some cases, malpractice can be a criminal offense.

To date, the only nuclear power personnel who are required to be licensed are Reactor Operators and Senior Reactor Operators, although many express the opinion that other positions within the plant should also be licensed. Licensing requirements include most of the elements named above. Entrants into the operator positions serve first as Auxiliary Operators, progressing through three levels of responsibility. During this time, they qualify by virtue of their experience, formal and on-the-job training, and other (sometimes irrelevant) personnel practices, to be selected for the RO training program. The utilities provide training via their own, or contracted, staff for hot or cold licensing. After this training/screening period, the trainee is certified to the NRC by the utility as being ready for the RO exam. A similar process takes place for the upgrade from RO to SRO.

Besides the RO and SRO positions, other jobs are affected since they require SRO licenses as a prerequisite (Shift Supervisor), or may draw partially upon licensed operators' assets (Shift Technical Advisor, instructors, examiners, and inspectors).

There are three fundamental, but intertwined, questions to be addressed in this discussion of licensed personnel training:

a) What training should these personnel receive prior to licensing?
b) How should trainees be evaluated in order to qualify for licensing?
c) What training should licensed personnel receive subsequent to licensing?

The process of Instructional System Development (ISD), discussed at length in a previous section, includes the means for answering substantial portions of these questions. That process especially provides a means for determining the training objectives based on the incoming trainees' skill and knowledge levels; the optimal syllabus, including course content, appropriate instructional strategies, and training aids and equipment; and the criteria for determining that trainees have met the training objectives. The ISD process also provides for quality control which, in this case, is a means for ensuring that the training objectives, course content, and evaluation criteria are valid with respect to the job requirements.

The answer to the second question should now also be apparent. The criterion tests developed via the ISD process are the most appropriate tests to be used to judge if a trainee is ready to do the job, for in fact, that is why they were developed in the first place. In practice, the licensing examinations are poorly related to the job requirements since they are developed independently of a job/task analysis or any other systematic process, and are not subjected to validation against job performance. As a consequence, training programs have been directed toward ensuring that trainees pass their licensing examination, rather than ensuring that all of the job-related skills, knowledges, and attitudes are learned.

The answer to the third question is much more difficult. The ISD process can tell us which tasks are not performed very often, and of those, which are critical or difficult to perform. Our state of knowledge, however, has not advanced to the point where a training technologist can predict how often and how much those tasks must be practiced in order that they can be adequately performed if ever required. Unfortunately, those tasks are the most likely to be related to emergency situations. Requalification requirements were established on the basis of expert opionion. To do any better requires lengthy research and much data to establish on a statistical basis the "forgetting curve" for each type of task.

CONSTRAINTS

There are several factors not under the control of training developers that influence the nature of the licensed personnel training. Foremost among these are the current licensing requirements, including the examinations used. The importance of ensuring that enough licensed operators are available to a utility causes the content of the training to be emphasized in the topics that match the examinations. Specific courses that are required by the regulations are also imposed on the syllabus. In each of these cases, there is no basis for objectively demonstrating that knowing the content of the examinations or the required courses is related to the specific job requirements. Even more so, many individuals have commented that requirements for college degrees will do nothing to ensure proficient operators. In fact, any such difficult requirement, especially ones that cannot be referenced to job requirements, can disrupt the production of operators by causing experienced personnel to drop out of the upgrade program. Such individuals may be perfectly capable of doing the advanced job, but may not be motivated to do so much more book work, or may find that particular level of schooling too difficult and stressful.

Another influencing factor in any instructional system is the source of trainees. The nuclear Navy has been the prime source of well-qualified entrants into the commercial nuclear power industry. Although the control room assignments in the two environments are different, the ex-Navy personnel have a wealth of knowledge that easily transfers with them. Other entrants into the auxiliary operator program come essentially off the street with no more than a high school level education, or from the non-operator plant positions via labor union seniority agreements. This range of experience, capability, and selection processes poses a tremendous challenge for the training developers.

PRESENT STATUS

Training programs for initial licensing, requalification, and upgrade are well established by the utilities through in-house and/or contracted training vendors. The NRC takes a strong hand in the administration of the licensing examinations, but relies on utility certification to confirm that the trainee has completed the requisite instruction and experience. The NRC role in requalification is to audit the training program to verify that it is being conducted according to the training plan prepared by the utility. NUREG/CR-1750 summarizes the NRC role as follows:

> . . . The current practice of exercising split responsibility between OLB and IE for operator training casts doubt on the effectiveness of this practice. No single organization within the NRC is fully responsible for licensed operator initial or requalification training. OLB is responsible for utility and training center program approval, training center program audits, and audits of utility requalification examinations. IE is responsible for cold program audits and periodic audits of requalification programs (less the annual requalification examination). No organization is responsible for auditing licensed operator replacement (hot) programs. . . . (p. 2-87)

Most of the other conclusions that those authors derived may be characterized as particular results of the lack of a system approach, or ISD, methodology in the establishment of training programs and the evaluation criteria thereof.

In fairness to the theoretically-inclined purists, it should be pointed out that there is a contradiction in being simultaneously concerned about the diffuseness and gaps in the

NRC's responsibilities, and the lack of an ISD-based training/evaluation system. If the principles of ISD were rigorously followed, the only concern that the NRC would ever have is when a trainee (initial, upgrade, or requalification) does not demonstrate mastery of an objective during an examination. Even then, the feedback mechanisms that would have been built into the overall system would correct the causes, be they in the examinee, the syllabus, or the evaluation criteria of the examination itself.

Current attempts by the NRC to upgrade training are primarily to raise the passing grades on examinations or to require that certain courses or topics be added to the curriculum. Although adequate for a short-term solution to obvious deficiencies, neither solution can be justified from objective job-referenced criteria.

The NRC has achieved a major step with the analysis reported in NUREG/CR-1750. With the exception of the "performance predictive indices," which suffer from methodological difficulties, the conclusions and recommendations are ones with which we generally agree. The same is true of NUREG/CR-1482 ("Nuclear Power Plant Simulators, Their Use in Operator Training, and Requalification"), which also supports the call for an integrated approach to training as an incidental observation along with its primary purpose of presenting a technique for the rational development of simulator scenarios.

Industry efforts are also underway to put operator training on a sc nd basis. INPO's program to carry out job/task analyses and to develop model training programs are encouraging starts. As the job/task analyses are completed (licensed operator analyses are currently underway), the training programs will be updated. INPO has adopted the Florida State University model of ISD for the job/task analysis, which is one of the more comprehensive procedures available. EPRI's most notable contribution to date in the rationalization of training is the EPRI NP-783 Report, "Performance Measurement System for Training Simulators" (PMS). While the PMS that they demonstrated is meant for full-scale simulators, such a system is expandable to aid in the entire spectrum of training and testing and in the instructional development process.

PLANNED ACTIVITIES

INPO will carry out the job/task analysis for non-licensed personnel in FY82 and continue to update their guidelines for personnel training. Although there is no apparent unified movement, it may be expected that an increasing number of utilities and training centers will acquire training equipment, other than full-scale simulators, for licensed operator training. This, of course, does not guarantee that they will be optimally used, but in the hands of the professional training development personnel of the vendors and some utilities such equipment will most certainly be an asset.

The NRC has given a high priority to satisfying the need for a Reactor Operator Task Analysis which is the title of the RES task formulated in response to H. R. Denton's statement of NRR needs. A contractor will be chosen to carry out the effort during 1982, but the content and format of that task analysis will be determined during the proposal and negotiation process. The Statement of Work, however, appears to be slanted more toward a human engineering analysis than a training analysis. This also appears to be the case in another program, scheduled for completion in FY82, being conducted for ORNL. The Statement of Work for that program, entitled "Task Analysis for Safety-Related Operator Actions," lists the minimum task elements to be used, including many that are useful to a training analyst, but that program, too, is directed primarily at human engineering problems. Training is of interest, but is not a central theme.

Another RES proposed task is directed toward the improvement of "Operator Examinations." That effort, "pending the availability of qualified personnel," would study present and alternative examination techniques vis-a-vis on-the-job performance.

Two other proposed tasks considered by RES to be of about equal priority are the "Validation of Current Educational and Training Requirements for Operators and Senior Operators," and a similar, but less experimentally-oriented study to develop "Education and Training Requirements for Shift Supervisors, Shift Technical Advisors, and Upgrading of Senior Operators." Both of these studies would make use of the pending task analysis results and recognize the long-term data gathering in the operational environments that is required to validate licensing requirements.

The LQB Safety Technology Program Plan elaborates on the goals and background for the NRC research tasks which combine TAP items into coordinated units. The content of the research goals are consistent with those that have been supported in these discussions.

MISSING ELEMENTS

It is evident from the NRC research plans that the NRC is sensitive to the major issues confronting the licensed operator training. What is missing is the unification of the individual problems into a unitary process -- Instructional Systems Development (ISD). It was pointed out in the section on ISD that the tasks which are the subject of the projected NRC research are part of interconnecting data generation and trade-off events. Separating these processes risks basing decisions on incomplete databases, inconsistent criteria, or design goals.

Similarly, there seem to be resources within the NRC which are not integrated fully. In particular, the IE personnel have capabilities for imparting insight into the evaluation of operators and utilities that go beyond their current responsibilities. Expanded roles have been suggested relating to the administration of examinations; however, the day-to-day observations made by resident inspectors seem to make them ideally suited to evaluating the readiness of operators to take examinations, to constructing the plant specific portion of the written examinations, and in fact, to administer all parts of the required examinations. Since these NRC personnel already are required to stay apart socially from plant personnel, they would not be in any more conflict of interest with such an expanded role. Blending these potential resources into OLB functions does not seem to be receiving attention. Although it has already been suggested (NUREG/CR-1750) that instructors be given a role in the conduct of examinations, the integration of all resources, IE, instructors, and perhaps others, into the training process is not specifically identified as a research need.

Many specific issues may be addressed by the proposed research efforts, but one cannot determine on the basis of extant documentation if they will be, or if they will not be, fed back to all potential users. For example, will the unusual training requirements for the SPDS (which is not used on a day-to-day basis, but may be critically important when needed) be recognized by the ISD personnel from data supplied by those responsible for the development of the SPDS requirements? Will management training for senior control room personnel and shift supervisors include training in the techniques for subjectively evaluating their subordinates? Will the concept of a "passing score" on an examination be scrutinized for validity considering that "need to know" skills and knowledges require a perfect score (except when it is established that the trainee will have sufficient opportunity to "master" the remaining material when on the job)?

TECHNICAL FEASIBILITY AND PROBLEMS

The problems are those of any ISD process:

a) finding enough individuals with the proper education and experience to conduct or evaluate the efforts; and
b) developing a basis for evaluating the short-term adequacy of the efforts. Long-term evaluation is a self-adjusting process via the feedback loop built into the ISD concept.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The inputs to the development of a training system for licensed personnel are most closely tied to the licensing qualifications and requirements. Training (or education) at any level is most commonly measured at the time for graduation, viz., does the trainee pass the examinations. Another input, but one which is usually of lesser visibility, is the sources of personnel, i.e., the personnel selection process.

The initial design and modifications of the plant and control room have an obvious impact on the content of training, but so does the equipment available for training. The adequacy of individual or team practice may depend on the fidelity of the mathematical model that runs the simulator.

RECOMMENDATIONS

Section 5.1 recommends the issuance of a Regulatory Guide for ISD procedures and the responsibility for evaluating and monitoring training program development and quality control for all plant positions including licensed personnel. An additional recommendation follows.

The NRC should adopt the recommendations of NUREG/CR-1750, Section 2.10, License Training Instructors, which are summarized (NUREG/CR-1750, pp. 6-16, 6-17) as:

1. Before any instructional assignments, all training personnel (including Training Managers) should attend a certified course or program specifically aimed at the familiarization with and application of instructional methods and techniques.

2. During periodic audits, ensure that instructional staffs have received training or possess the equivalent education necessary to demonstrate effective training practices.

3. Utilities implement periodic workshops or retraining programs for assessing and improving instructional skills. (This is not an NRC activity.)

4. In evaluating instructors, utilities should consider several measures, including: (a) meeting of well-stated, valid objectives; (b) periodic observation by an instructional specialist; (c) trainee feedback; (d) trainee performance on the job (supervisor feedback); and (e) Training Coordinator or Senior Instructor observation using a detailed, structured observation list.

Importance/Urgency: Immediate staff action

5.3 Non-Licensed Personnel Training

REQUIREMENT

NUREG/CR-1750 presents data from a job task analysis to support the view that each of the following plant personnel positions has some job requirements that are safety related:

a) Radiation Protection Technician
b) Engineers and Technical Support Personnel
c) Maintenance Personnel
d) Chemistry Technicians
e) Instrumentation and Control Technicians
f) Quality Assurance and Quality Control Inspectors
g) Auxiliary Operators
h) Shift Technical Advisor
i) Managers
j) Independent Review Personnel

Requirements are in place for utilities to provide suitable training programs for all such personnel. For the purposes of this report, we would add utility or vendor instructors and the NRC's inspectors and examiners, all of whom have obvious safety-related functions.

Currently, training practices run the gamut from the most formalized training programs for auxiliary operators, and many other curricula run by utilities and training vendors, to nothing more than on-the-job training (OJT). Unlike that for the licensed positions, the only NRC responsibility for the unlicensed plant position training is to audit to ensure that the programs are being run in accordance with the utility's approved plan. The utilities are responsible for the conduct and quality control of their programs. As a special case, the training program of the American Society for Nondestructive Testing is used for inspection and testing personnel.

Instructors of licensed personnel are generally drawn from licensed operator assets of the utilities. If not already licensed, simulator instructors are now required to pass the SRO examination. The experience level of instructors for nonlicensed personnel is considerably more heterogeneous. In most cases, instructors are not instructed themselves in the techniques of teaching; however, the better training programs audit the quality of instruction and attempt to employ "good" teachers. Training centers run by utilities and vendors tend to be well-supplied with training aids, classrooms, laboratories, and other work areas (not to mention simulators for the operator training programs) which can provide the proper learning situations.

The NRC IE personnel run an abbreviated training program for their personnel and for other NRC personnel who request it.

Their program is designed to give enough background and technical information to satisfy the needs of the IE regional offices.

In addition to full-time examiners, OLB part-time examiners are drawn from partially experienced personnel (e.g., from university research reactors), but are required to attend in-service training. Many comments have been made that the trainees often know more than the examiners.

It is apparent that the major requirement is to adequately define training objectives for all of the positions considered in this section, preferably using the well-established techniques of Instructional System Development (ISD) as more fully discussed in a previous section. From that product, the NRC can make rational decisions as to the need for other plant positions to be licensed (many believe that I&C technicians need such consideration), and appropriate training curricula can be designed. The problem of accreditation and quality control of the training becomes a decision between self-regulation by an industry organization, such as INPO, or a new responsibility for the NRC.

CONSTRAINTS

Qualitatively, the constraints on the development of nonlicensed personnel training are similar to those for licensed personnel training; however, the existing regulations for the former are insignificant in comparison.

The availability of personnel to conduct the required ISD analysis is always a problem due to the relatively small number of experienced individuals with the appropriate training.

PRESENT STATUS

NUREG/CR-1750 has provided an excellent review of the nonlicensed personnel training area and presents some recommendations that are consistent with the requirement stated above.

INPO has completed model training programs in the form of "Guidelines for Qualification Programs" for nonlicensed operators (Class A, B, and C Auxiliary Operators). This work will be subjected to updating as part of the INPO ongoing program.

No other completed or current work is known to us. The present status reflects the low priority that has been given to this aspect of the nuclear power system.

PLANNED ACTIVITIES

Two specific programs proposed by the NRC in the RES response to NRR needs are directed toward the analysis of tasks for nonlicensed personnel. One is "Task Analysis for Operational Support Activities"; the second is simply entitled, "Plant Maintenance." Both of these low-priority tasks are primarily aimed at the improvement of procedures and requirements, but each includes improved training requirements as a goal.

INPO is planning to extend its job/task analysis effort to all plant positions and to develop model training programs for use by the member utilities. The accreditation program that INPO is beginning will provide a quality control function over all utility training which is modeled after accreditation of higher-education institutions. INPO's efforts will be coordinated with the NRC need to ensure that adequate training is available to all utility personnel.

MISSING ELEMENTS

It appears that all of the mechanisms are in place through the efforts of INPO to guide the development of training programs and quality control for non-licensed utility personnel. It is not clear that further improvement in the training of IE personnel is required, since that training apparently is meeting their needs. That is not to say that structured training objectives are not needed. If the quality of the IE training is due to a few key individuals, then the application of a streamlined ISD process (the trainee throughput may be too small for the full process) may ensure continued adequacy in the future. If the role of IE is expanded to include license examination functions, then certainly a formal training analysis will be required for new job requirements. The problems attendant to the training of licensed operator examiners is in need of such an analysis, but more importantly, the manner in which the examination process is conducted (examination construction, criteria development, administration practices, and so forth) needs to be investigated with a view toward introducing modern testing technology and practice.

TECHNICAL FEASIBILITY AND PROBLEMS

The problems are those of any ISD process:

a) finding enough individuals with the proper education and experience to conduct or evaluate the efforts; and
b) developing a basis for evaluating the short-term adequacy of the efforts. Long-term evaluation is a self-adjusting process via the feedback loops built into the ISD concept.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

A prime driver of the content of training programs for nonlicensed utility personnel is the plant design and operating procedures. Personnel selection requirements are always a factor in scoping the training requirements. In this area, one finds a more heterogeneous source of trainees than for licensed operators, which emphasizes an interesting interaction. It is sometimes the case that personnel practices and union seniority rules require that non-operator personnel (e.g. technicians) be allowed to enter the licensed operator pipeline via the auxiliary operator program. The good feature is that such employees have already had exposure to general background and indoctrination training; the bad news is that these personnel are not initially selected on the basis of operator selection criteria.

The training requirements for OLB examiners is dependent, too, on the selection requirements for examiners and the allocation of examination functions to those personnel.

RECOMMENDATIONS

Section 5.2 recommends that the NRC adopt the recommendations of NUREG/CR-1750, Section 2.10, Licensee Training Instructors. It is recommended that the same items (summarized in Section 5.2 Recommendations) be adopted for Non-License Training Instructors.

Importance/Urgency: Immediate staff action.

5.4 Training Equipment

REQUIREMENT

Full-scale control room simulators are required for training and for use in examinations in current and proposed licensing requirements for control room operators. This use has been accompanied by a growing acceptance of training equipment of various kinds (other than simulators) into training programs. The acquisition and use of full-scale simulators has followed a course which parallels the introduction of full-mission simulators into military and civilian aircrew training. Namely, such devices are designed to serve the sole goal of replicating the operational equipment in every way; programs of instruction are designed around the simulator capabilities. The changing philosophy in the aircrew setting will be discussed shortly.

The value of simulators in nuclear power operations training is well-known and also parallels the military needs:

a) They can be acquired before the operational equipment is ready;
b) They are less expensive to repair if damaged during practice;
c) Emergency procedures can be safely practiced;

d) Instructional features are available that allow more useful training events per unit of training time than can be achieved with operational equipment;
e) Situations can be experienced which might never occur during the operation of real equipment; and
f) Plant operations are not disrupted by the presence of trainees.

As a matter of general interest, it should be noted that the manufacturers of nuclear power plant simulators also have ample experience in producing training simulators for military and civilian aircrews. Hence, the same high quality products are produced.

Full-scale simulators are not the only training equipment to be considered, however. The following provides a spectrum of the possibilities:

a) pictorials -- two-dimensional photos or diagrams -- can be used for familiarization and mental rehearsal
b) scale models and "cold" mock-ups -- scale or full-sized replicas of equipment; may have movable parts, but no response to actions -- can be used for familiarization and mental rehearsal
c) computer-aided instruction -- computer controlled graphics representation of equipment supported by mathematical models; graphics change in response to trainee inputs -- can be used for concept development, systems knowledge, and procedures training
d) "hot" mock-ups -- full-sized replicas of equipment; partially interactive (low-fidelity) with trainee's actions -- can be used for procedures training
e) systems trainers -- two or three dimensional replicas that also present status information for system components not included in operational equipment displays -- can be used for concept development and systems knowledge
f) part-task trainers -- fully interactive, high fidelity replicas of a subset of systems -- can be used for procedures training and decision-making training
g) full-scale simulators -- fully interactive, high fidelity replicas of all systems -- can be used for procedures training, decision-making training, and team training.

It is important to note that these generic devices are as applicable to maintenance and technician training as they are to operator training. The manner in which one decides which to use is a special topic within the Instructional System Development (ISD) process. This brings the discussion back to the changing philosophy in aircrew training equipment acquisition.

For the last several years, there has been a growing trend for training equipment to be specified (at least functionally) on the basis of behavioral data. The best known early research was initiated by the Air Force Human Resources Laboratory (at Wright-Patterson AFB) in which the criticality, difficulty, and frequency of each task to be taught with hands-on equipment were examined. Priorities were then established for inclusion of the necessary system representation in the trainer, and were traded off with the cost of their inclusion. The result was a trainer that had a high training utility at the lowest cost. No unnecessary functions were included.

This process has been extended to major military ISD programs in which task analysis data and the resulting training objectives were used not only to provide functional specifications for full-scale simulators, but also entire suites of training devices for specific familiarization, procedures training, and part-task training. These devices can be specified in conjunction with the development of syllabi for each trainee curriculum so that the entire sequence of training is a mix of classroom and tailored hands-on training. The question may be asked, "Why bother?" Part of the answer is in terms of resource management; part is based in instructional theory.

If only one type of operator training equipment is available, viz., full-scale simulators, one readily finds that it is difficult to optimally schedule trainees so that the trainee's available time at the simulator is efficiently spent. This is especially true early in training when the operations of single systems is being taught. Some simulators are designed in such a way that the individual panels cannot be exercised independently and, hence, even if trainees are at different panels, any actions that they may take must still be coordinated with the other trainees. For team training, this is fine, but for concept development, systems knowledge, and some procedures training, it is disruptive.

The process of syllabus development is to build skills and knowledges on top of the previous learning experiences. The learning environment for each teaching event should maximize the efficiency of acquisition and the stability of retention. While not nearly enough is known about skill retention over long periods of time, several principles of efficient learning acquisition are well-established. Among these are the need for active movement in skill acquisition, maintenance of an optimal level of arousal, spacing practice over time rather than massing all practice at one time, building on prior knowledge, order effects (first and last items are learned first), and the positive and negative transfer effects that derive from the nature of the tasks being learned.

Training equipment that is tailored to the needed learning environments can shorten the total training time required via efficiencies accrued, which can free more time for a trainee to practice more problem solving separately on part-task trainers, or as part of a team on the full-scale simulators. It is obvious that the more exposure an operator trainee has to combinations of malfunctions and other emergency conditions in a training situation, the more likely an operator will have the mental flexibility to solve new problems. Much of this type of practice can be achieved on equipment which is much less expensive and less demanding of facilities than the simulators. If one had one or more trainers available, they could be located in a training center, or even in the plant itself, with minimal or no need for instructor intervention. This, again, could increase the efficient use of a trainee's available time and also make training equipment available for more frequent refresher training. Parallel arguments justify a similar desirability for a range of training equipment for maintenance and technician personnel. Their needs, however, can usually be satisfied with a smaller number of different trainers and a greater emphasis on the use of actual equipment.

To one extent or another, the following points can be made for any of the spectrum of training equipment listed near the beginning of this section:

a) They can be integrated into formal training programs to provide better (i.e., faster and/or more complete) training;
b) They can be located in-plant for self-paced refresher or update training;
c) They can be made plant-specific and available long before the full-scale plant-specific simulator is available; and
d) They could be used for licensing and requalification requirements in those areas where they are valid.

The combination of the last three points might provide an alleviation of the problem reported in NUREG/CR-1482, that requalification training on a generic (non-plant-specific) simulator can be ineffective for some trainees whose attitudes make them critical of even small deviations from expectations. It may, or may not, be the case, however, that the trainees are right! If the simulator is causing the trainee to perform a response that is in conflict with the actual equipment procedure, that would be a classic case of negative transfer which should be rejected by the trainee. If the trainee's series of actions is the same as on the actual equipment, and the simulator is displaying dissimilar, but not incompatible cues, then we are faced not with an inappropriate training environment, but with a difficult-to-solve motivation problem.

The section that discussed the general application of the ISD process to nuclear power training programs made a point of the need for a performance measurement system (PMS) which, among other functions, would be a repository of trainee performance on hands-on practice and classroom performance. Such a data base would be useful to the instructional staff to ensure that the content of a simulator practice session is matched with the objectively derived needs of the trainee. One implication is that the scoring system for simulator lessons is derived from the same training objectives upon which the rest of the training system is based. It is particularly attractive to be able to trace poor performance during a hands-on session to the prerequisite hands-on and classroom lessons. It is then possible to identify remediation for a poorly learned lesson segment or perhaps identify a need to strengthen the course content if a common problem is encountered. Especially if a PMS existed to ensure objectivity, it becomes increasingly feasible to utilize instructors as examiners for the licensing process, as has been recommended in NUREG/CR-1750.

Subsequent to the TMI accident, it became apparent that if simulators were to meet their requirement in operator training and licensing, the models that drive them must be made capable of reproducing the full sequence of events in multiple failures and emergencies. Requirements came forth for licensees to take part in simulated malfunctions for requalification training, which created the further need to provide a sound approach to determining which malfunctions should be included. NUREG/CR-1482 provides an approach based on available data which is adequate for an interim solution to ensuring a good choice of license experience. For the long term the ISD process can provide the best process for determining the best combinations of simulator experiences for training and testing, the required fidelity of the systems to be represented (and, hence, the particular training equipment required), and the instructional/PMS features needed. The more comprehensive ISD approach can then provide a basis for establishing instructor and examiner training requirements.

There is one other process which has gained attention recently in the acquisition and use of training equipment. Simulator certification (SIMCERT) or training effectiveness analysis (TEA) is now being required by military communities, following the FAA lead for commercial aircrew programs for new equipment added to their inventory. The SIMCERT/TEA process of validating that the training equipment provides adequate training to meet specific training objectives is equally applicable to operator, maintenance, or technician training devices. It goes beyond determining that the device was built to specifications, and requires proof in the form of transfer-of-training tests that the training device can be substituted for actual equipment to meet some particular learning objective. The objective may be any subset of skills required for the terminal objective (i.e., the job requirements). This process has been used by the FAA to allow them to certify commercial airline simulators for use as substitutes for particular airborne training phases. The outcome of the SIMCERT/TEA process, then, is a specification for what tasks adevice can be used to train, and by implication, how it is to be used. The obvious corollary is that it also determines those tasks that can be adequately tested on the equipment.

It is expected that there will be an increasing demand for the use of full-scale simulators, and perhaps other equipment such as concept development and systems trainers, to aid control room personnel in handling emergency situations and in the evaluation of plant and control room modifications. These are traditional roles of engineering simulators. The engineering simulator, however, is much more than a training simulator. The engineering simulator must have more flexibility for initialization conditions, parameter outputs, mathematical model comprehensiveness, hardware modifications, and scenario generation. An engineering simulator, on the other hand, is only a good training simulator if the appropriate instructional features are included.

To summarize the requirement, the process of ISD should be used to determine the mix of training equipment and their place in the syllabus for all plant positions (and for other personnel who receive training, such as instructors and NRC inspectors and examiners). The same process should be used to select the practice scenarios and examination situations. As part of the validation process, a SIMCERT/TEA methodology should be used to ensure that the training (and testing) objectives can be met with the hardware/software/instructonal features included in the equipment design.

CONSTRAINTS

Technically, there are no bounds to what can be designed into a training device. New technologies such as synthetic speech open up exciting possibilities for team training without the rest of the team. VHSIC (Very High Speed Integrated Circuits) programs in the military will make real time execution of the most complex models "old hat." Microprocessors, video disk, and the like already have vendors beating on the doors of the utilities to provide low-cost training devices. Computer-aided instruction (CAI) systems, such as PLATO, TICCIT, or others, make CRT-displayed simulations an ideal tool for process control, troubleshooting, and concept development training.

Reluctance of some utilities to invest in these aids is the only real constraint. Financial constraints of some smaller utilities to acquire their own simulators are overcome by renting time on simulators owned by other utilities or training vendors.

PRESENT STATUS

Operator hot and cold license requirements and requalification requirements require and/or allow the use of simulators for the substitution of performance on actual control room equipment. Generic (non-plant-specific simulators) are allowed where plant-specific simulators are not available.

Requirements for specific activities such as start-ups, shut-downs, and responses to malfunctions, have been established on the basis of equipment capability/availability and expert opinion regarding potential emergency situations. There has been no attempt to blend these factors into comprehensive training-testing programs. There has been no NRC attention given at all to the training equipment needs of non-operator personnel. This latter situation has given rise to a corresponding lack of attention by some utilities to the training program needs of their non-operator personnel. Pressure by the NRC may be the only way to achieve high quality training for all personnel at all utilities.

Many utilities or their training vendors do an impressive job of providing scale models, laboratories, and other work areas for technician and maintenance trainees. Others rely heavily on on-the-job training. A large number of plants will have their own simulators and some are introducing "basic principles" (systems) trainers.

The Canadian CANDU training simulator is also an engineering simulator, of sorts, in that a multitude of parameters and all operator actions are output for later analysis. That arrangement, however, does not take the place of a PMS which assists the instructors, rather than burden them with too many choices. The approach toward the development of a PMS reported in the EPRI NP-783 report is commendable. That report very astutely points out that the role of the instructor is also very important. Many subjective factors such as attitude, motivation, confidence, communication skills, and leadership skills, can only be assessed by an experienced human observer.

A notable NRC study, NUREG/CR-1482, provides useful data and a suitable short-term approach to the selection of malfunctions that should be available for simulation for training scenarios. That report also emphasizes the need for a "system approach to training," another name for ISD.

PLANNED ACTIVITIES

The RES response to the NRR research needs lists one program that impacts on training equipment and another which has a secondary impact. The first is "Capability of Training Simulators" (Tasks A1 and A2 of RR-NRR-81-5). Work is proposed to continue the effort in NUREG/CR-1482 to define the emergency tasks that need to be simulated in order to provide adequate training. These decisions will be based on accident sequences from documented risk analyses. In another effort of that program, data will be gathered on simulators from other technologies for comparison of their capabilities, uses, fidelity, design, and procurement practices. Both of these efforts should prove useful, if included in a broader range program to integrate trainers and simulators into training programs. The second RES program is "Research Dependent on Advanced Simulators" (Task B of RR-NRR-81-5). Its importance lies in the potential of using an advanced engineering simulator to validate the design of full-scale training simulators and part-task trainers. The availability of such a device for that use is unprecedented in the simulator community. It would be nice, but could not be justified as necessary. Our survey did indicate that no other activity in industry is planned, with the notable exception that training device vendors are actively pursuing the marketing of procedures, systems, and other non-full-scale equipment to the utilities.

MISSING ELEMENTS

The design and incorporation of training equipment into the training programs for nuclear power personnel has made little use of modern technology except in the simulators designed for control room operators. There is also no program for the validation of training (or testing) carried out on training equipment, i.e., SIMCERT or TEA studies. Although adequate training can be carried out using a wide variety of methods and media, it should be kept in mind that the goal is to provide the most efficient training program to ensure that a trainee has acquired the necessary enabling and terminal skills required by the licensing examinations, and to provide that training using techniques that maximize skill retention.

TECHNICAL FEASIBILITY AND PROBLEMS

The military training community has advanced the technology of training for the operation and maintenance of large systems to the point where there is no impediment to the transfer of that technology to the nuclear power industry.

INTERACTION WITH OTHER SYSTEMS

Presently, training equipment is treated as an entirely separate entity; however, in an ISD-based nuclear power training system, training equipment interacts with the following:

a) Modifications and updates to the operational equipment must be reflected in the training equipment. This, in some cases, is even best done before the actual equipment changes, so that trainees are saved the trouble (and negative transfer) of learning the old configuration first.

b) The mathematical models needed to drive any of the high fidelity, interactive training equipment could be based on plant engineering development models. Conversely, the availability of comprehensive simulator models could provide a resource to evaluate engineering changes.

c) Simulators are already being incorporated into the licensing examination procedures. Other forms of training equipment may also be determined to have a valid place as testing equipment.

d) The existence of training equipment impacts on the design and efficiency of all training programs in which they are used (operators, technicians, maintenance and other support personnel, and IE and other NRC personnel).

RECOMMENDATIONS

The NRC should publish a Regulatory Guide for the certification of the training effectiveness of training simulators and other devices upon which terminal training objectives will be met.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 1 year

Resources: 2 man-years

Implementation: Career Human Factors Professional; support from Educational Technologist and Subject Matter Expert.

Dependencies: None

6.0 INCIDENT RESPONSE PLAN AND NRC FACILITIES

One of the specific results from investigations of the accident at Three Mile Island Unit 2 was a recommendation that the NRC improve its capability for response to nuclear emergencies. Several related efforts are presently under way to address this recommendation. For the sake of simplicity and clarity, these efforts have been divided into four areas for discussing human factors concerns. These areas are:

1. The incident response plan and NRC facilities,
2. The NRC Operations Center,
3. Utility emergency response facilities, and
4. The Safety Parameter Display System.

6.1 Incident Response Plan

REQUIREMENT

Human factors issues within the broad area of emergency preparedness planning are beyond the scope of the present contract and thus no specific concerns were identified. Obviously, however, the planning for and implementation of major emergency response actions has a great deal to do with human factors and other people-related issues.

PLANNED ACTIVITIES

The most significant planned activities by the NRC with respect to emergency response plans and preparedness will be to run incident response exercises. These exercises will simulate various events of national concern and presumably will be used to provide feedback in subsequent modification to plans and facilities as necessary. Similarly, utilities are conducting exercises simulating emergency response activities as a result of incidents in the vicinity of the nuclear station.

MISSING ELEMENTS

Nothing applicable at this time.

TECHNICAL FEASIBILITY AND PROBLEMS

There do not appear to be any technical feasibility or other problems which would preclude resolution of any of the human factors concerns.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The entire network of personnel, facilities, communications, procedures, etc., involved in planning and response to incidents represents extensive interaction between all components. Those components considered in this section specifically interact with the other components to be discussed in subsequent sections that deal with utility emergency response facilities and the safety parameter display system.

CONSTRAINTS

There do not appear to be any technical implementation constraints. The major constraints would appear to be in the political-social area because of the significant interaction and cooperation required among other federal, state, and local organizations.

PRESENT STATUS

The Atomic Energy Act, as amended, and the President's statement of December 7, 1979, with the accompanying fact sheet require the NRC and FEMA to provide guidance and acceptance criteria to NRC licensees, state, and local governments to develop radiological emergency plans and improved Emergency Preparedness. The NRC responded by issuing NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The purpose of this document is to provide a common reference and guidance source for:

1. State and local governments and nuclear facility operators in the development of radiological emergency response plans and preparedness in support of nuclear power plants.

2. Federal Emergency Management Agency (FEMA), Nuclear Regulatory Commission (NRC), and other Federal agency personnel engaged in the review of state and local government, and licensee plans in preparedness.

3. The Federal Emergency Management Agency, the Nuclear Regulatory Commission, and other Federal agencies in the development of the National Radiological Emergency Preparedness Plan.

Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Plants," reissued in 1981, reflect recent pertinent amendments to 10 CFR Part 50, and endorses NUREG-0654. All licensed utilities are currently preparing incident response plans in accordance with NUREG-0654.

RECOMMENDATIONS

The focus for human factors concerns during this project was the nuclear power plant control room. The Incident Response Plan goes well beyond the control room, but its provisions were briefly reviewed because of their significance. The primary human factors concerns that were identified relate to the utility emergency response facilities and the safety parameter display system which are discussed in sections to follow.

A systems analysis should be done to identify more precisely the behavioral and human factors issues related to planning for response to emergencies. There will be many people involved if an emergency response plan is implemented, and the preparedness of these people is a significant human factors concern.

Importance: Medium

Schedule: Urgency - 1-2 years

Duration - 3 years

Resources: 3 person-years

Implementation: Requires a career human factors professional and a social scientist experienced in emergency planning and behavior.

Dependencies: None

6.2 NRC Headquarters Operations Center and Regional Facilities

REQUIREMENT

The NRC Operations Center for incident response and support was visited on two occasions and discussions were held with personnel responsible for the Center design and operations. In addition, some documentation on the Center design and operations has been reviewed. Based on this preliminary information, the following potential human factors issues have been identified, although it must be recognized that the list is neither complete nor necessarily totally valid:

1. The mission of the Headquarters Operations Center and consequently, the responsibilities of its members do not appear to be firmly established.

2. The decision-making functions have not been thoroughly identified and therefore the information and communication requirements of the various organizations and individuals cannot be readily assessed.

3. The present facility seems to be highly dependent upon telephone communications, which raises concerns over the timeliness and accuracy of information transmitted.

4. Storage and display of information does not seem to be based on any systems or task analysis. Consequently, techniques such as common or shared displays, or the ability for individuals to obtain scientific information rapidly and accurately may not be effectively employed.

It should be strongly emphasized that the present facilities and equipment of the NRC Headquarters Operations Center does not reflect the current state of planning or thinking that has been accomplished by the personnel responsible for the design and operations of the Center. A complete conceptual design has been prepared; however, there have been no long-range commitments for implementation of this design. Consequently, it is recognized that the types of human factors issues identified above may well be reduced or eliminated if resources are committed to implement the longer-range system concept.

NRC Region Office incident response centers will play a minimal role compared to the NRC Headquarters Operations Center and the utility operations personnel and facilities. The Incident Response Center at Region I was observed during a one-day visit to their headquarters. Thus, the comments in this report are based almost exclusively on the information obtained from Region 1 personnel and documentation. In general, the region will be active in incident management only until local on-site personnel can assume the responsibility. Their mission seems rather clear and the facilities, personnel, procedures, and other resources necessary to carry out that mission seem adequate from a human factors point of view. However, it is assumed that problems of coordination, communication, teamwork, data recording and retrieval, and others that might be discovered during training exercises would be noted and corrected.

CONSTRAINTS

There do not appear to be any major technical constraints preventing implementation of the NRC Emergency Response Facilities. The major constraints appear to be budgetary and the recognition of a higher priority for upgrading the NRC Incident Response facilities.

PRESENT STATUS

Public Law 96-295 contains a request for the NRC to provide three reports to Congress, all related to improvements in the NRC response to nuclear emergencies since the accident at Three Mile Island Unit 2 on March 28, 1979. The reports prepared to answer that request are:

- o NUREG-0728, "Report to Congress: NRC Incident Response Plan"
- o NUREG-0729, "Report to Congress on the NRC Emergency Communications"
- o NUREG-0730, "Report to Congress on the Acquisition of Reactor Data for the NRC Operations Center."

These reports summarize the status of many of the actions taken to date and provide the basis for continued upgrading of the NRC Incident Response Program.

The NRC Incident Response Plan assigns responsibilities for performing the functions and making the decisions that comprise the NRC response. The NRC plan will be made consistent with plans being prepared by the Federal Emergency Management Agency.

As for physical facilities, the NRC Headquarters Operations Center is operable in Bethesda, Maryland, and each of the five regional headquarters has an operational incident response center.

PLANNED ACTIVITIES

As stated previously, the present facilities and equipment of the NRC Headquarters Operations Center do not reflect the current state of planning or thinking that has been accomplished by the personnel responsible for the design and operations of the Center. A complete new conceptual design has been prepared; however, there have been no long-range commitments for the implementation of this design. Consequently, human factors concerns that exist now may well be reduced or eliminated if the resources are committed to implement a longer-range system concept.

Documentation describing a new plan for the headquarters design and operations includes:

- o NUREG/CR-1739, "Conceptual Design of the NRC Headquarters Operations Center"

- o MITRE Report 79W00393, "Communications System Specifications for the Incident Response Center."

The complete implementation of the incident response facilities is obviously a significant undertaking by the NRC and it is beyond the scope of the present effort to comment on the specific needs for the facilities.

No significant plans to change the regional incident response center design or operations are known to the authors at this time.

MISSING ELEMENTS

Nothing applicable at this time.

TECHNICAL FEASIBILITY AND PROBLEMS

There do not appear to be any technical feasibility or other problems which would preclude resolution of the human factors concerns.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The entire network of personnel, facilities, communications, procedures, etc., involved in planning and response to incidents represents extensive interaction among all components. Those components considered in this section specifically interact with the other components to be discussed in the sections which follow dealing with utility emergency response facilities and the safety parameter display system.

RECOMMENDATIONS

The focus for human factors concerns during this project was the nuclear power plant control room. The NRC indicent response facilities go well beyond the control room, but were briefly assessed because of a special request by NRC.

A complete systems analysis of the NRC incident response need and the facilities to meet that need should be done to derive human performance requirements. These requirements, which will be primarily decision making tasks, can then be further task analyzed to determine specific information and communication requirements and ultimately job designs and the necessary staffing and organization. The proper human factors evaluation of the NRC Operations Center can then be performed and design specifications prepared based on these analyses.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 1 year

Resources: 3 person-years

Implementation: Skills required are a career human factors professional, and the NRC Operations Staff.

Dependencies: None

6.3 Utility Emergency Response Facilities

The utility emergency response facilities are part of the overall incident response plan and facilities as discussed in the previous section. This section is limited to those utility emergency response facilities found onsite (or nearby) which are part of the nuclear power plant facility. The safety parameter display system (SPDS) may be considered part of the onsite facilities, but it is discussed separately in the section to follow because of its unique importance and human factors considerations. The emergency response facilities (ERF) discussed in this section are the technical support center (TSC), onsite operational support center (OSC), and nearsite emergency operations facility (EOF), as well as a brief discussion of the nuclear data link (NDL).

REQUIREMENTS

Emergency Response Facilities have not been systematically observed at many different sites since most utilities are in the process of designing them. Therefore, our review and assessment are based upon:

1. The requirements for the ERF as documented in NUREG-0696 and NUREG-0814.
2. Discussions with numerous utility companies regarding their plans and progress.
3. Discussions with architect engineer firms regarding their commitments on some emergency response facilities.
4. Discussions with INPO, EPRI, and NSAC regarding research and development they are performing, particularly that concerned with the safety parameter display system (SPDS).

The human factors concerns for each of the four emergency response facilities identified above are described below.

6.3.1 Technical Support Center (TSC)

The TSC requirement in general seems responsive to human factors issues derived from the experience at TMI-2. Relieving the control room personnel of tasks for communications not directly related to reactor control is certain to increase the effectiveness of personnel in the control room. Similarly, simply removing some personnel from the control room and providing more technical support when requested appears to be a positive feature.

Some general human factors concerns that must be considered in the final design and operation of the TSC are:

1. The actual number of personnel, their responsibilities, and specific information and communication needs to carry out those responsibilities. This suggests some form of job analysis although some of this information may be in licensees emergency response plans.

2. Layout and arrangement of equipment and work space to optimize movement and coordination.

3. Training for TSC staff personnel.

Very little human factors criteria have been developed to address the above concerns.

6.3.2 Operational Support Center (OSC)

The OSC also appears to be responsive to some manpower and personnel needs and coordination tasks as derived from the TMI-2 accident. In general, the OSC should reduce unnecessary personnel and traffic in the control room, and serve as a central point for logistics support. The number of personnel and their responsibilities should be defined and presumably will be contained in the licensee's emergency response plan. No other specific human factors issues have been identified.

6.3.3 Emergency Operations Facility (EOF)

The EOF will be the basis for overall management of the emergency response by the licensee, including coordination with federal, state, and local officials. The responsibility of the EOF to adequately and reliably implement emergency response actions involving the general public will require displays, communications, personnel, staffing, and procedures, all of which have implications for human factors issues.

Two specific human factors issues have been raised concerning the EOF to date.

First, there is a question as to what data are required in the EOF to support the functions of that facility. NUREG-0696

(Section 4.8) requires the entire REG. Guide 1.97 Data Set in addition to the REG. Guide 1.23 and NUREG-0654 Revision 1 Appendix 2 Data to be provided in the EOF. These data requirements do not appear to be derived from any kind of functions analysis or job analysis and should be re-examined in this context.

Second, there is the question of whether the SPDS displays should also be provided to the EOF as required by Section 4.2 of NUREG-0696. The EOF primary purpose is to provide a near site facility for the management of overall licensee emergency response (including coordination with federal, state and local officials), coordination of radiological and environmental assessments, and determination of recommended public protective action. The human factors issue is what data are required to support the above responsibilities. If the EOF does not involve the diagnosis of plant conditions, which presumably is the responsibility of the TSC, the question of providing too much data, including the SPDS, is a legitimate human factors concern because the EOF personnel can be distracted from their primary responsibility.

6.3.4 Nuclear Data Link (NDL)

The NDL is proposed as a data transmission system for providing reactor performance data for the NRC Headquarters Operations Center. The display at Headquarters could be equivalent to an SPDS. Aside from the human factors issues of the SPDS identified above, the critical human factors issue here seems to be the use of such information. The intent would be that the NDL and its display system provide plant system data to be used by the NRC for analysis and technical support -- not for management. The concept would appear to unburden some of the other communications between the site and the NRC Headquarters but needs to be more thoroughly considered in light of the mission and responsibilities of the NRC Headquarters Operations Center, as discussed earlier.

Human factors concerns in the onsite emergency response facilities were an incidental task in this project and were not the main thrust of the effort which was oriented towards the control room. However, the human factors concerns discussed above could be significant in the event of an alert or emergency simply because large numbers of people will be involved and a great deal of coordination will be required between individuals and organizations.

CONSTRAINTS

There do not appear to be any major constraints which would impede the application of good human factors to the design and operation of the site emergency response facilities. Since the emergency preparedness plan and the ERFs for each facility will be different, the design and implementation must be tailored to each separate installation. Similarly, state and local organization policies will vary and thus also constrain the design and implementation of functions of the site ERFs.

PRESENT STATUS

The NRC has issued two principle documents for guiding the design and evaluation of onsite emergency response facilities:

NUREG-0696, "Functional Criteria or Emergency Response Facilities," describes the facilities and systems to be used by the nuclear power plant licensees to improve responses to emergency situations. The facilities include the Technical Support Center (TSC), Onsite Operational Support Center (OSC), and Nearsite Emergency Operations Facilities (EOF), as well as a brief discussion of the emergency response function of the control room. The data systems described are the Safety Parameter Display System (SPDS) and the Nuclear Data Link (NDL).

NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities," was prepared to provide licensees an early insight into the approach the NRC staff will use in reviewing emergency response facilities proposals. The text is primarily in the form of questions, which will be used by the staff to review conceptual designs which are presently being submitted by licensees. It covers all of the emergency response facilities that are identified in NUREG-0696 and, in addition, has a chapter on system support requirements which has a few questions concerning user documentation and training, although no specific requirements or criteria are identified for these concerns. NUREG-0814 was published in August 1981 as a draft report for comment and will be re-issued in final form at a later date.

It is beyond the scope of this project to present a description of what the industry is doing with respect to site emergency response facilities. In general, because of the individual differences required at each site, there seem to be plans under way for the design and construction of the major facilities, with the exception of the nuclear data link. Thus, the TSC, OSC, and EOF are in various stages of construction. However, the emphasis seems to be on the construction of the facility, and much less emphasis on the instrumentation and controls, communications, and other features important from the human factors point of view which would be derived from use considerations. In short, attentional operational use requirements seems to be minimal with little or no job or task analysis which would define the requirements for data, communications, and staffing in a systematic manner.

One exception is the emergency response information system (ERIS), which has been conceptually described by the BWR owners' group. This system is less concerned with the facilities and more concerned with the information in those facilities, and will be described in somewhat more detail under the section to follow dealing with the safety parameter display system.

PLANNED ACTIVITIES

In the near future, the NRC will re-issue NUREG-0814 as a final report. The staff apparently is prepared to review conceptual designs of site emergency response facilities conceptual design submitted by licensees. No specific longer-range activities are planned by the NRC as far as the authors of this report know. An exception to this statement might be in the longer-range plan of the NRC to look at human factors issues such as management and organization, personnel and staffing, and licensing of other personnel besides operators. These projects might conceivably look at the personnel and human factors issues related to the site emergency response facilities.

MISSING ELEMENTS

The completeness of the activities in this area of concern was discussed above and no specific missing elements can be singled out for separate discussion.

TECHNICAL FEASIBILITY AND PROBLEMS

There are no technical impediments to resolving human factors issues of concern in the design and operation of the onsite emergency response facilities. A relatively straightforward, top-down, system analysis approach which would identify the specific functions of each facility and then would proceed to identify the necessary personnel and human factors requirements can and should be done.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The utility emergency response facilities are part of the overall incident response plan and facilities. Thus, the ERFs discussed in this section would interact with the general incident response plan (6.1) and facilities of the NRC (6.2) discussed in previous sections and will also interact with the Safety Parameter Display System discussed in the section to follow. It is also important to note that all of the facilities and activities associated with the emergency response plan and facilities will, of course, require interaction with the activities in the control room in the event of an emergency. This includes consideration of the changes which may be made in the control room as a result of NUREG-0700, and Reg. Guides 1.47 and 1.97.

RECOMMENDATIONS

It is recommended that some form of system analysis should be conducted to determine the human factors requirements for the emergency response facilities. This should be a straightforward analysis starting with the identification of major functions of each of the facilities and then some form of job or task analysis to identify the responsibilities expected of the user personnel. Design of specific procedures, displays, and the facility layout would evolve from this analysis. The NRC should provide some guidance on conducting this type of system analysis and they should further provide evaluation criteria for the review of designs submitted in accordance with that guidance.

Importance: High

Schedule: Urgency - 1-2 years

Duration - 2 years

Resources: 8 person-years

Implementation: Skills required are a career human factors professional, a nuclear engineer, and an I&C engineer.

Dependencies: None

6.4 Safety Parameter Display System (SPDS)

The SPDS is part of the emergency response facilities, but is being treated separately in this report because of its significance from the human factors point of view. However, the previous sections dealing with the incident response plan and the NRC and the utility emergency response facilities are relevant to this section and should be reviewed if the reader is not familiar with them.

The purpose of the safety parameter display system is to assist control room personnel in evaluating the safety status of the plant. The SPDS will be located in the control room with additional SPDS displays provided in the TSC and EOF. An SPDS is apparently also being considered for installation at the NRC Headquarters Operation Center.

REQUIREMENT

The primary function of the SPDS is to aid the operator in the rapid detection of abnormal operating conditions. The SPDS will, therefore, directly affect the operator's role in the control room and is thus considered to be the most significant human factors issue of the emergency response facilities.

As an operator aid, the SPDS will be vital to control room personnel to provide awareness of the plant status during steady state, transient, and abnormal or emergency conditions. It is thus imperative that all facets of human factors, including human engineering, procedures, personnel training, and operator acceptance be considered in this design. NUREG-0696 does acknowledge the importance of human factors and specific evaluation criteria for human factors is provided in NUREG-0835.

Further, it appears that a backup SPDS will be required if the primary SPDS display will not function during an earthquake. This implies that the backup system would be made up from other required control room instrumentation needed to comply with REG Guide 1.97. It also implies that the backup instrumentation be concentrated in one area on the control board. Because of the significance of the SPDS for control room personnel to assess the overall safety status of the plant several human factors issues must be addressed:

1. The requirement for a backup SPDS with a different design and different instrumentation should be reconsidered (although the seismic qualification requirement is, of course, beyond the scope of any human factors issue). If the backup instrumentation is different than the primary SPDS, some question exists about the utilization and acceptance of two different systems.

2. If a backup is required, the need to install separate seismic instrumentation in a concentrated area is also questionable from a human factors viewpoint. Further, this may also conflict with the design criteria of REG Guide 1.97 which states, "it is prudent to select the required accident-monitoring instrumentation from the normal power plant instrumentation to enable operators to use, during accident situations, instruments with which they are the most familiar." This suggests that the instrumentation will be associated with the various systems and not concentrated in one area.

3. A considerable amount of research and development is being carried out by industry to develop safety parameter display system concepts. However, in most cases, the R & D efforts are focusing on new technology display systems with minimum regard for human factors. New technology does not equate to good human factors, and this final concern may well have an effect upon both initial acceptance and long-term utilization by operating personnel.

The SPDS is an important and potentially costly modification to the control room, in addition to being required for the TSC and EOC. The underlying need for an SPDS has not been clearly established, and it represents a significant human factors concern with respect to user acceptance. See the discussion on this subject in Section 3.1.1, Volume 2 of this report.

CONSTRAINTS

There are no significant constraints identifiable with the implementation of the SPDS concept, only with the definition of its design requirements.

PRESENT STATUS

The NRC has issued NUREG-0696, "Functional Criteria for Emergency Response Facilities," which defines the general requirements for the safety parameter display system. According to NUREG-0696, the SPDS ". . .provides a display of plant parameters from which the safety status of operation may be assessed in the control room, TSC, and EOF. The primary function of the SPDS is to help operating personnel in the control room make quick assessments of plant safety status." NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System," has been issued for public comment.

The industry currently has a number of research and development activities to develop safety parameter display system concepts. Several vendors appear to be developing concepts which they presumably will try to sell to the utilities; owners' groups are developing SPDS concepts, and NSAC has contracted for several studies for SPDS concepts and hardware and software alternatives.

In brief, SPDS activities to date do not appear to have been well integrated, and many alternative approaches are being pursued. From a human factors point of view, it would appear prudent to define the user needs more explicitly before detailed design alternatives are pursued. This suggests some test and evaluation plan which incorporates not only engineering considerations but human factors ones as well.

PLANNED ACTIVITIES

The NRC finalized NUREG-0835, "Human Factors Engineering Design Review Acceptance Criteria for the Safety Parameter Display System," in early 1982. The draft report has been reviewed and the following comments can be made. In general, the content and format appear to be quite well done. A few specific comments follow. First, there does not seem to be much, if any, input from professional human factors personnel; second, the document relies heavily on guidelines developed in NUREG-0700; and third, the document does not address the user needs of tasks, but rather, addresses the more specific considerations of the man-machine interface.

From the industry side, the most significant activity planned is by the EPRI, which has a study scoped to address some of the designs of the owners' groups (and perhaps others) and has obtained the services of a recognized professional human factors contractor to support this effort. The specifics of this planned activity are not known at this time, and cannot be further reported on.

One further significant industry activity mentioned in the section on utility emergency response facilities is the BWR Owners Group Emergency Response Information Systems (ERIS). ERIS is described as an "integrated system that gathers the required plant data, stores and processes that data, generates visual displays for the operator and other personnel who need plant status informaton, provides printed records of transient events, and has the capability to transmit essential information to the NRC should this become a requirement. The basic components of ERIS are the Data Acquisition System, the Central Processor Units, and the Graphic Display Consoles." Specifics of the ERIS are proprietary information and cannot be discussed in this report. The information received about ERIS is not sufficient to judge its adequacy from a human factors point of view. It does appear, however, to be primarily hardware and computer system oriented, with some consideration given to the man-machine interface, but apparently no user needs or task analysis has been performed.

MISSING ELEMENTS

From a human factors point of view, the missing element with respect to the safety parameter display system seems to be any form of functional analysis, user needs analysis, or task analysis to support either the need for the SPDS and to or its basis for deriving the specific information requirements.

TECHNICAL FEASIBILITY AND PROBLEMS

Technical feasibility does not represent a problem for SPDS. Rather, the reverse might be true in that the state of the art in display systems and computer systems might be driving the design of many SPDS alternatives, rather than the functional and user requirements. Similarly, as stated earlier, the requirement for a backup SPDS with a different design and configuration can present human factors problems for control room operators. Finally, the need for the SPDS in the TSC and EOF in particular, has been even less thoroughly established than has the need for the SPDS in the control room.

INTERACTION WITH OTHER SYSTEM REQUIREMENTS

The SPDS is part of the overall emergency response facilities, and therefore will interact with the other elements of those facilities as described in NUREG-0696. Discussion of the onsite facilities and the incident response plan, including the NRC Emergency Operations Center, were provided in the three sections of this report preceding this section on SPDS.

In addition, the SPDS interacts with all other principal elements of the personnel subsystem, particularly human engineering, procedures and operator aids, and training.

RECOMMENDATIONS

The need for an SPDS has not been established from any system or task analysis effort. A well-designed control room may be satisfactory without an SPDS. Therefore, a thorough systems analysis should be done. The job/task analysis being done by INPO and the Reactor Operator task analysis being done by the NRC must be coordinated with any similar analysis for SPDS. The general approach and recommendations for any type of operator aid, including the SPDS, are discussed in Section 3.3, Operator and Maintenance Aids.

If any SPDS is to be developed, then the following tasks must also be a part of the human factors considerations:

(a) Evaluate the need for a backup SPDS as specified in NUREG-0696.

(b) If a backup SPDS is required, evaluate the need to install separate seismic instrumentation in a concentrated area.

(c) Review the potential conflict with REG Guide 1.97.

(d) Develop evaluation criteria for user acceptance.

Finally, the work effort described above should be integrated with that recommended in Section 3.3, Operator and Maintenance Aids.

Importance: High

Schedule: Urgency - immediate

Duration - 1-2 years

Resources: NRC staff, plus an undetermined level of effort in conjunction with Section 3.3

Implementation: Personnel skill required are a career human factors professional, computer systems analysts, and reactor safety parameters subject matter experts.

Dependencies: Should be contingent upon the INPO and NRC job/task analysis efforts. However, a unique special systems analysis could be performed which would not delay SPDS if this analysis justifies the need.

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LIST OF ACRONYMS AND INITIALISMS

The following acronyms and initialisms have been used in this report. They are listed in alphabetical order for the convenience of the reader.

ACRS	Advisory Committee for Reactor Safeguards
AE or A/E	Architect-Engineer
AEC	Atomic Energy Commission
AEOD	(Office of) Analysis and Evaluation of Operational Data
AFSCDH	Air Force Systems Command Design Handbook
AFSCR	Air Force Systems Command Regulations
AIF	Atomic Industrial Forum
ANS	American Nuclear Society
ANSI	American National Standards Institute
BNL	Brookhaven National Laboratory
BOP	Balance of Plant
BPNL	Battelle Pacific Northwest Laboratory
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CAI	Computer-Aided Instruction
CEMS	Critical Equipment Monitoring System
CFMS	Critical Function Monitoring System
CODAP	Comprehensive Occupational Data Analysis Program
CP	Construction Permit
CR	Control Room
CRDR	Control Room Design Review
CRGR	Committee to Review Generic Requirements

CRT	Cathode-ray Tube
DASS	Disturbance Analysis and Surveillance System
DCRDR	Detailed Control Room Design Review
DEDROGR	Deputy Executive Director for Regional Operations and Generic Requirements
DHFS	Division of Human Factors Safety
DOD	Department of Defense
DOE	Department of Energy
EDO	Executive Director for Operations
EEI	Edison Electric Institute
EOF	Emergency Operations Facility
EPRI	Electric Power Research Institute
ERF	Emergency Response Facilities
ESF	Engineered Safety Feature
FAA	Federal Aviation Administration
FSAR	Final Safety Analysis Report
HE	Human Engineering
HEP	Human Error Probability
HER	Human Error Rate
HFEB	Human Factors Engineering Branch
HF	Human Factors
HFE	Human Factors Engineering
HFS	Human Factors Society, Inc.
HIAPSD	Handbook of Instructions for Aerospace Personnel Subsystem Designers
HTGR	High-Temperature Gas-Cooled Reactor
IE	(Office of) Inspection and Enforcement
IEEE	Institute for Electrical and Electronics Engineers

IEORS	Integrated Operational Experience Reporting System
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IREP	Interim Reliability Evaluation Program
ISD	Instructional System Design
ISEG	Independent Safety Engineering Group
ISFSI	Independent Spent Fuel Storage Installation
JPA	Job Performance Aid
LER	Licensee Event Report
LLNL	Lawrence Livermore National Laboratory
LLTF	Lessons Learned Task Force
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test
LQB	Licensee Qualifications Branch
M-M	Man-Machine ("Man" is used in the generic sense.)
NDL	Nuclear Data Link
NPP	Nuclear Power Plant
NPRDS	Nuclear Plant Reliability Data System
NRC	U. S. Nuclear Regulatory Commission
NREP	National Reliability Evaluation Program
NRR	(Office of) Nuclear Reactor Regulation
NSAC	Nuclear Safety Analysis Center
NSSS	Nuclear Steam Supply System
NTOL	Near-Term Operating License
ODPS	Operator Diagnostic and Display System
OJT	On-the-job Training

OL	Operating License
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
OSC	Operational Support Center
PDRI	Personnel Decisions Research Institute
PMS	Performance Measurement System
PORC	Plant Operations Review Committee
PRA	Probabilistic Risk Analysis
PSF	Performance Shaping Factor
PTRB	Procedures and Test Review Branch
PWR	Pressurized Water Reactor
RES	(Office of) Nuclear Regulatory Research
RO	Reactor Operator
SAT	Systems Approach to Training
SER	Safety Evaluation Report
SIG	Special Inquiry Group
SIMCERT	Simulator Certification
SME	Subject Matter Expert
SNL	Sandia National Laboratory
SPAR	Standards Project Authorization Request
SPDS	Safety Parameter Display System
SRG	Special Review Group (of the Office of Inspection and Enforcement)
SRP	Standard Review Plan
SRO	Senior Reactor Operator
SS	Shift Supervisor
STA	Shift Technical Advisor
TAG	Technical Advisory Group

TEA	Training Effectiveness Analysis
TIM	Task Identification Matrix
TSC	Technical Support Center
TMI-2	Three Mile Island, Unit Two
TVA	Tennessee Valley Authority
USAF	United States Air Force

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This comprehensive long range human factors plan for nuclear reactor regulation was developed by a Study Group of the Human Factors Society, Inc. This Study Group was selected by the Society to provide a balanced, experienced human factors perspective to the application of human factors scientific and engineering knowledge to nuclear power generation.

The report is presented in three volumes. Volume 1 contains an Executive Summary of the 18-month effort and its conclusions. Volume 2 summarizes all known nuclear human factors activity, evaluating this activity, wherever adequate information is available, and describes the recommended long-range (10-year) plan for human factors in regulation. Volume 3 elaborates upon each of the human factors issues and areas of concern that have led to recommendations in the long range plan.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Human factors engineering
Human engineering
Procedures
Job performance aids
Personnel selection
Staffing
Training
Training devices
Human feasibility
System integration
System engineering
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