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An Initial Empirical Analysis of Nuclear Power Plant Organization and Its Effect on Safety Performance

Prepared by J. Olson, S. D. McLaughlin, R. N. Osborn, D. H. Jackson

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Pacific Northwest Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This report contains an analysis of the relationship between selected aspects of organizational structure and the safety related performance of nuclear power plants. The report starts by identifying and operationalizing certain key dimensions of organizational structure that may be expected to be related to plant safety performance. Next, indicators of plant safety performance are created by combining existing performance measures into more reliable indicators. Finally, the indicators of organizational structure were related to the indicators of plant safety performance using correlational and discriminant analysis. The overall results show that plants with better developed coordination mechanisms, shorter vertical hierarchies, and a greater number of departments tend to perform more safely.

EXECUTIVE SUMMARY

Many informed observers have proposed that utility management is a key element underlying the safe operation of nuclear power plants (NPP). One way that management likely influences plant safety performance is through the organizational structures it consciously creates or allows to exist. This report examines the relationships between some important dimensions of plant organizational structure and measures of plant safety performance. While only an initial effort, the results reported here establish (a) the feasibility of reliably measuring plant safety performance, (b) the importance of organizational structure in determining plant safety performance, and (c) the potential usefulness of this approach in regulatory activity.

The report documents work performed on three tasks. The first task concerned the creation of measures of organizational structure. An earlier review of the literature supported the position that organizational structure (e.g., the way the work of the organization is divided, administered, and coordinated) is a likely determinant of plant safety performance. While data were not available on some salient dimensions of organizational structure, Final Safety Analysis Reports (FSARs), Technical Specifications, and a survey of plant technical resources (NUREG/CR-1656) allowed for measurement on three primary dimensions. These are the vertical structure of the plant (e.g., the number of ranks and the ratio of supervisors to subordinates, the horizontal structure of the plant (e.g., the way the organization is divided into administrative and work units), and the coordinative structure of the plant (e.g., the ways that work units are linked).

The second task was concerned with the creation of indicators of plant safety performance. Earlier reviews of plant performance data have identified several major problems with using them as indicators of safety performance. First, the available measures tend to be influenced by factors other than safety performance <u>per se</u> (e.g., vendor, age, size, region). Second, each performance measure reflects only part of what is meant by safety. To offset these weaknesses, Licensee Event Reports (LERs), forced outages, Inspection and Enforcement (I & E) violations, and Systematic Assessment of Licensee Performance (SALP) data were adjusted for vendor, age, size, and region, and combined in a factor analysis. The results of this analysis identified four general indicators of plant safety performance: Regulatory Noncompliance, Human Error, Hardware Failure, and Plant Nonreliability. For 1981, plants could be given performance scores on each of these four safety indicators.

The third task was to correlate the measures of organizational structure with the four plant safety performance indicators. The results of this analysis show that organizational structure is related to each of the safety indicators, though not always in the same way. For instance, plants with a larger number of vertical ranks generally had poorer safety performance. Plants with more departments and a larger ratio of subordinates to supervisors (except in operations) had better safety performance. Plants with better developed coordinative mechanisms tend, also, to have safer performance. The data are limited to 1981 and to about two-thirds of the currently operating plants.

Based on these analyses, the following statements can be made:

- It appears feasible to use existing plant performance data to create refined, reliable indicators of plant safety performance.
- The plant safety performance indicators are potentially useful for identifying poor performers, identifying correlates and causes of poor performance, and for providing base safety performance measures against which attempts to improve safety performance can be evaluted.
- Organizational structure appears to be an important predictor of plant safety performance. Nuclear Regulatory Commission (NRC) concern with organizational structure and other aspects of utility management appears to be warranted.

1. INTRODUCTION

This report summarizes work to date on an empirical analysis linking selected aspects of nuclear power plant (NPP) organizational structure to a set of plant performance indicators of safety[1].

1.1 Objectives and Background

The objective of this report is to describe an initial empirical analysis of associations between organizational characteristics and safety performance for a sample of NPPs. This analysis was performed as part of the "Utility Management and Organization Guidelines" project in the Pacific Northwest Laboratory (PNL) contract funded under FIN B2360. The objective of this larger effort is to assist the Nuclear Regulatory Commission (NRC) in developing a technical basis for evaluating the management capabilities of utilities seeking NPP operating licenses. Draft guidelines have been created to help assess proposed organization and administration plans to be submitted by each utility seeking an NPP operating license.

While considerable anecdotal evidence exists indicating the importance of organization and administration for NPP safety, minimal systematic empirical analysis of this relationship has been conducted. Thus, there was a need to conduct the empirical analyses described in this report that could 1) support the NRC's use of resources in regulatory activities in this area, 2) identify the organizational characteristics important to safe plant performance, and 3) suggest alternative ways to organize for safer NPPs.

1.1.1 The Development of NPP Safety Indicators

One of the important first steps to an analysis of the relationship between organization and safety is the development of logical and defensible measures of NPP safety performance. Existing measurements of safety (e.g., LERs) have been justifiably criticized on the grounds of

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[1]This report was written under a contract to Pacific Northwest Laboratory and the Battelle Human Affairs Research Center for the Division of Human Factors Safety, Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC) (FIN B2360). Though based primarily on the project "Utility Management and Organization Guidelines," this work has also benefited from the support of two other projects: the "Manpower and Staffing" project of NRC's Safety Technology Program for work on the safety indicators used in the analysis, and the "Management and Organizational Safety Assessment" project supported by the Division of Facility Operations, Office of Nuclear Regulatory Research (FIN B2457), for work on the development of the organizational perspective on NPP safety reflected in this analysis.

both reliability and validity. This report attempts to satisfy the need for more reliable and valid assessment of NPP safety performance through the use of a multiple indicator approach. We will discuss the problems associated with using existing approaches to measuring NPP safety, and will describe the analytical approach we chose to ameliorate these problems. Essentially, this analysis suggests that existing measures can be combined into more reliable indicators.

1.1.2 The Development of Organizational Structure Indicators

A second prerequisite for an empirical analysis of the relationship between NPP organizational structure and safety performance is the development of indicators of important dimensions of organization. This report draws on the organizational literature to define such important dimensions, assesses the availability of data on these dimensions in existing sources, and describes the methods used in creating indicators of organizational structure from the existing data.

1.1.3 The Relationship Between Organizational Structure and NPP Safety

After describing the development of both the organizational structure and plant safety performance indicators, the report analyzes the relationship between them. The analysis attempts 1) to show the overall importance of organizational structure for NPP safety, 2) to identify specific relationships between characteristics of organizational structure and indicators of safety performance, and 3) to identify potential areas for further analysis.

1.2 Limitations of the Report

It is important at the outset to point out the limitations of the analysis reported here. The analyses are based on the available data which are incomplete and not formally designed for analytical purposes. Consequently, the results reported here should be used tentatively and to guide future investigations. Within these limits the report demonstrates promise for the creation of improved ways of measuring plant safety performance, and support for the contention that organizational structure is central to the determination of NPP safety. The results provide considerable insight into the potential safety significance of various organizational and staffing decisions made by the management of NPPs.

2. AN ORGANIZATIONAL APPROACH TO NPP SAFETY

This section describes the general organizational approach to NPP safety that guides the subsequent analyses. The approach was developed and reported in NUREG/CR-3215 which provides a detailed discussion and review of the relevant literature. Here the concern is with briefly outlining the approach and describing the construction of the organizational measures used in the present analyses.

2.1 An Organizational Perspective on NPP Safety

The perspective taken in this report starts with the assumption that a NPP is a socio-technical system. Both technical (hardware and procedures) and the social factors (e.g., administration, communications, group structures) need to be considered in this system. A key to the organizational perspective is that it takes a systems approach. A systems orientation requires the consideration of the relationships among the technical and social factors as the main area of concern. The contribution of each social or technical factor to NPP safety, then, is to be understood in terms of the position it holds and the role it plays in the overall system. The emphasis is on examining patterns of activity, rather than the isolated individual, particular procedure, or piece of equipment.

The focus on the patterns of activity among technical and social factors is the source of our concern with organizational structure as a cause of NPP safety. Organizational structure refers to the ongoing patterns of interaction, whether formally defined by management or simply emergent in day-to-day activity. The organizational structure defines the nature of the significant social units in the plant, and also provides the linkages among the units and of the units to the technical dimensions. In a sense, the organizational structure is what allows the discrete social and technical elements to act as a system, that is, to interact. While there is very little empirical literature which has attempted to link organizational structure to NPP safety performance, our review of the wider organizational literature (NUREG/CR-3215) suggests a number of mechanisms whereby structure can influence safety performance.

The formal structure is important because it establishes a basis for determining what work is to be done, who will perform given tasks, what standards are to be applied to work, as well as how activities are to be budgeted, coordinated, and controlled. In other terms, the formal structure is a primary administrative tool available to management. A well designed formal structure allows the other aspects of the socio-technical system (staff, money, and equipment) to meet normal problems effectively, anticipate repeated difficulties, and adapt to unforeseen challenges. While structure is not the only important aspect of the socio-technical system, it has been repeatedly identified in other settings as a major factor influencing performances (see NUREG/CR-3215).

Because structure is seen as a tool for management and part of a more complex socio-technical system, it is important to recognize that a single optimal structure for all phases and purposes of the organization is unlikely to be found for the industry. This does not mean that all structures are equally safe. Certain approaches may be unsafe under any conditions. Alternatively, a range of approaches may maximize safe performance across a range of utility settings and characteristics. Different types of technical systems, different organizational histories and unique forms of management philosophy may call for different structures. For instance, one does not expect the structure needed for a small, single-unit facility to be the same as needed for a large multi-unit operation.

Within the field of organizational analysis, the term structure has been used in several different ways. It will be defined below. Here it is important to realize that some of the factors utility executives often consider a part of management are put under the label of structure. That is because this report focuses on underlying patterns of action reflected in the formal documents of the utility. Thus, questions of coordination, control, and development are considered aspects of structure. The term structure goes far beyond a simple table of organization.

2.2 Dimensions of Organizational Structure for Nuclear Utilities

Utility management faces a number of choices concerning the type of structure to employ. These choices revolve around the questions of how to subdivide the work, control the activities of organizational members, coordinate the activities of different specialized units, and implement positive change and prepare for the future. While utility management may use a variety of resources to implement choices (e.g., policies/procedures, technical aids, individuals, external assistance (such as consultants), the formal structure is a central strategy. In this report, we focus on the use of formal structure to subdivide the work into specialized units, and organize the units in systems of control and coordination.

2.2.1 Formal Structure of Work

The formal structure of work refers to the planned assignments of duties, responsibilities and reporting requirements. There are two key issues. The first issue is the decision of how to subdivide the total job into discrete tasks or functions. How many and what types of units are needed? Different choices will yield different numbers and types of departments. Generally, if organizations decide narrowly defined specialties are needed and should be grouped together, one finds numerous units at the bottom of the organization. If, on the other hand, broader jobs are defined, or if generalists are needed, there are often comparatively few units at the bottom of the organization. (See Osborn, et al., 1980, and NUREG CR/3215 for a discussion of the organization of work.) Normally, one expects to find a division between line and staff activities. Due to the complexity of nuclear operations a further distinction can be made. Line activities refer to the work of operations personnel. Staff activities refer to administrative, clerical, and other service type work. A third category, Intermediate, has been defined as those support jobs that are more directly related to plant design and/or safety performance than the staff jobs. These refer to the maintenance, health physics, and engineering activities at the plant.

A second issue concerns the number of ranks or levels of management needed to accomplish the work assigned to various units. With standardized routine jobs, there may be comparatively fewer ranks since managers are not needed to help solve job problems or resolve technical issues. With somewhat more difficult jobs, there may be many management ranks where each higher level manager helps solve even more difficult and unique problems. With highly specialized and unique professional tasks, one may again see comparatively few levels for resolving technical issues. Very simply, higher level managers cannot keep up technically and, thus, must defer to the professional judgments of subordinates[2].

The question of ranks or levels is also important because the tasks of managers can be expected to vary at different levels. Beginning at the top of an organization, one can often identify three different levels of management -- the institutional (top), the managerial (middle), and the technical (bottom). The institutional level is primarily concerned with linking the organization as a whole to its broader organizational environment. For example, the institutional level provides the linkages to the political arena, sources of revenue and the like. For the most part, it is comprised of top management. The managerial level focuses on the administration of the work and related processes while the technical level focuses on conducting the work itself.

Not all aspects of a utility's insitutional, managerial, or technical levels are equally relevant for NPP safety, though some aspects of all three levels will be crucial. For example, one logically expects that the technical units of operations, maintenance, and engineering-related staff will be more important than peripheral support areas (e.g. accounting, customer service). Figure 1 shows the relevant units of analysis as the shaded area cutting through the institutional, managerial and technical levels. One longer range goal would be to identify those aspects of each of the three levels which clearly are important for NPP safety. For now, the corporate, nuclear division, and plant organization are useful units of analysis for discussion purposes.

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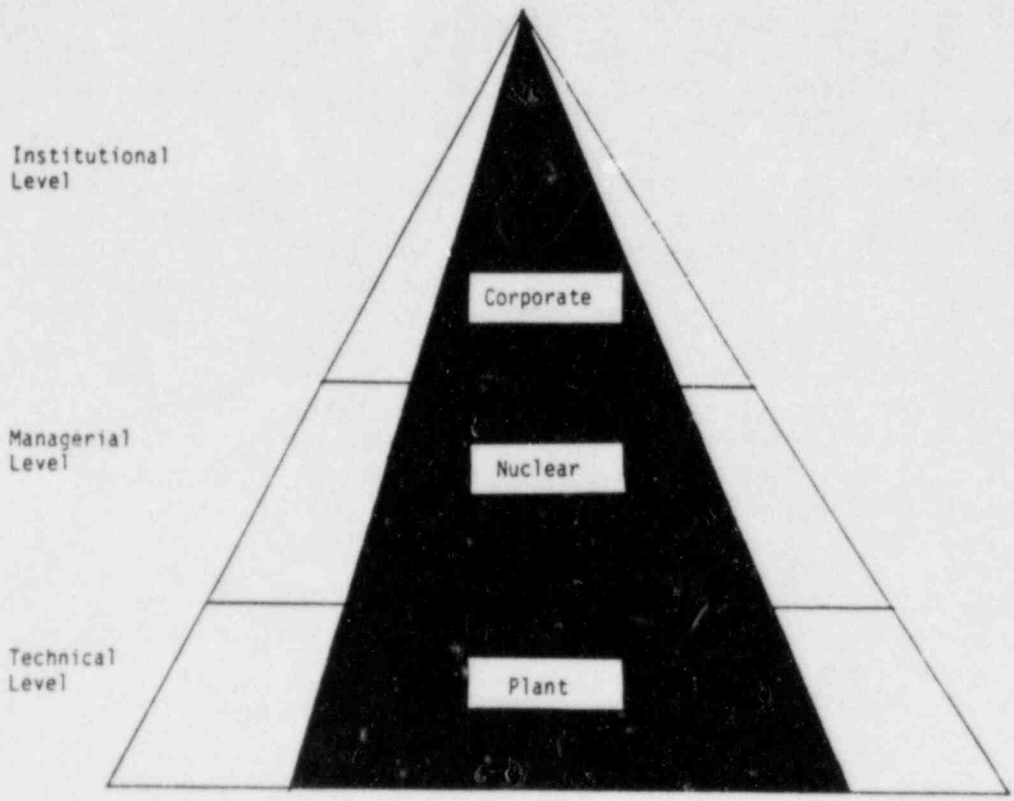
[2]Following Perrow (1983) and Osborn et al., (1980), these estimates should not be used to infer level of centralization since they pertain primarily to the technical and not the other aspects of management (e.g., budgeting and personnel selection).

FIGURE 1: LEVELS OF ANALYSIS FOR ANALYZING DIMENSIONS



It is expected that the three levels of analysis will be directly interconnected through a chain of command. This is reflected in the fact that the institutional, managerial, and technical units roughly correspond to the corporate, nuclear division, and plant units in the utility. However, the boundaries for these levels of analysis may not always be clear-cut. Particular types of activities, such as maintenance, may overlap the primary units of analysis. For a given utility, for instance, maintenance may be considered a "centralized" function which serves both fossil and nuclear plants. The division between plant and nuclear may or many not correspond to simple physical location. Individuals located "on-site" may be part of the nuclear unit of analysis, for example, because they are not directly linked to the plant hierarchy. Finally, selected units may be located and administered "on-site" but report directly to higher level officials (e.g. a corporate staff vice president) in corporate headquarters. Quality assurance (QA) is a common example. Nonetheless, such distinctions among units are useful for discussion and analytic purposes. In this report, we will focus on the technical, or plant level.

Because sites vary in terms of the number of reactors present, a number of additional units need to be defined. A <u>facility</u> comprises all administrative units located at a particular geographic location. A <u>plant</u> is all administrative units associated with a particular reactor-turbine combination licensed by the NRC. The reactor unit is the set of units reporting to the plant operations head.

To summarize, the number of units and the number of ranks within each unit of analysis (e.g., corporate, nuclear division, plant), are two important dimensions of the formal structure of work.

2.2.2 Coordination

Coordination deals with the linkage of specialized units and their outcomes into a whole. With such quite diverse and highly specialized tasks and associated units and levels, problems of coordination within nuclear plants are expected to be substantial. Several coordinative mechanisms are available. One mechanism is to deploy individuals with experience in one functional area, such as operations, into another, such as maintenance. Indicators of this type of coordination for nuclear plants might include the number of degreed individuals (engineers) in operations and maintenance and the number of licensed individuals outside of operations. A second approach to coordination is to provide for linkages in the formal structure itself. An example indicator might be the point in the hierarchy at which the key functional areas of maintenance, operations, and engineering are joined. Where the manager assigned authority for all three functions is far removed from actual work, coordination might be more difficult.

2.3 Reviews of Existing Data Sources

In this section a number of potential sources of data on organizational structure are reviewed. The purpose of the review is to identify the extent to which data are available for the empirical analysis necessary to analyze the relationship between organizational structure and NPP safety. (A more detailed review can be found in NUREG/CR-3215).

Our survey of potential sources of organizational information included the following:

(1) Final Safety Analysis Reports (FSARs), Chapter 13 (Conduct of Operations) and Chapter 17 (Quality Assurance Program Reviews)
(2) Technical Specification, Chapter 6 (Administrative Controls)
(3) Health Physics Appraisal Program Reviews (Cunningham, et al. 1982)
(4) Teknekron Evaluation of Utlity Management and Technical Resources (NUREG/CR-1656)
(5) Institute of Nuclear Power Operations (INPO) Staffing Survey (INPO 1981)
(6) Edison Electric Institute (EEI) Nuclear Plant Staffing Survey (EEI 1980)
(7) INPO Plant and Utility Evaluations (INPO 1982)
(8) Utility Annual Reports

These sources were chosen for review because each of them is relevant to organizational issues and covered most, if not all, of the operating plants. It was not possible to use additional sources of secondary data (such as Public Utility Commissions) because of the lack of a centralized source for these data and the time and resources that would be required to collect them.

Each source was evaluated on the basis of a set of criteria developed from both general methodological concerns and the requirements of the specific analytic problems posed by the need to link organizational and safety data. Table 1 presents a summary of our evaluation of the various data sources.

As can be seen from Table 1, there are limitations on the ability of existing data sources to support either a complete mapping of the central dimensions of organization or a comprehensive analysis of the relationship between these dimensions and NPP safety. Specifically, the following factors serve to limit the available data:

(1) Many of the potentially useful data sources are proprietary, and have not been made available for public use. This is particularly crucial in the areas of staffing, since no publicly available source of this important aspect of organization exists for all plants.

TABLE 1: REVIEW OF EXISTING ORGANIZATIONAL DATA SOURCES

	Availability	Reliability	Validity	Timing	Scope	Concept Coverage	Overall Evaluation
FSARs	YES	SOME PROBLEMS	SOME PROBLEMS	FAIR IF UPDATED	FAIR ON PLANT AND CORPORATE	LIMITED TO CHART	USEFUL FOR INITIAL ANALYSIS
Technical Specifications	YES	SOME PROBLEMS	SOME PROBLEMS	FAIR IF UPDATED	FAIR FOR PLANT. WEAK FOR CORPORATE.	LIMITED TO CHART MEASURES	USEFUL FOR INITIAL ANALYSIS
Health Physics	YES	GOOD	GOOD	FAIR: DATA BECOMING OLD	LIMITED TO HEALTH PHYSICS FUNCTION	GOOD COVERAGE	CAN AUGMENT OTHER SOURCES
Teknekron	YES	FAIR	FAIR	FAIR: DATA BECOMING OLD	LIMITED TO TECHNICAL RESOURCES	LIMITED TO COORDINATIVE	CAN AUGMENT OTHER SOURCES
INPO Staffing Survey	NO	FAIR	FAIR	GOOD	GOOD	LIMITED TO STAFFING	USEFUL BUT NOT AVAILABLE
EEI Staffing Survey	NO	FAIR	FAIR	FAIR: DATA BECOMING OLD	GOOD BUT LIMITED TO PLANT	LIMITED TO STAFFING	USEFUL BUT NOT AVAILABLE
INPO Evaluations	NO	UNKNOWN	UNKNOWN	FAIR: LONG EVALUATION CYCLE	WIDE SCOPE	UNKNOWN	OF QUESTIONABLE USE AND NOT AVAILABLE
Utility Annual Reports	YES	WEAK	WEAK	ADEQUATE	WEAK	WEAK	NOT USEFUL

(2) The available data frequently suffer from problems of reliability, validity, and timing, since they are not collected primarily for the purpose of comparative analysis.

(3) The available data provide considerable information on administrative form, but provide much less coverage of the other aspects of organization.

Given these limitations, the analysis reported here cannot be comprehensive. However, the data will support an initial analysis organized around the organization of work into formal structures. Specifically, we have used existing data sources to compile organization charts, which have then been used to compute a number of basic measures of organizational structure. In addition, we have been able to look, tentatively, at the important area of coordination by employing information derived from the Teknekron survey of technical resources (NUREG/CR-1656). This initial examination has, despite data limitations, been able to demonstrate the importance of organization as an area predictive of NPP safety.

2.4 Specific Measures of Formal Structure

In this section we describe the organizational variables to be used in the analysis.

2.4.1 Compilation of Organizational Charts

Organizational charts have been used with success to investigate the formal structure of organizations (e.g. Blau and Schoenherr, 1971). Though there are limits on this approach (Perrow, 1983), it provides a very important view of the intended structure of work and authority relationships. For many organizations, the intended structure comes reasonably close to those relationships actually experienced. For even more organizations, the individual formal structure is an important underlying feature of its basic design.

Unfortunately, existing data sources do not provide complete, reliable, and timely organizational charts for analysis. Though a number of the sources contain charts (FSARs, Technical Specifications, Health Physics Appraisals), the charts are frequently of poor quality. The two major problems are that they are sometimes out of date and that they lack detail.

Reviews of the FSARs and the Technical Specifications suggest that the charts contained in the base documents are not always updated. This finding is not unique. A number of the INPO plant evaluations reviewed pointed out disparities between the actual organizational plan and the charts still contained in the Technical Specifications and FSARs.

A more important problem concerns lack of detail. In comparing Figures 2 and 3, it can be seen that there is a wide range in detail provided by the utilities. Figure 2 provides reasonably complete information on the plant, the entire range of functions, complete indications of lines of authority, and detail, even down to the staffing levels of first level workers. Figure 3, however, does not provide this amount of information. Only major units have been defined and their internal configurations have been ignored. It is not possible from such a chart to generate an accurate description of the organizational form. A substantial number of the existing charts are incomplete in this manner. Fortunately, other sources of information could sometimes be used to clarify and refine the charts.

A variety of sources were used to compile the composite charts. For each plant, the organizational chart in the Technical Specifications was compared to the chart in the FSAR. There was frequent disagreement between these two sources. When disagreement occurred, the source that was 1) most recent, and 2) most detailed was selected for further refinement. This strategy still yielded incomplete charts for many of the plants. Additional sources were used to augment the charts. Specifically, charts and narratives of the Health Physics Appraisals were employed to fill in some of the blanks. Personal knowledge based on site visits conducted for related projects was also employed.

This approach yielded charts for 42 plants. These charts provide reasonable detail and accuracy for the first level supervisor and above. To this point, accurate data on direct worker staffing patterns has not been found for most of the plants. It is anticipated that additional cases will be added to the current sample of 42 in future analyses. While it is feasible to develop equivalent charts for the corporate and nuclear levels, the current effort has been limited to the plant level.

2.4.2 The Derivation of Measures from the Charts

Once the charts were compiled, a number of measures concerning the formal structure of work and coordination were calculated. These included measures of 1) the vertical aspects of the formal structure, 2) the horizontal aspects of the formal structure, and 3) selected linkages across functional units to provide partial estimates of coordination.

2.4.2.1 Vertical Measures

Two general aspects of the plant's vertical configuration could be addressed -- the height of the organization (number of ranks) and leader deployment patterns (supervisory spans of control.) Both reflect key decisions regarding the structure of work. The height of the organization is the number of ranks from the first line supervisor to the top manager of the unit in question. Both the number of ranks in the longest and average lines were computed. Similarly, as a measure of the facility

FIGURE 2: COMPLETE CHART FROM FSAR

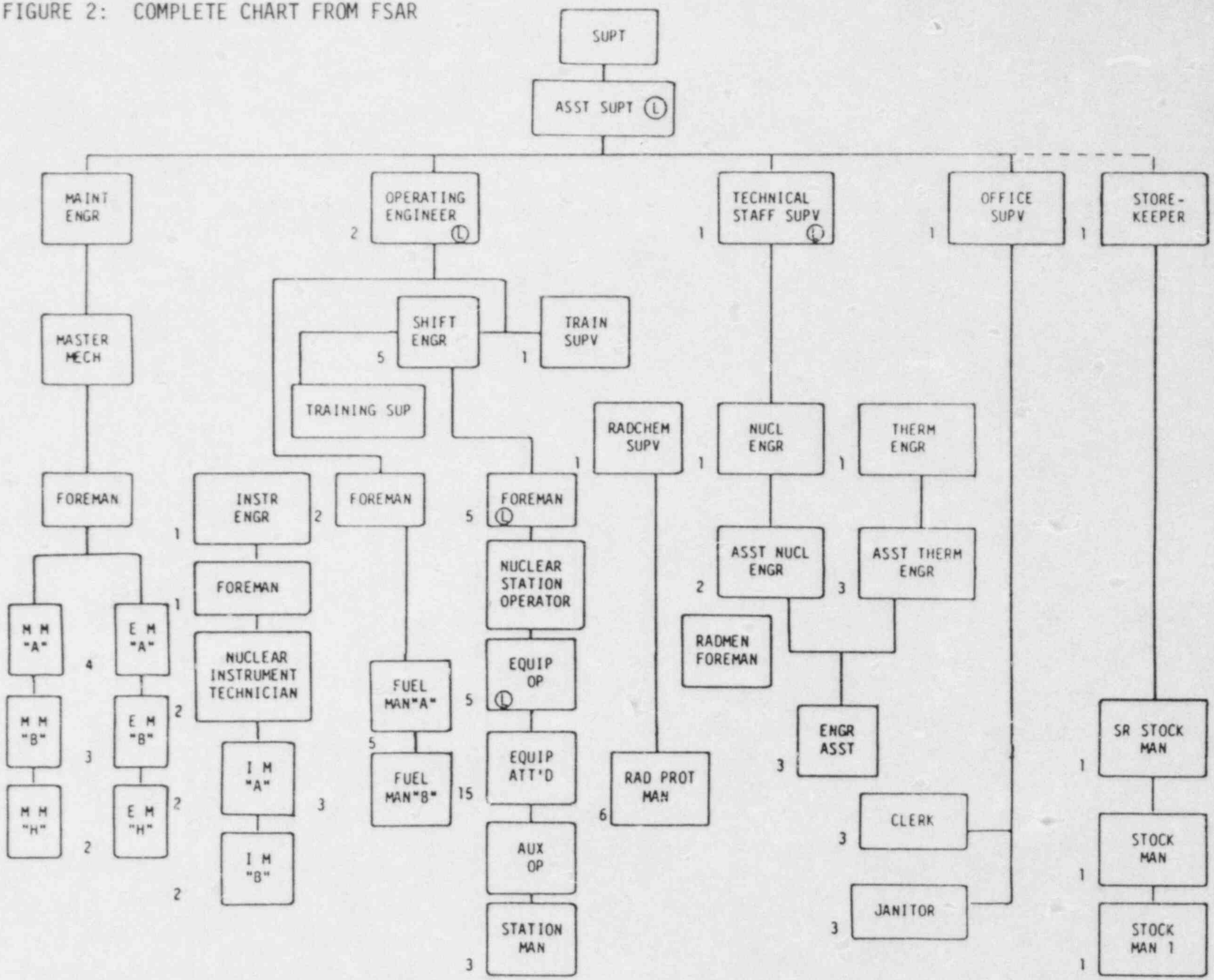
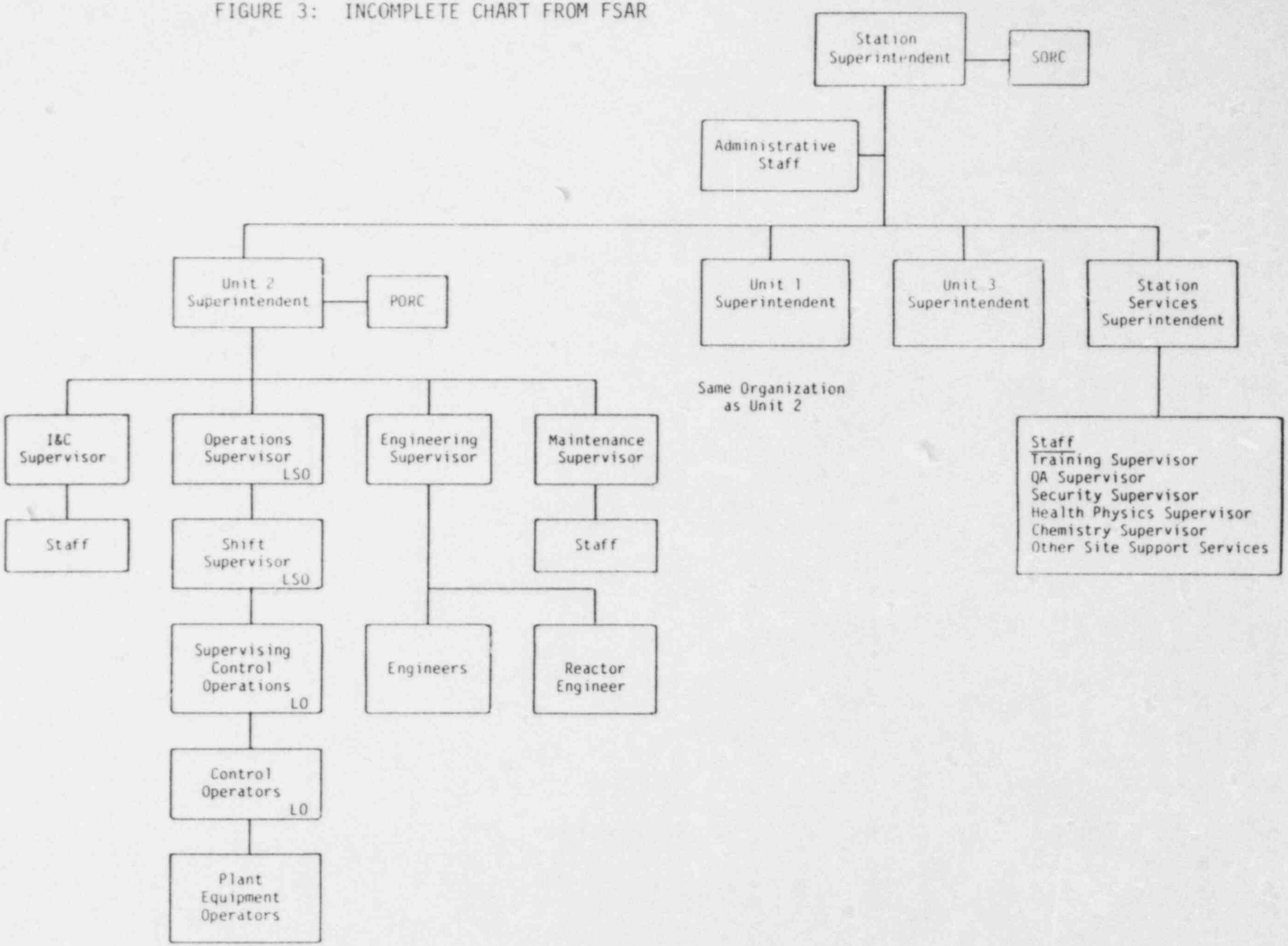


FIGURE 3: INCOMPLETE CHART FROM FSAR



manager's corporate rank, we have measured the number of ranks from the facility manager to the Nuclear Vice President or apparent equivalent.

Supervisory span of control measures, or leader/lead ratios, are the number of people reporting to various types of supervisors. Here the concern is with unit managers, second level supervisors, and third level supervisors. Spans within the line, staff, and intermediate areas were also calculated. Since the same average score could result from widely different patterns of leader deployment, the range of variation in supervisory spans for each unit in question was calculated. This gives a measure of consistency in supervisory patterns across departments and levels.

2.4.2.2 Horizontal Measures

Also of concern is the strategy adopted by the utility for dividing the organization into functional and administrative units. Some organizations may have many small departments, while others may have a few large ones. The strategy employed has implications for NPP safety, since different approaches will lead to different problems of coordination, communication, control, and the like.

The horizontal measures used in this analysis are simply the number of administrative units given by the charts for each organizational level. Specifically, the number of units at the first level, second level, third level, and level directly under the top manager were calculated for both the facility and the operations component.

2.4.2.3 Coordination Measures

Two types of coordination measures were calculated. The first, derived from the organizational charts, refers to the formal mechanism used for linking three important plant functions. Specifically, for the functions of operations, maintenance, and engineering (technical), the level at which the functions meet in a supervisory unit has been recorded. The hypothesis is that functions that converge further down in the organization will be better integrated than functions that converge higher in the organization.

For the three variables defining the point of convergence of operations and maintenance, operations and engineering, and maintenance and operations, scoring is comparatively simple. Plants are given a zero if the point of convergence is outside the plant organization; one, if the point of convergence is at the plant manager, two, if the point of convergence is at the assistant manager; and a three, if the point of convergence is at the superintendent level or below. Thus, the higher the score, the more form is used to facilitate coordination.

A second approach to measuring coordination is through an analysis of the extent to which the organizations employ staff in a given function that are cross-trained in other functions. The hypothesis is that such cross-training results in a better understanding and working relationships across functions and, hence, better safety performance.

The Teknekron survey of technical resources (NUREG/CR-1656) was used to examine cross-training. This survey provides measurement on 1) the number of operations personnel with engineering degrees, 2) the number of maintenance personnel with engineering degrees, 3) the number of maintenance personnel with operations experience, and 4) the number of engineering staff with operations experience. An investigation of the data suggested that plants varied in the extent to which they reported the backgrounds of non-supervisory personnel. Consequently, the measures reflect only the number of supervisory personnel in the plant with each type of cross-training.

2.4.2.4 Summary of Organizational Structure Measures

Table 2 provides the names and definitions of the organizational variables used in the analyses in Section 4 of this report.

2.5 Summary

In this section, the organizational perspective has been defined, key dimensions of organizational structure have been identified, potential data sources for organizational structure were reviewed, and specific measures of vertical, horizontal, and coordinative aspects of structure were calculated. While the available measures are limited to 42 plants and a subset of all relevant dimensions of organizational structure, they will allow for an initial investigation of the relationship between organizational structure and NPP safety performance.

Table 2: ORGANIZATIONAL VARIABLES

Vertical Measure:

Longest Line: Number of ranks in the longest line from first line supervisor to the top of the reactor or facility unit.

Average Line: The average number of ranks across all lines from the first line supervisor to the top of the reactor or facility unit.

Facility Rank: Number of ranks between the facility manager and the Nuclear Vice President.

Manager Span: Number of people reporting to manager of reactor or facility unit.

Third Line Span: Average number of people reporting to third level supervisors.

Second Line Span: Average number of people reporting to second level supervisors.

Line Span: Average supervisory spans in operations.

Intermediate Span: Average supervisory span in maintenance, engineering, radwaste, health physics, and chemistry.

Staff Span: Average supervisory span in staff functions.

Horizontal Measures:

Units under Manager: Number of departments in the reactor or facility unit.

Third Level Units: Number of third level supervisory units.

Second Level Units: Number of second level supervisory units.

First Level Units: Number of first level supervisory units.

Coordination Measures:

Operations/Maintenance: Level at which operations and maintenance functions converge.

Operations/Technical: Level at which operations and engineering functions converge.

Maintenance/Technical: Level at which maintenance and engineering functions converge.

Degreed-Operations: Number of operations supervisory personnel with engineering degrees.

Licensed-Maintenance: Number of maintenance personnel with current or past operator licenses.

Licensed-Technical: Number of engineers with current or past operator licenses.

Degreed-Maintenance: Number of maintenance prsonnel with engineering degrees.

3. CONSTRUCTION OF NPP SAFETY INDICATORS

This section describes the issues associated with the creation of measures of NPP safety and the results of an effort to construct a multiple indicator approach to safety. It is divided into four subsections. The first provides a brief introduction to the logic of the multiple indicator approach. The second describes the search for measures of plant performance that would be useful in the construction of a multiple indicator model of safety and summarizes our evaluation of potential measures. (Previous research employing each potential measure of plant safety has been reviewed in detail in NUREG/CR-3215. This section also outlines procedures used to operationalize the measures for analysis. The third subsection describes sources of bias for each measure and the statistical procedures used to adjust each for these biases.

The final subsection describes the analytic technique used to test a number of hypotheses about the relationship between the adjusted measures and several underlying dimensions of plant safety. The technique is then applied resulting in the identification of four general indicators of plant safety performance.

3.1 The Multiple Indicator Approach

Without describing the technical details of these procedures, the methodology requires that the investigator identify one or more factors underlying the associations among a set of observed measures. These factors, or unmeasured variables, are hypothesized to cause all or part of the association among the measures. Simply put, it is assumed that each measure is an imperfect measure of the underlying concept of interest (safety), but that the correlations among the measures is at least partially due to the fact that they are measuring, albeit imperfectly, the same underlying concept. Patterns of correlation among the measures can be investigated to determine which measures are strongly correlated (i.e., seem to be measuring the same dimension of the concept), less strongly correlated (i.e., seem to be measuring different dimensions of the concept) or are, perhaps, unrelated to the concept. To the extent that the measures are correlated, they can be combined to create more reliable indicators of the underlying concepts. Observed associations among the measures can be due to their mutual relation to safety or such unmeasured sources of error as self-report bias. The confirmatory factor analysis approach used here permits an hypothesized model of factors and indicators to be tested statistically to determine if it is consistent with the data. If it is not, the model can be modified to improve the fit.

The proposed technique can be illustrated under a number of simplifying assumptions. First, let the Licensee Event Report (LER), inspection data, forced outage (operating) data, and Systematic Assessment of Licensee Performance (SALP) Reports serve as the four major sources of information. (These data are described below.) Second, assume for purposes of this illustration, that these four sources are not further differentiated by time, system, cause or any other factor. Third, assume that each measure has been statistically adjusted for sources of bias (e.g., age, plant size). Finally, assume that there are four underlying safety-related factors present in these measures: regulatory compliance, quality, efficiency, and innovation. Again, these are intended to serve as initial hypotheses that can be confirmed or rejected only after the data are analyzed.

One model for stating the relationships among these hypothetical factors and the four measures is portrayed in Figure 4. The four unmeasured safety-related factors are illustrated by the circles. The first, regulatory compliance, is a factor thought to influence variance in three of the four measures: LERs, inspection data, and the I & E survey. This represents the hypothesis that the extent to which a plant conforms to the NRC regulations will be reflected in the number of LERs, the number of inspection findings, and the opinion of I & E staff regarding the plant.

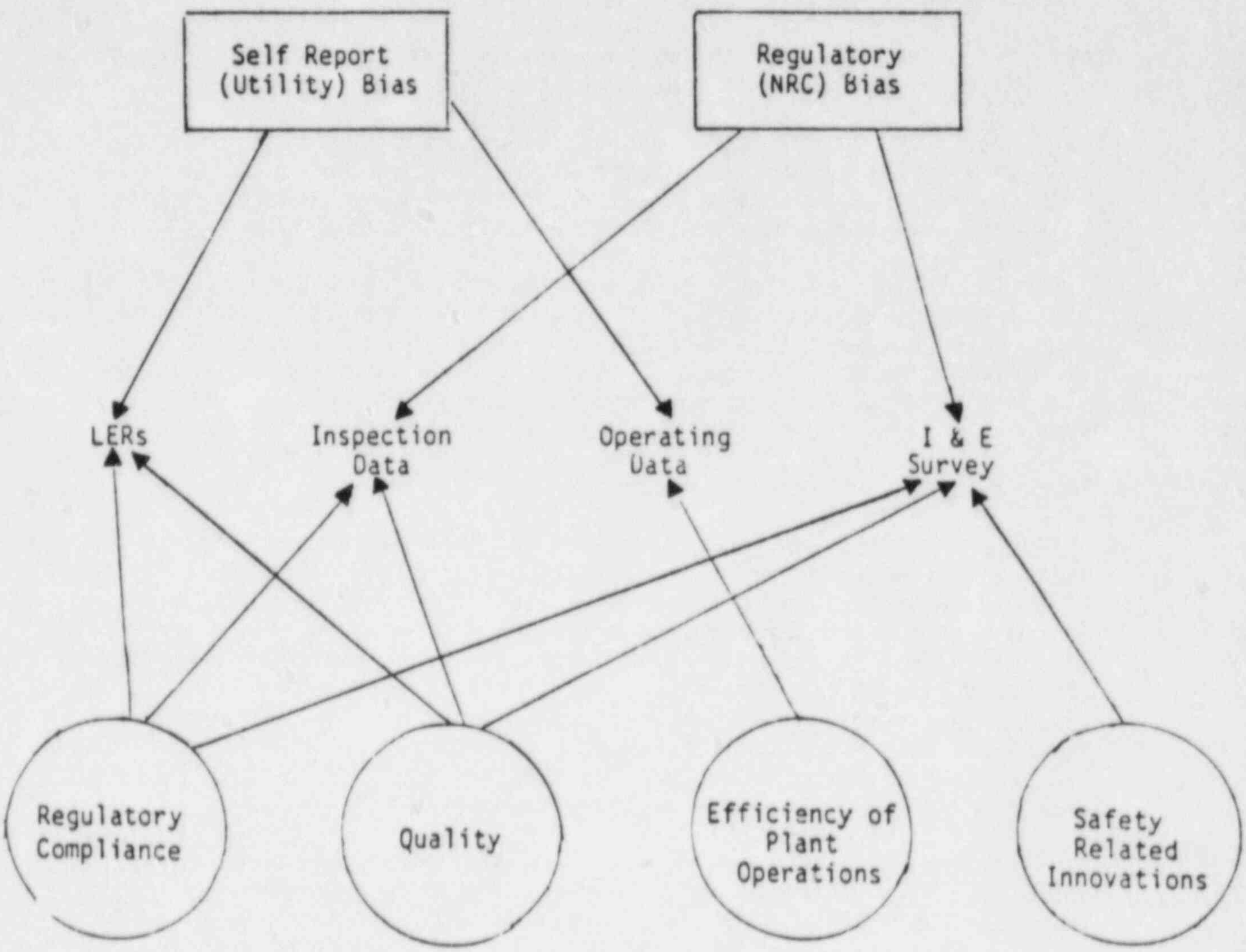
In Figure 4, the four measures are shown to be affected by two "bias" factors, illustrated by the two boxes labelled "Self Report Effects" and "Regulatory (NRC) Effects." These two unmeasured factors are intended to serve as examples of nonsafety-related causes of variation in the measures. For purposes of illustration, the self-report bias is expected to account for variation in LER and forced outage data because they are data sets provided to the NRC by the utilities themselves. Utility practice regarding the filing of LERs and recording outage cause codes may have an effect on the variance of the two measures. Likewise, the two measures with the NRC as the source (inspection and SALP data) may have some association due to the fact that both involve interpretations by NRC inspectors.

Once such a model is defined, the next step is to estimate the parameters and assign a value to each plant on each of the underlying dimensions of safety. The solution to a measurement model such as described in Figure 4 will provide an equation for each of the four factors that can be used to assign each plant a score. This score then becomes the indicator of plant performance on the specific plant safety performance factor. To derive such a solution, data from the four sources selected must be tabulated, residualized (i.e., adjusted to remove variation due to plant characteristics), and factor analyzed. It is, therefore, critical that appropriate data be used for subsequent analyses.

3.2 Plant Safety Performance Measures

This section describes the results of an evaluation of existing plant safety measures to determine their utility in the assessment of the safety implications of plant organizational structure. As a first step in this effort, attention was focused on existing (primarily public) data in order to determine whether adequate indicators of one or more dimensions of plant safety could be developed. Measures from ten data sources were

FIGURE 4: HYPOTHETICAL MODEL FOR MULTIPLE
INDICATOR APPROACHES TO THE MEASUREMENT
OF PLANT SAFETY



considered: LERs, Inspection and Enforcement Data, Forced Outage Data, SALP Reports, I & E Staff Survey, Personnel Exposure Data, Effluent Release Data, Operator Exam Scores, Liability Insurance Risk Assessments, and INPO Plant Evaluations. Each of the potential measures was evaluated on the basis of the following five criteria:

(1) the extent to which the measure was reliable (e.g. repeated measurements would produce the same results);

(2) the extent to which the measure was readily available in a form suitable for analysis;

(3) the extent to which the measure was valid (i.e., a specific dimension of plant safety is actually being measured);

(4) the extent to which the measure was objective (i.e., not affected by bias from differential interpretation and/or perception);

(5) the extent to which the measure is conceptually distinct from the predictor variables (indicators of safety that conceptually overlap the measured aspects of plant organization will lead to tautological arguments and, therefore, must be avoided).

Figure 5 provides a brief overview of the evaluation of the measures along with a summary of the conclusions regarding each measure's standing on each of these five criteria. It is important to note that these evaluations concern the use of the measure as a source of plant safety information in a statistical analysis. None of the measures were specifically developed for that purpose. Consequently, the following evaluation says little about the measure's ability to perform the function for which it was originally designed.

A five-point scale was used to reflect the judgments regarding each measure's standing on the five evaluation criteria. A rating of 1 in Figure 5 means that the measure is judged to be completely acceptable in its current form on the relevant criterion. A rating of 5 represents the conclusion that the measure is unacceptable on the criterion even after all feasible and realistic adjustments. Values 2 through 4 represent intermediate levels of acceptability; a 2 means that the measure is acceptable with minor modifications; a 3 means the measure has significant weaknesses but may be adequate with adjustments; and a 4 means the measure is only marginally acceptable with major modifications.

The judgments summarized in Figure 5 are discussed in detail in NUREG/CR-3215. Several general comments should be made here. First, the summary column indicates that no single measure is considered adequate by itself. This should not be interpreted to mean that no empirical indicators of plant safety can be developed. As will be demonstrated in subsequent sections, the weaknesses of some measures are strengths of

FIGURE 5: SUMMARY OF EVALUATIONS OF PLANT SAFETY INDICATORS

Indicators	Reliability	Availability	Validity	Objectivity	Conceptual Distinctness	Summary Evaluation
Licensee Event Reports	2	2	3	4	3	3
Inspection & Enforcement Data	1	1	3	3	2	2
Operating & Outage Data	1	1	3	2	3	2
Systematic Assessment of Licensee Performance (SALP)	3	2	3	3	3	3
I & E Staff Survey	3	5	3	4	2	3
Personnel Exposure Data	1	1	3	3	3	3
Effluent Release Data	2	2	2	2	2	2
Operator Exam Scores	3	3	4	3	5	5
Liability Insurance Risk Assessments	3	5	3	3	3	5
INPO Plant Evaluations	4	3	4	4	5	5

The values in Figure 5 represent evaluations of the indicator on each factor and an overall summary evaluation. A "1" represents the best evaluation and a "5" represents the worst. See text for definitions.

other measures which raises the possiblity of a multiple indicator approach.

Three measures were assigned an "unacceptable" summary evaluation: operator exam scores, liability insurance data, and Institute of Nuclear Power Operations (INPO) plant evaluations. The "unacceptable" evaluation was due primarily to the nonavailability of the data. In addition, the personnel exposure and effluent release data are not given lengthy treatment because non-trivial variations in either exposure or release data are recorded in Licensee Event Reports (LERs), or in Inspection and Enforcement Reports. These measures, therefore, are redundant. The measures receiving major attention in this report are LERs, Inspection and Enforcement Reports, forced outage data, and SALP reports. Following a brief discussion of each of the measures, a procedure for combining them to create indicators of plant safety is described.

3.2.1 LERs

The most commonly used source of information regarding plant performance is the Licensee Event Report. Utilities submit LERs to the NRC in compliance with the plant's Technical Specifications and in compliance with NRC regulations. Approximately 5,000 LERs are currently reported each year by power plant licensees. Most LERs are based on violations of technical specifications, but others reflect an event that may have potential public interest. The primary reference concerning LER requirements is "Regulatory Guide 1.16, Reporting and Operating Information -- Appendix A Technical Specifications."

There are two separate categories of LERs: those requiring 24 hour notice and 14 day follow-up (two week LERs), and those requiring a 30 day written report only. These two LER types are meant to distinguish between those events with major safety significance versus those with lesser significance. About 20% of all LERs are the more significant, two-week type.

An apparent distinction between the two LER types is that the events requiring more immediate reporting are those involving a safety-related system becoming nonoperational and/or a condition requiring plant shutdown. Events requiring a report within 30 days appear to be those in which the event causes less than optimal operation of a system for which a redundant system could be substituted.

The analysis of LER data has been confined to two-week LERs because they represent more serious threats to safety and plant personnel, and are of greater immediate significance to the NRC. A focus on two-week LERs also helps to minimize utility-to-utility variation in reporting philosophy and practice.

LER data are readily available from a variety of public sources. The Atomic Industrial Forum Newsletter (AIFN) was selected as the LER data source in this study. It is published biweekly and identifies two-week LERs for all operational plants by system and by cause. Moreover, the AIFN records two-week LERs on the date they occur rather than the date they are reported to the NRC. All two-week LERs for events taking place in the calendar year 1981 were tabulated and listed by the following causes:

(1) component failure
(2) design failure
(3) personnel error
(4) other sources of system failure
(5) procedural error

The 1981 two-week LER data for 67 plants were aggregated across cause codes forming two general categories: human error and hardware failure. All two-week LERs attributable personnel error and procedural error were summed to form the human error variable (HPLER). The second variable, hardware failure, is represented by the sum of the two week LERs in the remaining three cause codes.

During 1981, a total of 108 two-week human error LERs (HPLERs) were reported for the 67 plants. The number of HPLERs ranged from none for twenty-four plants to six for Arkansas-1 and Beaver Valley-1. The mean number of HPLERs for the 67 plants was 1.6 with a standard deviation of 1.7. The distribution of HPLERs is skewed to the right with the modal value at zero and the median at 1.2.

The distribution of hardware failure LERs (OLERs) is more varied. During 1981, a total of 361 two-week OLERs were reported for the 67 plants. The average number of OLERs was 5.4 with a standard deviation of 5.0. Only five plants (Ginna, Surry-2, Cooper, Trojan-1, and Arkansas-2) reported no OLERs for 1981. At the upper end of the distribution of OLERs, San Onofre-1 reported 15 OLERs, Pilgrim-1 reported 17 OLERs, Salem-1 reported 19, and Hatch-1 reported 22 OLERs.

3.2.2 Inspection and Enforcement Data (766 File)

NRC's Office of Inspection and Enforcement (I & E) provides another data set relevant to plant performance in terms of compliance with regulations. Information generated from inspection and enforcement activities, all of which utilize Form 766 are entered into a "766 Computer File" which is managed by the NRC. On the average, there will be between two and five thousand entries to the 766 file for any given year (the number of entries has increased dramatically during recent years) which reflect violations of approved operating practices and parameters.

The 766 File data used in this study were obtained directly from the NRC. At Battelle's request the NRC provided a printout of 766 file data for operational plants containing the frequencies of plant violations by quarter covering the period 1 July, 1980 to 31 March, 1983. The data also provide the distribution of quarterly violations by severity code for each plant. The data used in the present analysis are for the 70 plants studied and from the four quarters of 1981. The six severity codes range from I, most severe, to VI, least severe. There were no infractions for level I and very few for levels II and III. Hence, levels II and III were combined to form one severity level. Levels IV, V, and VI were each used as separate measures.

For the 70 plants analyzed, there were a total of 92 level II and III violations. Most plants (41) had no level II or level III violations while 12 plants had only one. Browns Ferry-1, Browns Ferry-2, Browns Ferry-3, and Trojan-1 each had seven level II and III violations while Cook-1 and Cook-2 had eight and ten violations respectively. As a whole, the average number of 1981 level II and III violations was 1.3 with a standard deviation of 2.3. The distribution of II and III violations is skewed to the right as 53 plants had zero or one such violation. The median number of violations is .35 with the mode at zero.

3.2.3 Operating and Outage Data

All nuclear utilities are required by the NRC to provide monthly operating reports to the Office of Management and Program Analysis These data are provided monthly by the NRC in NUREG-0020 referred to as the Greybook.

Coded data from the monthly operating reports include whether or not a shutdown was forced or scheduled, eight different reasons for the shutdown, and six different methods by which the shutdown was accomplished. Coded reasons for each shutdown are:

(1) equipment failure
(2) maintenance or test
(3) refueling
(4) regulatory restriction
(5) operator training and license examination
(6) administrative
(7) operator error
(8) other

Coded methods by which the shutdown was accomplished are:

(1) manual
(2) manual scram
(3) auto scram
(4) continued
(5) reduced load
(6) other

In essence, there are two reasons for which plants are shutdown, regardless of method. The first, scheduled shutdown, refers to routine periods of maintenance or refueling operations. Generally, these periods are arranged before the plant is taken off line. The second type, forced outage, occurs due to equipment failure, operator error, or some other unanticipated event. In this study, forced outages are used as an indicator of reduced safety since they represent unexpected events causing the reactor to scram or to require a reduction in power.

As with LERs, two types of forced outages can be distinguished. The first are those that result from equipment failure or some other unanticipated factor which forces the plant to shut down. The second is attributable to personnel error. The data for these two types of variables were obtained from the February 1981 to January 1982 NUREG 0020 monthly reports (Vol. 5, No. 2-12, Vol. 6, No. 1). The frequencies of forced outages for 67 operational plants in 1981 were tabulated and aggregated to yearly totals. The yearly forced outage frequencies were grouped according to three cause variables: equipment failure (FA), personnel error (FG), and other causes (FO). For these 67 operational plants there was a total of 1004 forced outages: 61% were due to component failure, 7% were due to personnel error, and 32% were due to other causes.

Forced outages due to component failure was the major cause of forced outages in 1981. For the 67 operational plants, there was an average of 9.2 component failure outages per plant. Only one plant reported zero forced outages due to component failure (Point Beach-2) while Robinson-2 reported 25, Hatch-1 and Hatch-2 reported 34 and 41 respectively, and Salem-1 reported 63 component failure outages. For outages due to operator error, there was a total of 69 forced outages with a mean of 1.0 per plant and a standard deviation of 1.8. During 1981, 17 plants reported only one operator error outage while North Anna-2 reported seven and Trojan-1 and Arkansas-2 each reported eight.

Forced outages due to other causes totalled 320 occurrences with a mean of 4.8, and a standard deviation of 5.6. While most plants reported less than 3 forced outages of this type, Fort St. Vrain, Hatch-1, and Hatch-2 reported 18, 25 and 26 respectively.

To simplify analyses, the frequencies of component failure and forced outages due to other causes were combined. This permits the study to contrast human error outages with all other causes of forced outages. The combined category of forced outages, then, totaled 935 outages with a mean of 14.0 per plant and a standard deviation of 13.8.

It is important to note that comparing forced outages across plants by examining only the frequency of forced outages confounds plant reliability with the amount of time the plant was actually operational or was shut down due to a scheduled outage. Plants are at risk of forced outages only if they are online and not shut down due to a scheduled or NRC required outage. Therefore, all frequencies of forced outages, regardless of cause, were adjusted to reflect the period of time they were actually online (i.e., at risk of a forced outage).

This adjusting procedure can be expressed as follows:

$$OUTFG = \frac{1}{ONLINE + FHRS} \times FG$$

where OUTFG is the corrected factor of forced outage due to operator error, FG is the total number of operator error forced outages, ONLINE is the total hours the plant was generating power during 1981, and FHRS is the total hours during 1981 the plant was shutdown due to all forms of forced outages. By substituting the variable FAO (FA + FO) for FG, the corrected factor of forced outage due to equipment failure and all other causes of forced outages (OUTFAO) can be computed. The behavior of the reciprocal of (ONLINE + FHRS) facilitates the comparison of plants having the same frequencies of forced outages but also having different risks for being forced down. For example, both Surry-1 and Indian Point-3 had 13 forced outages due to non human causes. However, Indian Point-3 was at risk, or potentially online, for 8305.8 hours whereas Surry-1 was at risk to forced outages for 3864.4 hours. Hence, the adjusted rates for Surry-1 and Indian Point-3 are .00336 and .00156 respectively. In other words, Surry-1's rate is 2.15 times that of Indian Point-3 as it had the same number of forced outages in 46.5% of the potential operating time.

This adjustment procedure was used for the two general types of forced outages. The average rate for forced outages due to non-human error is .002 outages per operating hour with a standard deviation of .002. Plant rates range from zero for Point Beach-2 to .011 for Hatch-1. The mean rate for adjusted forced outages due to human error was .0001 outages per potential operating hour with a standard deviation of .0003. The highest rates were .001 for Arkansas-2, Trojan-1, North Anna-2, and Peach Bottom-3.

3.2.4 Systematic Assessment of Licensee Performance (SALP)

The NRC conducts annual reviews of the performance of all commercial nuclear power plants that are either operating or under construction. These reviews are conducted under the SALP. The objectives of this program are to improve licensee performance and assist the NRC in allocating resources for regulatory efforts. The NRC requires that each plant be evaluated annually. Plants are evaluated on the basis of functional areas by the following criteria:

(1) Management involvement in assuring quality
(2) Approach to resolution of technical issues from safety standpoint
(3) Responsiveness to NRC initiatives
(4) Enforcement history
(5) Reporting and analysis of reportable events
(6) Staffing (including management)
(7) Training effectiveness and qualification

These criteria are then used to assign each of several functional areas into one of three evaluation ratings: (1) reduced NRC attention may be appropriate; (2) NRC attention should be maintained at normal levels, or

(3) NRC and licensee attention should be increased. The SALP data used in this study were obtained from the NRC copies of SALP reports.

There are several problems with SALP reports[3]. First, while all plants are to receive annual review covering a 12 month period, current SALP report coverage extends over periods ranging from 12 to 18 months. Second, SALP reports are made at the facility level of analysis combining ratings for the individual plants at a single site. Third, 1980-81 SALP reports have yet to be filed for all plants. During the period betwen 1980 and 1982, some plants have been evaluated annually while others have only one report. Fourth, the number of functional areas, as well as the definitions of the functional areas, vary across SALP reports. Some reports review 14 functional areas, others review only nine. Moreover, the number and type of functional areas varies annually for the same plant. For the nine sites where SALP reports were available for both 1980-81 and 1981-82, the number of functional areas evaluated dropped from 14 for 1980-81 to a range of 12 to nine for 1981-82. A review of the most recent SALP reports for each of 36 sites (49 plants) yielded the following 18 different functional areas:

- *(1) Plant operations
- *(2) Refueling operations
- *(3) Maintenance
- *(4) Surveillance and testing
- (5) Personnel, training and procedures
- *(6) Fire protection and housekeeping
- (7) Design changes and modifications
- *(8) Radiological control
- *(9) Environmental controls
- *(10) Emergency preparedness
- *(11) Security and safeguards
- (12) QA audit, review and committee activity
- (13) Administration, QA, records, and procurement
- (14) Corrective action and reporting
- (15) Licensing activities
- (16) Confirmatory measures and environmental monitoring
- (17) Quality programs
- (18) Three Mile Island (TMI) actions

Of these 18 functional areas, only those areas of the above list which are starred were both defined in the same way and evaluated for each of the 36 SALP reports. Hence, only these functional areas are comparable across the sites. After careful consideration, it was determined that the functional areas most relevant to plant safety were the first four listed above. The mean score for these four areas became the basis for the SALP variable used in this study.

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[3]SALP reporting and analysis procedures have become more systematic since the 1980-1981 time frame for most of these data.

Despite the fact that SALP reports are given for sites, the basis of the evaluation is the plant. As a result, SALP data can be treated as an average of all plants at a given site under the same plant management. By assigning all plants located at the same site, with the same plant management, the same categorical evaluation (i.e., their site specific mean), the SALP data can be evaluated at the plant level of analysis. While this limits the possible variation in functional area rankings, it does permit an analysis of the general NRC evaluation of a group of plants. In all, 49 plants were assigned SALP scores.

For four plants, NRC evaluators noted that an insufficient period of inspection of refueling operations meant that this area could not be evaluated. Hence, for these four plants, the mean value of the three evaluated functional areas was computed instead.

3.3 Adjustments to the Raw Data

The measures described above are known to be influenced by factors other than plant safety. These "non-safety related" factors can be divided into two broad categories. The first category pertains to the method used to report the particular event type. Here an important distinction is between self report data (i.e., the utilities themselves provide the reports for LERs and forced outages) and NRC generated data (i.e., 766 and SALP). The second refers to physical characteristics of the operational plants themselves that can potentially influence the actual frequencies of events (e.g., see the above discussion on forced outages). Plant differences in age, size, vendor and pressurized water reactor (PWR) versus boiling water reactor (BWR) are appropriate. These two categories of variation or "bias" in the measures should be removed when the purpose is to measure safety-related plant performance.

In order to remove these effects each measure was statistically adjusted to control for plant differences in variables thought to affect the indicators. These plant characteristics are age, size, region, type, and vendor. Each of these is discussed in more detail below.

3.3.1 Plant Size

The most direct measure of plant size is the amount of electrical energy capable of being generated. Variation of electrical energy capacity directly affects the technical specifications of the plant which in turn affects the reporting of safety related events. Hence, the net megawattage per hour of electrical energy specifications of a given plant was used to indicate plant size. This variable (SIZE) was measured in megawatts per hour and obtained from Greybook summary statistics. The average size for the 70 plants used in this study is 736.9, with a standard deviation of 253.4. The size of plants ranges from 50 for LaCrosse to 1130 for Trojan-1.

3.3.2 Plant Age

As of January 1, 1981 the mean plant age was 81.3 months with a standard deviation of 41.7 months. Only seven plants had been operational for less than 36 months. Of the remaining plants, four had been operational for greater than 150 months.

The literature on plant performance suggests that there is a positive non-linear relation between plant age and safe operation. During their first three years of operation, plants may experience a substantial number of problems and complications and, hence, large numbers of LERs and 766 violations. Therefore, the age of the plant has been controlled for by two separate variables. The first variable, MOSCRIT, measures the number of months up to three years since the plant became operational. The second variable, MOSPOST, records the balance of monthly age for plants older than three years. For example, if a plant was six and a half years old (78 months) at the beginning of 1981, MOSCRIT has a value of 36 and MOSPOST has a value of 42. If a second plant had only been in operation for a year and a quarter (15 months), then MOSCRIT equals 15 and MOSPOST equals 0. Using both age variables in the statistical adjustment procedures permits the non-linear form of the relationship of age to performance to be taken into account.

3.3.3 Type (PWR vs. BWR) and Vendor

In order to control for vendor and reactor type, three dummy variables were constructed to reflect a composite of vendor and type. (A dummy variable takes on a value of one if a characteristic is present and a score of zero if it is not.) The first dummy variable, PWRW, has a value of 1 for Westinghouse supplied PWR reactors and 0 for all others. The second dummy variable, PWROTH, has a value of 1 for Combustion Engineering (CE) and Babcock and Wilcox (B & W) supplied PWR reactors. For all others vendor/types, the value is 0. The third dummy variable, BWR, has a value of 1 for BWR reactors of all types supplied by all vendors and 0 for all others. Altogether, 26 plants were classified as BWR, 26 plants as PWRW and 17 plants as PWROTH.

3.3.4 Region

Regional location for all plants was obtained from the World List of Nuclear Power Plants (Nuclear News, June, 1981). Four dummy variables were constructed to reflect the northeastern (NOREAST), midwestern (MIDWEST), southern and southwestern (SOUTH), and western and northwestern (WEST) regions of the U.S. For the 70 plants studied, 21 were in the NOREAST regional category, 22 in the MIDWEST category, 22 in the SOUTH, and five in the WEST. (The four regional categories are equivalent to NRC regions with the exception that the few plants in this sample from Region 4 have been combined with the closest other NRC region.)

3.3.5 The Statistical Adjustment of the Safety-Related Indicators

The procedure utilized to remove variation in the safety measures caused by these control variables is known as "residualization." Essentially, a residualized score is the difference between a plant's actual score and the score predicted by a regression analysis of the plant's characteristics. Here, the number of LERs, I & E violations, forced outages, and average SALP scores that would be expected to occur given the plant's value on each of the control variables (region, age, size, reactor type and vendor) was estimated. The second task is to calculate the difference between the actual frequencies of the safety measures and predicted frequencies estimated from the control variables. This difference, or residual, represents the degree of variation in the safety measures unexplained by the control variables. Hence, the residualized safety variables are interpreted as deviations from what would be expected of a plant due to its size, age, reactor type, vendor, and location.

As noted above, the first step in the residualization procedure is to estimate the effects of the control variables on the safety measures. A by-product of this adjustment procedure is the estimate of which, if any of the control variables is a significant predictor of the performance measures. Thus, it is possible to see which control variables are more important in explaining safety measures and the direction of their impact on each of the measures. The details of this analysis are reported in Appendix A. A brief summary of the results and implications is warranted here.

Overall, the results of the regression analyses between the safety measures and the control variables suggest that few of the control variables are important. Physical plant characteristics such as age, reactor type, vendor, and reactor size proved to be weakly related to variations in the frequencies and rates of the safety measures. Regional location, however, emerged surprisingly as the strongest predictor of variation in the measures. Region was found to be important for I & E severity levels II and III, IV, and V; it was also the strongest predictor variable for average SALP score and nonhuman caused forced outages. While no explanation is offered here of why regional location would be such a strong predictor of variation in the safety measures, further analysis of this relationship is suggested.

The subsequent confirmatory factor analyses detailed in the next section were based on the residualized scores. Plant specific values for each of the control variables were substituted into the general equations and an estimated value for each safety variable calculated. The difference between the observed and the estimated value is the residualized safety value for each plant. Using the residualized safety variable, the subsequent analyses were able to generate unbiased estimates of the underlying safety indicators and assign specific scores to each plant for these indicators.

3.4 The Estimation of the Multiple Indicator Model

The statistical procedure utilized to develop and estimate a multiple indicator model of plant performance was a form of factor analysis. Factor analysis is a widely used technicque for identifying the underlying dimensions of a correlation matrix. If the correlations among variables in a matrix are partially due to the fact that they are measures of the same thing (i.e., there are multiple indicators), then factor analysis can simplify the matrix by identifying the dimensions (factors) underlying the matrix.

The most common forms of factor analysis are "exploratory" in that the statistical procedure requires very little direction from the investigator. The correlation matrix is examined and factors are created according to statistical criteria rather than the substantive meaning of the variables.

In contrast to the standard exploratory factor analysis techniques, the analysis reported here is based on "confirmatory factor analysis." This is an hypothesis testing technique in which the investigator poses a number of hypotheses regarding the factors expected to exist and the specific variables thought to be indicators of each factor. Together, these hypotheses regarding factors and their indicators form the multiple indicator model. The confirmatory factor analysis procedure imposes these hypotheses on the data and generates maximum likelihood estimates for the parameters. The procedure then calculates a chi-square goodness of fit test to provide an indication of the extent to which the imposed model "fits" the data. By altering the hypotheses, and thereby making changes to the model, it is possible to compare the chi-square value to determine if a given model change improves or reduces the fit of the model to the data.

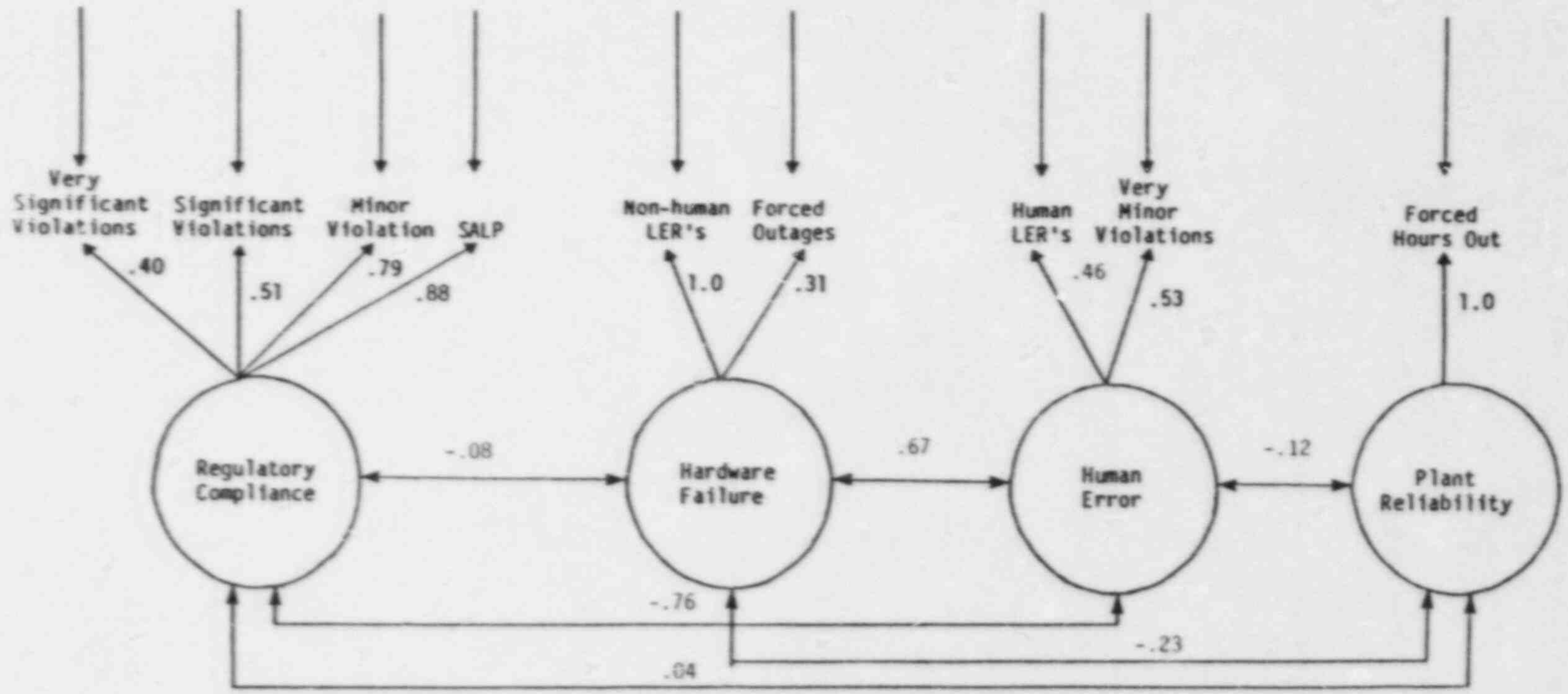
Technical discussion of the basis for exploratory factor analysis can be found in Harmon (1967). Confirmatory factor analysis is described in detail in Joreskog (1970, 1973), and Joreskog and Sorborn (1978).

The results of the confirmatory factor analysis approach are displayed in Figure 6. This model is the result of an iterative process in which a large number of alternative solutions or models were tested. The model shown is the "best fitting" model.

Before describing these results, a general comment should be made. The measure of forced outages due to human error does not appear in Figure 6. This indicator was found to be negatively associated with human error LERs as well as other indicators with which its association was expected to be positive. Several efforts to explain the findings and to incorporate the indicators in the model failed. Consequently, it was omitted from the analysis at this stage of the effort and will be given more attention in subsequent efforts. It should also be noted that we have not assumed any causal ordering <u>among</u> the safety measures.

FIGURE 6

THE BEST FITTING FACTOR ANALYSIS SOLUTION



As shown in Figure 6, there are four underlying dimensions of safety related plant performance. Regulatory Compliance is the indicator measured by adjusted SALP scores and adjusted violations for all but the least severe violations. This indicator represents the extent to which the plant conforms to NRC regulations (lesser violations for all but the least severe) and higher evaluations of plant performance by NRC staff (SALP scores). Hardware Failure is an indicator measured by non-human LERS and forced outages. It represents the extent to which the plant experienced problems related to the failure of hardware. The third indicator, Human Error, is measured by human LERs and the least severe violations. This factor represents the extent to which the plant experienced problems due to human error. The last indicator, Plant Reliability, has only one measure, so the measure and the indicator are identical. Plant reliability represents the extent to which the plant was generating power during 1981 after adjusting for scheduled outages.

On the arrows linking the indicators are zero-order correlations (See Figure 6). They conform to initial expectations in terms of both direction (positive versus negative) and magnitude. Regulatory Compliance is related to Human Error, in that plants scoring lower on the Human Error indicator score higher on Regulatory Compliance. Neither Hardware Failure or Plant Reliability is related to Regulatory Compliance. Hardware Failure is related to Human Error (the more Human Error the more Hardware Failure) and is moderately related to Reliability (the more Hardware Failure, the less reliable the plant). Human Error is related to Plant Reliability in a similar fashion.

The parameters linking the indicators to the measures represent the relative importance of each measure to the indicator. Thus, the non-human LERs are the most important contributor to the Hardware Failure indicator, and Very Significant Violations is the least important contribution to the Regulatory Compliance indicator.

With the parameters linking measures to indicators, it is possible to estimate a regression equation that will generate a predicted value for each plant on each of the indicators. This regression equation contains coefficients for each measure and a constant term. By multiplying each plant's score on the adjusted measure by the appropriate coefficient, summing the results and adding the constant term, it is possible to create a value for each plant on each of the four indicators. This procedure was performed and the results added to the data set. The resulting values are used as the safety-related plant perfomance indicators in the management and organization analysis described in the next section of this report.

4. ANALYSIS OF ORGANIZATIONAL FACTORS

In this section we report on the analysis of the relationship between organizational structure and the plant safety performance indicators. The first task is to describe the organizations which constitute the sample, both in terms of the representativeness of the sample and in terms of the distribution of the plant scores on the dimensions of organizational structure. The second task is to correlate the organizational indicators with the plant safety performance indicators. The third task is to combine the organizational indicators in a multivariate analysis of the relationship between organizational structure and NPP safety.

4.1 The Industry Context

Even though data are limited, it is possible to begin charting a profile of nuclear plants. Two units of analysis have been selected for description. The first is labeled facility and consists of those organizational entities under the facility manager. Where utilities have two reactor-turbine units, there are separate scores for each. The second unit of analysis is labeled reactor and is comprised of the organizational entities under the operations manager[4]. This unit of analysis is an attempt to capture the organization and administration of those directly involved with operating the plant as defined by the license holder.

4.1.1 Representativeness of the Sample

The sample plants for the current analysis are not a random sample of all plants operating in 1981. A census of all plants was expected from the search of regulatory documents and related material. Unfortunately, there was insufficient information in many cases. This raises the question, "how representative is the sample?" It is not possible to indicate whether the sample represents the universe of plants for organization and administration concerns. It is, however, possible to note representativeness on the basis of a number of environmental (e.g. region) and contextual (technology and size) conditions.

Table 3 compares the basic sample of 42 plants with the 70 initially considered for analysis. (For some analyses the number of plants is less, while in a few instances data are available for over fifty of the plants.)

In general, the data on Table 3 suggest that there is comparatively little bias by type of reactor or region. Facilities with three or more reactors are underrepresented. As for size, the sample and the population reflect

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[4]The operations manager is defined as the highest ranking manager with line responsibility for operations but who does not have responsibility for the entire plant. Consequently, the scope of this unit varies significantly across plants.

TABLE 3: A COMPARISON OF THE SAMPLE AND ALL OPERATING PLANTS

	All Plants 1981 (N = 70)	Sample Plants 1981 (N = 42)
Type of Reactor		
Boiling Water Reactor	37%	37%
Westinghouse PWR	37%	42%
Other Pressurized Water Reactors	24%	21%
Region		
Northeast	30%	36%
Midwest	31%	33%
West	7%	7%
South	31%	24%
Number of Plants At A Given Location		
One Reactor	44%	52%
Two Reactors	43%	47%
Three or More	13%	0%
Net Megawattage		
under 600	26%	26%
over 600	74%	74%

equal ratios of large to small reactors. Thus, we conclude that the sample likely represents the population of reactors in 1981 in terms of reactor type, region, and size.

4.1.2 Organizational Patterns Across the Sample Plants

Using the descriptive statistics in Table 4, it is possible to derive an overall picture of the formal organization strategies employed in the sample. It is also particularly interesting to note areas where there is wide variation within the industry in organizational structure.

Four issues are examined. First, what is the breadth of the formal organization? How many separate units are there? Second, what is the depth of the formal organization? How many ranks or levels of management are employed? Third, how do the utilities organize managers in relation to workers? What is the leader/lead ratio or span of control? Fourth, what is the internal consistency in the deployment of leaders and followers? Are line and staff treated the same way? Appendix B contains a detailed discussion of each variable. Table 4 provides the means and standard deviations for each variable. A brief discussion is warranted here.

The descriptive data, when analyzed as a set, suggest an interesting series of patterns for formal organizaton. The reactor operations parts of the plant appear narrow and deep and there is relatively little variation across the sample. Leader/lead ratios suggest comparatively few employees per manager, particularly for operations. In contrast, the formal organization for staff components appears broader, not as deep, and with substantial variation across the sample. Further, there also appears to be a strategic design decision regarding the placement of staff units within the operations area. While the typical plant does not contain staff components within reactor operation, a few plants do.

There also appear to be major differences across the sample plants in the number of individuals placed within either the tall, narrow structure of the line components, or the broader structure of the staff components. This partially accounts for the substantial variation in leader/lead ratios.

The data are consistent with the following dominant profile. It appears that the reactor components (operations) are bureaucratically oriented with comparatively few administrative units to levels (tall and narrow). In slight contrast, staff units appear somewhat less bureaucratically structured with more administrative units to levels (comparatively flat and broad).

When comparing the facility as a whole with the reactor operations area, it is quite obvious there is more consistency in the formal structure for operations. Standard deviations are considerably smaller. It is also quite obvious that one of two patterns is selected in the operations

TABLE 4: ORGANIZATION INDICATORS: MEANS AND STANDARD DEVIATIONS

	Mean $\bar{X}$	Standard Deviation
Division of Labor		
Average Number of Ranks	3.4	.5
Number of First Level Units	14.1	3.1
Number of Second Level Units	9.0	6.7
Number of Third Level Units	4.1	2.0
Control		
Number of Units Under the Facility Manager	4.2	1.3
Second Level Average Span	3.4	1.1
Third Level Average Span	2.4	.9
Average Span for Line Units	5.4	1.6
Average Span for Staff Units	7.3	3.2
Average Span for Intermediate Units	8.1	4.2
Consistency		
Range in Second Level Span	6.8	3.1
Range in Third Level Span	2.8	2.0
Range in Average Span for Line Units	6.9	3.3
Range in Average Span for Staff Units	12.4	7.3
Range in Average Span for Intermediate Units	17.3	6.7
Coordination		
Number in Operations with a Degree	1.7	2.5
Number in Maintenance with a Degree	2.4	3.2
Number in Engineering with Operating Licenses	2.3	3.7
Number in Maintenance with Operating Licenses	.4	.5
Level at which Operations and Maintenance Meet	1.7	.9
Level at which Operations and Engineering Meet	1.4	.5
Level at which Engineering and Maintenance Meet	1.2	.5

area. Either staff are included or they are not. This strategic choice accounts for the fact that mean values for the reactor operations area statistics are generally higher than modal responses.

The pattern suggested above is quite unusual for a highly automated, continuous process manufacturing operations staffed by professionals and highly trained technicians (e.g. chemical factory). One would have expected more units (particularly more first level units), fewer levels of command, broader spans of control, and less diversity in the treatment of line and staff units. The pattern is more similar to large-scale, quasi-automated batch processing organizations (e.g. auto manufacturing). Further, it is quite interesting to note that the breadth of the structure and its depth do not appear sensitive to an apparent increase in the number of personnel. That is, breadth and depth were similar even where there was wide variation in leader/lead ratios. This suggests that some aspects of the formal structure employed to organize the work are not particularly sensitive to increases in staff size. Unfortunately, staff size is not available to directly examine this potential condition.

4.2 Analysis of Organization and Safety Performance

In this section we analyze the relationships between the available measures of organization, administration, and perfomance and the four safety indicators. First, which organizational variables, taken one at a time are related to performance? Here, bivariate relationships using correlation analysis are examined. Second, as a group, do organizational variables distinguish between higher and lower rated plants? Here, discriminant analysis is used to determine how combinations of organizational variables differentiate between high and low performers.

4.2.1 Bivariate Relationships

Tables 5, 6, and 7 provide a summary of the initial analysis linking plant organization characteristics to the safety indicators. The tables focus on vertical, horizontal, and coordinative patterns, respectively. The coefficients in the tables are Pearson product moment correlations between each organizational variable and each of the four safety indicators.

The safety indicators, it will be remembered, are based on residualized data. In other words, the effects of plant age, region, vendor, and size have already been taken out of the safety indicators. The squared coefficients, therefore, can be interpreted as the proportion of variance in the safety indicators explained by each of the organizational variables after the control variables have been allowed to explain all the variance they can. This approach most likely underestimates the contribution of the organizational variables. In technical terms, all of the covariance between the control variables and the safety indicators that is shared with the covariance between the organizational variables and the safety indicators has been attributed to the control variables This is unlike

Table 5: VERTICAL PATTERNS AND SAFETY INDICATORS

(Correlations with Residualized Indicators)

	REGULATORY COMPLIANCE	HARDWARE FAILURE	HUMAN ERROR	PLANT RELIABILITY
RANK MEASURES				
LONGEST LINE				
REACTOR	.08	.08	-.02	-.18
FACILITY	-.06	.24*	.19	-.27**
AVERAGE LINE				
REACTOR	-.18	.17	.20	-.18
FACILITY	.29**	.26*	-.05	-.36**
FACILITY RANK	.29**	.04	-.16	-.13
SPAN MEASURES				
MANAGER SPAN				
REACTOR	.02	.17	.10	-.20
FACILITY	-.09	-.07	.07	.22*
THIRD LINE SPAN				
REACTOR	-.01	.10	.03	-.39**
FACILITY	-.04	.06	.04	-.21*
SECOND LINE SPAN				
REACTOR	.00	.12	.13	.05
FACILITY	.22	.12	-.03	.04
LINE SPAN				
REACTOR	.19	.20	.01	-.12
FACILITY	.11	.05	-.01	-.00
STAFF SPAN				
REACTOR	.01	.44**	.26	-.49**
FACILITY	.02	-.02	.02	.10
INTERMEDIATE SPAN				
REACTOR	.25	-.08	-.21	.18
FACILITY	.05	-.14	-.09	.17
NUMBER SUPERVISORS	.09	.40**	.16	-.27**

* .10 LEVEL OF PROBABILITY
** .05 LEVEL OF PROBABILITY
BASED ON 42 CASES WITH PAIRWISE DELETION OF MISSING DATA.

Table 6: HORIZONTAL PATTERNS AND SAFETY INDICATORS

(Correlations with Residualized Indicators)

	REGULATORY COMPLIANCE	HARDWARE FAILURE	HUMAN ERROR	PLANT RELIABILITY
1st LEVEL UNITS				
REACTOR	.19	-.03	-.12	.09
FACILITY	.06	-.23*	-.15	.21*
2nd LEVEL UNITS				
REACTOR	.19	-.01	-.09	.06
FACILITY	-.06	.19	.17	-.11
3rd LEVEL UNITS				
REACTOR	.00	.06	.03	-.13
FACILITY	-.15	.07	.14	-.27**
UNITS UNDER MANAGER				
REACTOR	.16	.09	.00	.10
FACILITY	-.19	-.03	.15	.26**

* .10 LEVEL OF PROBABILITY
** .05 LEVEL OF PROBABILITY

BASED ON 42 CASES WITH PAIRWISE DELETION OF MISSING DATA.

Table 7: COORDINATIVE PATTERNS AND SAFETY INDICATORS

(Correlations with Residualized Indicators)

	REGULATORY COMPLIANCE	HARDWARE FAILURE	HUMAN ERROR	PLANT RELIABILITY
DEPARTMENTAL LINKAGES				
OPERATIONS/MAINTENANCE	.11	.14	.04	-.03
OPERATIONS/TECHNICAL	.19	.31**	.05	-.16
MAINTENANCE/TECHNICAL	.21*	.17	-.06	-.14
PERSONNEL LINKAGES				
DEGREED - OPERATIONS	-.11	-.25*	-.08	.01
LICENSED - MAINTENANCE	.09	-.08	-.10	.07
LICENSED - TECHNICAL	.42**	.03	-.26*	.07
DEGREED - MAINTENANCE	-.14	-.26**	-.05	.09

* .10 LEVEL OF PROBABILITY
** .05 LEVEL OF PROBABILITY

BASED ON 42 CASES WITH PAIRWISE DELETION OF MISSING DATA.

the more general regression approach where mutual covariation between the control and organizational variables would be allocated according to their relative explanatory power. In other terms, the data show how much variation in performance is explained by an organizational factor, over and above plant age, region, vendor, and megawatts per hour produced.

In general, the analysis suggests that organizational factors are important in explaining NPP safety indicators. There are considerably more statistically significant correlations (those marked with asterisks) than would be expected by chance alone.[5]

Though we will discuss each of the significant correlations below, it is important to remember that each organizational variable is but one aspect of a more general organizational pattern (such as an emphasis on hierarchy or coordination). More study is required to determine how the various organizational variables together promote or threaten NPP safety.

Vertical patterns are associated with the safety indicators. Less hierarchial plants generally perform better. Horizontal patterns are somewhat less related to safety performance. Coordinative patterns appear to be important to safety performance. Plants scoring higher on coordination tending score better in terms of safety perfomance.

Different organizational variables are related to different safety indicators. For example, there is only one significant correlation between an organizational measure and the Human Error factor. However, the organizational variables are somewhat more consistently related to Hardware Failure and Plant Reliability. Among other things, this suggests that changes in organizational structure to promote organizational performance on one dimension of safety may or may not have an equivalent effect on other dimensions of safety. We now turn to a discussion of the specific findings.

4.2.1.1 Vertical Patterns

Facilities with a larger number of ranks and a larger average number of ranks tend to have more Hardware Failure and to have lower Plant Reliability than plants with less emphasis on vertical hierarchy. There is some indication that facilities with a broad manager span of control,

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[5]Due to the small number of cases in the present analysis and the exploratory nature of this investigation, correlations significant at the 0.1 level of probability have been judged to be substantively interesting. Selecting this level of significance means that one out of ten coefficients may be expected to be significant by chance, alone. Since there are roughly twice as many significant correlations than the "chance" level, the importance of the organizational variables is supported.

that is, where the heads of all or most major functions report directly to the top manager, have better Plant Reliability records. However, broad spans are not always related to better performance. Broader Spans appear somewhat less conducive to safety performance lower in the hierarchy. For example, narrower spans for the third level supervisors are associated with higher Plant Reliability. Then, again, increasing supervisory spans within the operations area (reactor unit) by adding staff functions appears to promote poorer performance on Hardware Failure and Plant Reliability.

Performance on Regulatory Compliance appears to actually be improved by the addition of hierarchial levels both within the plant as measured by the number of ranks in the average line, and within the utility, as measured by the number of ranks between the facility manager and the Nuclear Vice President. The reasons for this anomaly are not clear. One could argue that bureaucratically organized organizations are better equipped to deal with the bureaucratic aspects of regulation. What is clear, is that organizational factors promoting regulatory compliance may improve, degrade or have no effect on hardware failure, human error and plant reliability.

4.2.1.2 Horizontal Patterns

There are relatively fewer significant relations between the horizontal patterns and the safety indicators than there are for the vertical patterns and coordination. There is a slight tendency for facilities with a large number of first level units to have less Hardware Failure and better Plant Reliability. This same pattern holds for Plant Reliability and the number of units under the facility manager. However, a large number of third level units tends to be associated with lower reliability. Taken together, these relationships suggest that more reliable plants tend to be organized in such a way that the manager has unmediated contact with the various general functions (e.g. maintenance, engineering), but that further differentiation of units (e.g. mechanical vs. electrical maintenance) takes place at the lowest levels of supervision. It must be stressed, however, that these relationships are weak and suggestive at best.

4.2.1.3 Coordination Patterns

A number of significant and interesting relationships emerge between the dimensions of safety performance and the coordination measures. The various coordination measures based on the sharing of reciprocal knowledge across departmental/functional lines show considerable support for the notion that the degree of coordination is an important determinant of plant safety performance. The presence of degreed engineers among supervisory personnel in operations and degreed engineers in maintenance are related to lower rates of Hardware Failure. The presence of licensed personnel in the plant engineering areas is associated with lower rates of Human Error, and with better Regulatory Compliance. Consistent with these findings, the lower (organizational rank) in the organization the maintenance and technical functions are linked, the more compliant the organization. Again, however, a simple pattern does not emerge across all findings. Somewhat anomalously, the lower in the organization the technical and operations functions are linked, the greater the frequency of Hardware Failure. In sum, the bulk of the data suggest that coordination, or lack of it, lies at the core of variation in plant safety performance. The pattern of results also suggest a complex series of relationships among organizational factors and safety. Thus, the next section discusses a multivariate treatment of the data.

4.2.2 Multivariate Relationships

In this section we report a series of discriminant analyses designed to identify the combinations of organizational variables predictive of membership in the group of either high or low performers on each of the dimensions of plant safety.

Discriminant analysis can be used to define a linear combination of variables that is capable of predicting group membership.[6] In this case, we have divided the sample of plants into two groups depending upon whether they are above or below the mean on a given safety indicator. Then, a stepwise discriminant analysis has been performed. Here, organizational variables are added to an equation predicting an aspect of safety until they fail to add non-redundant, and significant explained variation. The result is a set of organizational predictors that are both individually predictive of safety performance and maximally predictive as a group.

This analysis suggests more complex models behind the organizational determinants of safety. It may be extended to include interactive terms (combination effects). The current analysis is limited due to the missing cases. Further, only a subset of the organizational variables discussed so far could be employed. Specifically, these analyses are based on the horizontal and vertical patterns data for the facility-wide unit. Plants were eliminated from consideration when any data in an analysis were missing. Because of missing cases, the number of plants has been reduced to 33 for Plant Reliability and 29 for the other safety indicators.

Table 8 presents the results of the discriminant analyses. For each of the safety indicators, the discriminant analysis program has been allowed to select the set of statistically significant (.05 level) organizational predictors. The resulting coefficients represent the change in the organizational indicator score associated with moving from one performance

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[6]At this initial stage more complex, non-linear models were not examined.

TABLE 8: DISCRIMINANT ANALYSIS

SAFETY INDICATOR	ORGANIZATIONAL* VARIABLES	UNSTANDARDIZED** DISCRIMINANT FUNCTION COEFFICIENTS	R^2	PERCENT CORRECTLY CLASSIFIED
REGULATORY	2nd Line Span Range	.36		
COMPLIANCE	Intermediate Span	-.49		
(N = 29)	Manager Span	-.79		
(Low Compliance = 1,	Line Span	.77		
High Compliance = 0)	(Constant)	.17	.55	84.9%
HARDWARE	# of Ranks	6.02		
FAILURE	# 3rd Level Units	-.70		
(N = 29)	# 2nd Level Units	-.13		
(Low Failure = 0,	(Constant)	-21.89	.38	68.6%
High Failure = 1)				
HUMAN	Intermediate Span	.50		
ERROR	Manager Span	1.36		
(N = 29)	# of Ranks	4.67		
(Low Error = 0,	(Constant)	-21.16	.41	71.4%
High Error = 1)				
PLANT	Staff Span	.65		
RELIABILITY	# of Ranks	4.58		
(N = 33)	# 2nd Level Units	-.12		
(Low Reliability = 1,	(Constant)	-20.11	.59	78.1%
High Reliability = 0)				

* All organizational variables are measured at the facility level.

** All coefficients are significant at the .05 level of probability and are listed in order of selection.

group to the next. (The coefficients are essentially equivalent to regression coefficients.) For example, moving from the high compliance group to the low compliance group would be associated with a .36 unit increase in the second line span range. Also given in the table is the percent explained variation in group membership accounted for by the linear combination of the organizational measures. Finally, the table also provides the percent of the plants that would be correctly classified into high and low performers using the predictive power of the discriminant function.

4.2.2.1 Discriminant Analysis of Regulatory Compliance

Four variables emerged as predictive of Regulatory Compliance. Plants with wider ranges of second line supervisory span tend to fall into the non-compliant group, as do plants with broad spans in the line or operations component. However, broad plant manager and intermediate function spans tend to define membership in the compliant group. Together, these variables explain 55% of variation in group membership, and 85% of the cases are correctly classified. This pattern of results calls for some interpretation.

Taken as a set, these variables describe a compliant organization as one where most plant functions report directly to the plant manager rather than through additional levels of hierarchy. Relatively narrow supervisory spans characterize the operations component, but relatively broad spans characterize the craft and professional groups in the rest of the organization. This finding is consistent with the general position in the organizational literature that craft and professional activities require relatively more autonomy than routine activities in order to perform adequately. However, counterbalancing this set of relationships is the fact that plants with widely varying spans (as measured by the range of second line supervisory span) tend toward less compliance. This result underscores the importance of multivariate analysis, since two competing causal forces appear to be at work. That is, the consistency in administration that would be characterized by roughly equal spans throughout the plant contributes positively to compliance, whereas the individual needs of specific functions for different supervisory spans are also important to compliance. This pattern suggests that a certain amount of function specific variation in spans does not harm compliance, but that extremes in spans might.

Again, it should be emphasized that the data available for analysis are limited and considerably more effort needs to be expended to more accurately model the organizational determinants of compliance. One particularly important variable is missing -- the number of employees. In other studies organizational size had been an important moderator (see Osborn et al., 1980 for a discussion of these effects). For instance, in plants with more employees, wider variations in span may be appropriate and be associated with compliance. With fewer employees just the opposite may be the case.

4.2.2.2 Discriminant Analysis of Hardware Failure

Three variables were selected by the discriminant analysis program as significantly related to membership in the high and low Hardware Failure groups. Plants with more vertical ranks tend to fall into the high Hardware Failure group. On the other hand, plants with relatively broad structures, as indicated by the number of second and third level units, tend to perform better in terms of Hardware Failure. Together, these variables explain 38% of the variation in group membership, and 69% of the cases are correctly classified by this analysis.

Plants with low Hardware Failure rates tend toward a form that emphasizes organization on the basis of horizontal specialization rather than hierarchy. These plants support more distinct departments and, perhaps, a wider range of specialties than the plants with higher levels of Hardware Failure. In addition, positive performers are shorter from top to bottom. These findings suggest several plausible explanations. Included is the possibility that increased functional specialization promotes a more sophisticated approach to technical problems and that shorter hierarchies do not place barriers between units. With fewer levels, the coordination necessary to make the high degree of specialization workable may be easier to obtain. As before, the role of number of employees should be considered, among other unmeasured factors.

4.2.2.3 Discriminant Analysis of Human Error

Results for Human Error show the immediate benefits of a more sophisticated analysis. While there were few significant bivariate relations, the discriminant analysis shows that three variables emerge as predictive of membership in high and low Human Error groups. Plants with more vertical ranks are, again, poorer performers and are significantly more likely to fall into the high Human Error group. As opposed to the conditions for Regulatory Compliance, however, broad plant manager and intermediate spans are more likely to experience human error. Together, these variables explain 41% of the variation in group membership, and 71% of the cases are correctly classified.

Taken as a group, the data suggest that Human Error is reduced by minimizing the height of the organization while maintaining narrow managerial and supervisory spans. This combination, of course, is easiest to achieve in small organizations. In fact, this combination of predictors may actually be acting as a proxy measure for size of the organization and it may simply be the case that large organizations promote a greater frequency of human error. This inference has considerable intuitive appeal and demonstrates the importance of the fact that data on the number of employees in the plant have not been included in the present analysis. Such data is not currently publicly available. However, these data are crucial for an understanding of the organizational dynamics behind plant safety performance.

4.2.2.4 Discriminant Analysis of Plant Reliability

The three variables that emerge as significantly related to Plant Reliability are staff span, the number of ranks in the organization, and the number of second level units. Plants with broader spans in the staff component tend to be less reliable. Since the staff component is comprised primarily of administrative functions, it is difficult to picture a direct mechanism causing the observed relationship. However, analysis of the means and standard deviations suggests a possible hypothesis for investigation. Namely, staff may be grouped together into staff units or staff may be assigned to particular line managers. If assigned to line managers (resulting in narrow staff spans), there may be an improvement in Plant Reliability.

Once again, an emphasis on many vertical ranks appears to be associated with poorer performance. Finally, a large number of second level units tends to be associated with better Plant Reliability. Together, these variables explain 59% of the variation in group membership and 78.1% of the cases are correctly classified.

Despite the anomalous finding concerning staff span, plant reliability seems to be a function of the same basic pattern predictive of hardware failure. That is, short, broad organizations tend to perform somewhat better. More work is needed to determine the precise mechanisms underlying this general pattern is translated into performance.

4.2.2.5 Summary of Discriminant Analyses

Several substantive and methodological issues emerge from the discriminant analysis. On the substantive side, it can be noted that substantial amounts of variation in plant safety indicators can be explained by the linear combinations of a few organizational variables. Second, with some deviations from the general pattern, plant organizations that de-emphasize hierarchy and emphasize horizontal specialization tend to perfom better. Third, the specific organizational aspects emerging to predict dimensions of safety are not identical. Different sets of organizational factors are important for different dimensions of safety. Fourth, the question of developing safety appears to be quite difficult since the organizational variables combine in complicated ways to predict plant safety performance.

On the methodological side, the analysis has demonstrated the need for better measurement of relevant organizational factors in order to refine the initial models that this preliminary analysis has begun to define. Particularly important would be data on the number of employees.

5.0 SUMMARY AND CONCLUSIONS

This report shows a direct empirical relationship among organizational factors and safety indicators for nuclear power plants. While the data are far from definitive, the initial empirical analysis strongly suggests that how a plant is organized makes a substantial difference in safety performance.

Section 1 of this report has outlined a strategy for linking characteristics of NPP organization to NPP safety. It began by describing the likely importance of organizational characteristics to NPP safety.

Section 2 described a way of looking at organizations that is both systematic and relevant to safety issues. This perspective was then compared to existing data sources to determine whether currently available data would permit an empirical analysis as a basis for a better informed regulatory approach. The conclusion drawn from this comparison was that the available data allow only for an incomplete analysis of the relationship between organization and NPP safety. This section concluded with a description of the methods used in taking advantage of existing data to construct variables for a preliminary analysis. The measures included estimates for the vertical and horizontal dimensions of administrative form, as well as a few measures of coordination.

Section 3 concerned plant safety performance indicators. It began with a review of the literature which pointed out the problems with using existing, single indicators of plant safety. A brief review of LERs, outage data, violations data, and other sources pointed out biases which, if left uncorrected, would lead to potential errors in their use as safety indicators. A strategy was introduced for dealing with bias in the indicators. Specifically, LERs, violations, outages, and SALP ratings were residualized on a set of control variables (plant region, size, age, and vendor) to remove sources of bias. Then, the residualized indicators were factor analyzed to identify underlying patterns indicative of plant safety performance. Four factors were statistically identified: Regulatory Compliance, Hardware Failure, Human Error, and Plant Reliability.

Section 4 related the available organizational variables to each of the safety indicators. The section started with a description of industry patterns on each of the organizational variables. While the operations area is organized in a fairly standard way, this analysis shows that there is wide variation in the way that utilities set up NPP organization. Next, the bivariate relationships between the organizational variables and the safety indicators were addressed. A substantial number of these relationships proved to be statistically significant. A general theme is that an emphasis on hierarchy is inconsistent with performance on most of the safety indicators. However, different elements of organization are important for different dimensions of safety.

Discriminant analysis was then used to look at broader organizational patterns relative to safety. Again, it was found that an emphasis on vertical ranks appears counterproductive. It was also found that different organizational factors are frequently predictive of different safety indicators. The discriminant analyses were able to explain substantial proportions of variation in plant performance as well as demonstrating the importance of the multivariate approach. Specifically, over half the variation in Regulatory Compliance and Plant Reliability was associated with a few organizational factors. For Human Error and Hardware Failure the proportion of variance associated with organizational factors was .41 and .38, respectively.

There are a number of limitations on the analysis. First, the available organizational data are incomplete and of unknown reliability. Basic organizational characteristics, such as the number of workers in each plant, are simply not available. This severely limits the ability to adequately model the relationships among organization and plant safety. Second, only a subset of plants has been included. This subset is somewhat biased in terms of single unit sites. In addition, data were restricted to plants. The coding and analysis of corporate level data was beyond its scope of the project. It is expected that the relation between plant organization and corporate organization is important for NPP safety.

The safety indicators constructed here also have limitations. At a minimum, the analysis should be replicated with more recent data. Second, attempts should be made to refine the analysis by taking into account causal linkages among events and components and adding other performance measures where possible.

Finally, the analysis reported here provides only a first look at the complex relationship between organization and NPP safety. Additional empirical work is essential. .

6.0 RECOMMENDATIONS

This section outlines a series of recommendations based on the findings reported here. They have been grouped into three sets: recommendations concerning the availability of data, recommendations concerning future analysis of the relationship between organization and nuclear power plant safety, and recommendations concerning the safety indicators.

6.1 Availability of Data

Given the increasing evidence that organizational factors are important to safety, and that the NRC must remain knowledgeable in this area, basic descriptive information on the organization of the plants and utilities could be extremely useful to an informed regulatory strategy. Much of the descriptive information is not currently available in a highly reliable form. Therefore, the following are recommended:

1. The NRC should develop a standardized, baseline data set on the organizational characteristics of plants and utilities. This data set should be comprehensive, including information on the factors described here as relevant to safety.

2. Whether through updating FSARs and Technical Specifications, the use of industry groups, resident inspectors, or some other means, the NRC should assure that the data remain current.

6.2 Analysis of Organizational Factors

The current analysis needs to be replicated and expanded to provide a more thorough assessment of the relation between organizational structure and NPP safety. Therefore, the following are recommended:

1. Existing data should be further explored to try to expand the coverage of key concepts and the reliability of the measures.

2. Equivalent analyses should be directed at the corporate organization.

3. A more systematic analysis should be undertaken.

4. More detailed multivariable modeling should be supported as additional data become available.

6.3 Safety Indicators

While the safety indicator analyses reported here show promise, it should be validated and extended.

1. The current model should be replicated with 1982 data to provide validation for this approach.

2. The current analysis should be extended by adding additional indicators (such as positive measures of performance), and by looking at the current indicators in more detail. For example, causally linking indicators over time would likely add considerably to the strength of the analysis.

3. The feasibility of developing indicators more directly relevant to sub-areas of the plant should be explored.

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APPENDIX A

RESIDUALIZATION APPROACH

1. INTRODUCTION

This appendix describes the procedure used to statistically adjust the safety indicators utilized in this report. The purpose of the procedure is to remove variation in the safety measures that is due to physical plant difference in factors not related to safety.

2.0 RESIDUALIZATION APPROACH

The safety measures used in the report fall into four groups: frequencies of LERs by cause, I & E violations by severity level, forced outages by cause, and composite SALP based upon functional area categorical ratings. The variables used to adjust each of these indicators were plant size, age, type and vendor, and region. Plant size was measured in net megawattage, and age was broken down into two variables: one for the months between 0 and 36, and another for the months beyond 36. For region, reactor type and vendor, a set of dummy variables was created to categorize plants on these factors. For region, four dummy variables were computed corresponding to plant locations in the west, midwest, northeast, and south of the U.S. Each of these four dummy variables are assigned either the value "1" if the plant is located in that region or the value "0" if the plant is not located in that region.

The same logic was applied to reactor type and vendor which was broken down into three dummy variables. The first (BWR) was given a value of "1" if the plant was a BWR and a "0" for all other types, the second (PWRW) taking the value "1" if the plant was a Westinghouse PWR and a "0" for all others, and the third (PWROTH) was given a "1" if the plant was a PWR manufactured by Babcock and Wilcox (B & W) or Combustion Engineering (CE) and "0" for all others.

The procedure utilized to remove variation in the safety indicators caused by these control variables is known as "residualization." This is accomplished by, first, estimating the number of LERs, I & E violations, forced outages, and average SALP scores that would be expected to occur given the plant's value on each of the control variables (region, age, size, reactor type and vendor). The second task is to calculate the difference between the actual frequencies of the safety indicators and predicted frequencies estimated from the control variables. This difference, or residual, represents the degree of variation in the safety indicators <u>unexplained</u> by the control variables. Hence, the residualized safety variables are interpreted as frequencies among the safety indicators deviating from what would be expected of a plant due to its size, age, reactor type and vendor, and location.

As noted above, the first step in the residualization procedure is to estimate the effects of the control variables on the safety indicators. A by-product of this adjustment procedure is the estimation of the regression equation with coefficients for each of the control variables regressed against a particular safety variable. In other words, it is possible to see which control variables are more important in explaining safety indicators and the direction of their impact on each of the indicators.

Nine regression equations were constructed using each safety indicator as the dependent variable and the control variables as the independent variables. The use of dummy variables in regression analysis, requires that one dummy be omitted for each set of dummy variables (e.g., the set of region dummies) and the effect of the included variables are calculated with respect to the omitted category. The value of this omitted category is measured by the value of the constant term (see Table A-1). For the analyses of safety indicators conducted for the report, the South and PWROTH dummy variables were omitted. Essentially, the regression coefficients in Table A-1 are interpreted as difference between the omitted category and the mean of the included categories. In other words, the constant term, in this case, represents the mean value of the safety indicator for southern plants whose reactor types are PWRs manufactured by B&W or CE. The coefficients of the other variables represent the positive or negative decrement from this value.

3. RESULTS OF RESIDUALIZATION

Turning to the results of the regression analysis in Table A-1, it can be seen that for five of the nine equations, physical plant characteristics account for 33% or more of the variance ($R^2 = .33$). For I & E Severity Level V violations, almost 50% of the violations can be attributed to the control variables. However, for the LER variables and I & E Severity Level VI violations, physical plant characteristics provide very little predictive power.

A general guide in interpreting the regression coefficients in Table A-1 is that those regression coefficients more than twice as large as their standard errors are statistically significant. These regression coefficients are identified with an asterisk. The first column presents the regression coefficients for the Human and Procedural LERs equation. None of the independent variables are significant predictors in this or the second column showing the coefficients for the other type of LERs.

Of the four categories of control variables, region is by far the most important. Size of plant has no significant effects in any of the nine regression equations. The age of plant between 0 and 36 months has significant effects for human caused forced outages with the older plants having fewer of these outages. The type and vendor variables indicating that the plant is a BWR has a positive effect on the number of level II and III violations. Thus, controlling for all other variables in the equation, BWR plants have 1.7 more of the most serious violations than southern CE or B&W plants.

The most important set of control variables are those representing region. In 5 of nine regressions, region proved to be a significant predictor of the safety indicators. Overall, southern plants appear to have a significantly greater number of I & E level V violations than the other regions. Holding constant the type and vendor of plants, southern plants have an average of 13.6 level V violations as opposed to 4.1 for northeastern plants, 5.0 for midwestern plants, and 4.3 for western plants. Examining the two other I & E severity level categories, southern plants are undifferentiated from midwestern plants in the frequencies of level II and III and level IV violations (i.e., their average violation frequencies are not significantly different). In these cases, northeastern plants have 2.9 more level IV violations than southern plants but are not significantly different from southern plants with respect to average frequencies of level II and III violations. Conversely, western plants are not significantly different from southern plants with respect to average frequency of level IV violations, but they have 2.8 significantly more level II and III violations than southern plants.

Finally, region is an important explanatory variable in explaining differential rates of SALP scores. The average SALP score for southern plants, holding all other control variables constant, was 1.8 (or, an "average" rating on the 1 to 3 SALP scale). Midwestern and western plants having an average SALP score of 1.5 and 1.7 respectively were not significantly different from the southern plants. However, northeastern plants had an average SALP score rating of 1.3. This full half point difference between northeastern and southern plants is significantly different.

Two considerations must be borne in mind when examining these regression coefficients and their interpretation. First, the use of more than one dummy variable in a regression equation means that the constant term or the omitted category, contains the joint effects of the dummy variable categories utilized. In this analysis, the regional effect of the southern regional variable is mixed in with the effects of the PWROTH variable. Hence, the above discussion of regional effects assumes that there is no interaction between region and vendor/type. Further analyses are required to examine such interactional effects.

In sum, the residualization of safety indicators accomplishes two things. First, it permits the analysis of plant safety and dimensions of plant safety using indicators that have been controlled for systematic bias due to plant characteristics. Controlling for these characteristics removes whatever "noise" they may cause in the variation of safety indicators. Second, residualization also permits the examination of exactly what effects the control variables in fact do have on the safety indicators. It is this aspect of residualization that has been examined here.

4. SUMMARY

Overall, the results of the regression analyses between the safety indicators and the control variables show a general lack of predictions. Physical plant characteristics such as age, reactor type and vendor, and reactor size proved to be weakly related to variations in the frequencies and rates of the four safety indicator data sets. Regional location, however, emerged surprisingly as the strongest predictor of safety indicator variation. This was found to be the case for I & E severity levels II and III, IV, and V; it was also the strongest predictor variable for average SALP score and nonhuman caused forced outages. While no explanation is offered here of why regional location would be such a strong predictor of safety indicator variation, further analysis of this relationship is suggested.

The confirmatory factor analyses detailed in Section 3 of the report utilized the residualized safety variables. These variables were calculated using the results of the regression equation just discussed. Plant specific values for each of the control variables were substituted into the general equations whose coefficients are given in Table A-1 and an estimated value for each safety variable calculated. The difference between the observed and the estimated value is the residualized safety value for each plant. Using the residualized safety variable, the subsequent analyses were able to generate unbiased estimates of the underlying safety components and assign specific scores to each plant for these components.

TABLE A-1: REGRESSION COEFFICIENTS AND STANDARD ERRORS FOR CONTROL VARIABLES REGRESSED AGAINST SAFETY INDICATORS

CONTROL VARIABLE		SAFETY INDICATORS: HUMAN AND PROCEDURAL LERs	OTHER LERs	I & E SEVERITY LEVELS 2&3	I & E SEVERITY LEVEL 4	I & E SEVERITY LEVEL 5	I & E SEVERITY LEVEL 6	HUMAN CAUSED FORCED OUTAGES	NONHUMAN CAUSED FORCED OUTAGES	AVERAGE SALP SCORE
NOREAST	b	-.14	1.36	-.36	2.89*	-9.49*	-.07	-.03@	-.00	-.51*
	se	(.69)	(1.94)	(.82)	(1.44)	(2.16)	(.85)	(.09)@	(.00)	(.17)
BWR	b	-.32	2.61	1.71*	1.28	1.25	-1.00	.07@	.00	.24
	se	(.69)	(1.93)	(.81)	(1.43)	(2.15)	(.84)	(.09)@	(.00)	(.17)
SIZE	b	.00	.00	.00	.00	.00	-.00	.00@	-.00@	.00
	se	(.00)	(.00)	(.00)	(.00)	(.01)	(.00)	(.00)@	(.00)@	(.00)
AGE (0-36)	b	.05	.16	-.04	-.17	-.10	.02	-.03@*	-.06@	-.01
	se	(.05)	(.15)	(.06)	(.11)	(.16)	(.06)	(.01)@	(.05)@	(.01)
WEST	b	.79	4.26	2.77*	.89	-9.26*	-1.93	.00	-.00	-.14
	se	(1.11)	(3.13)	(1.32)	(2.32)	(3.49)	(1.37)	(.00)	(.00)	(.27)
PWRW	b	-.63	.08	.06	-.26	-1.03	.24	.05@	.00	-.13
	se	(.69)	(1.93)	(.81)	(1.43)	(2.14)	(.84)	(.09)@	(.00)	(.17)
MIDWEST	b	-.36	-.47	.51	-.11	-8.60*	-.60	-.04@	-.00*	-.31
	se	(.70)	(1.95)	(.82)	(1.45)	(2.18)	(.85)	(.09)@	(.00)	(.17)
AGE (37+)	b	.00	.01	.02	-.01	.02	-.00	.00@	-.01@	-.00
	se	(.01)	(.03)	(.01)	(.02)	(.03)	(.01)	(.00)@	(.01)@	(.00)
CONSTANT	b	-.70	-5.02	-4.45	7.97	13.60	3.08	.00	.01	1.83
R^2		.08	.17	.33	.23	.48	.12	.38	.35	.33

@NOTE: COEFFICIENTS MULTIPLIED BY 1000.
* = Significant at .001 level.

APPENDIX B

ORGANIZATIONAL CHARACTERISTICS OF THE INDUSTRY

1. INTRODUCTION

This Appendix provides a detailed discussion of most of the organizational variables used in the analysis. The purpose of this discussion is less to describe the distribution of the variables, themselves, than to use the variables as a set to provide a description of the organizations in the sample. Using descriptive statistics, it is possible to derive an overall picture of the formal organization strategies employed by the organizations in the sample. It is also particularly interesting to note areas where there is wide variation within the industry in organizational structure.

Four issues are examined. First, what is the breadth of the formal organization? How many separate units are there? Second, what is the depth of the formal organization? How many ranks or levels of management are employed? Third, how do the utilities organize managers in relation to workers? What is the leader/lead ratio or span of control? Fourth, what is the internal consistency in the deployment of leaders and followers? Are line and staff treated the same way? Figures B-1 through B-26 provide the frequency distribution, means, and standard deviations for each of the organizational variables. Figures B-1 through B-13 provide information for the facility level of analysis. Figures B-14 through B-26 provide information for the reactor level of analysis.

2. FACILITY ORGANIZATION

2.1 Breadth of the Facility Organization

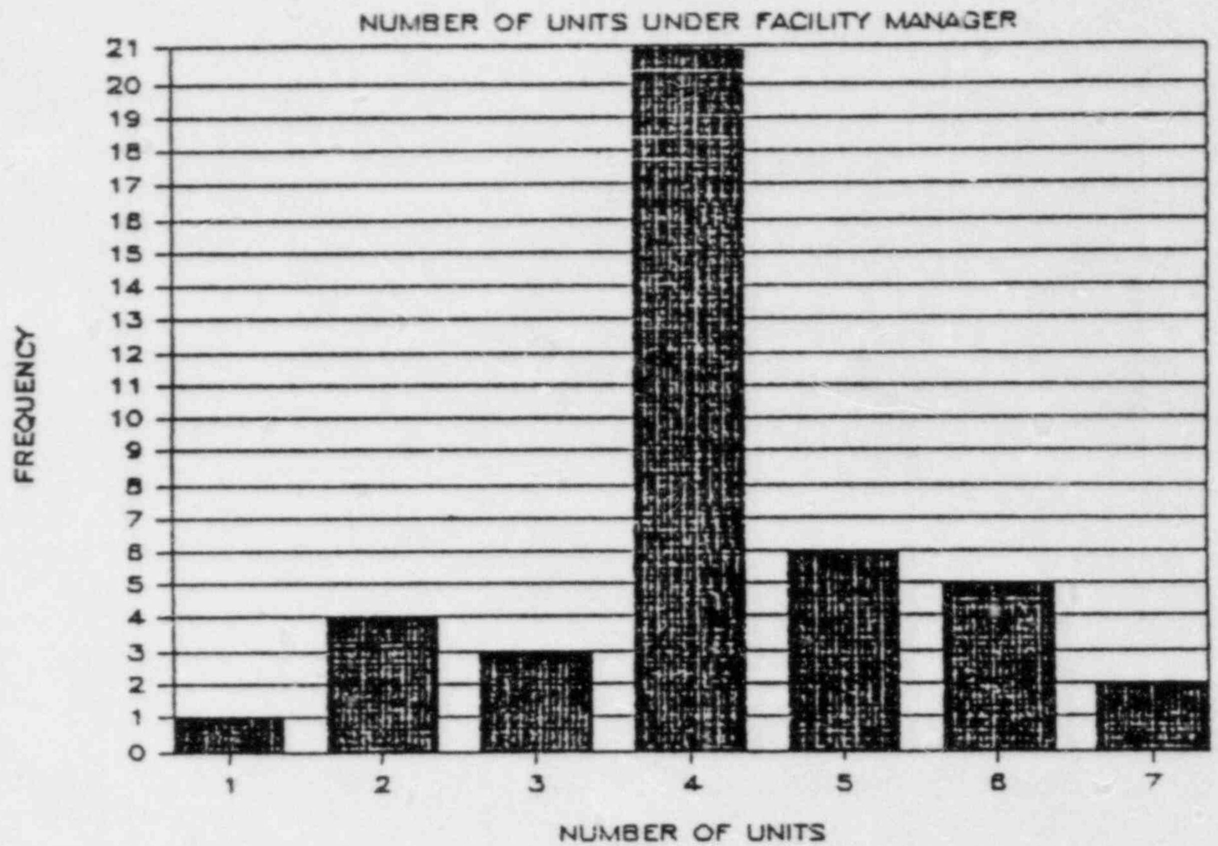
Figures B-1 through B-4 show summary statistics for the number of administrative units. As expected, the number of units systematically increases as one moves down the hierarchy. While there is a tight range for the number of units under the facility manager (Figure B-1) and the number of third level units (Figure B-2), there is considerable diversity in the number of second (Figure B-3) and first level units (Figure B-4). The greatest disparity in the sample occurs at the second level. Responses are as follows for the number of units:

(a) Under the facility manager - the mode is 4; while there is a strong central tendency, the range is quite substantial.

(b) For the third level - the modal response is again 4, but there is more variation (standard deviation of 2.0) than for the number of units under the facility manager.

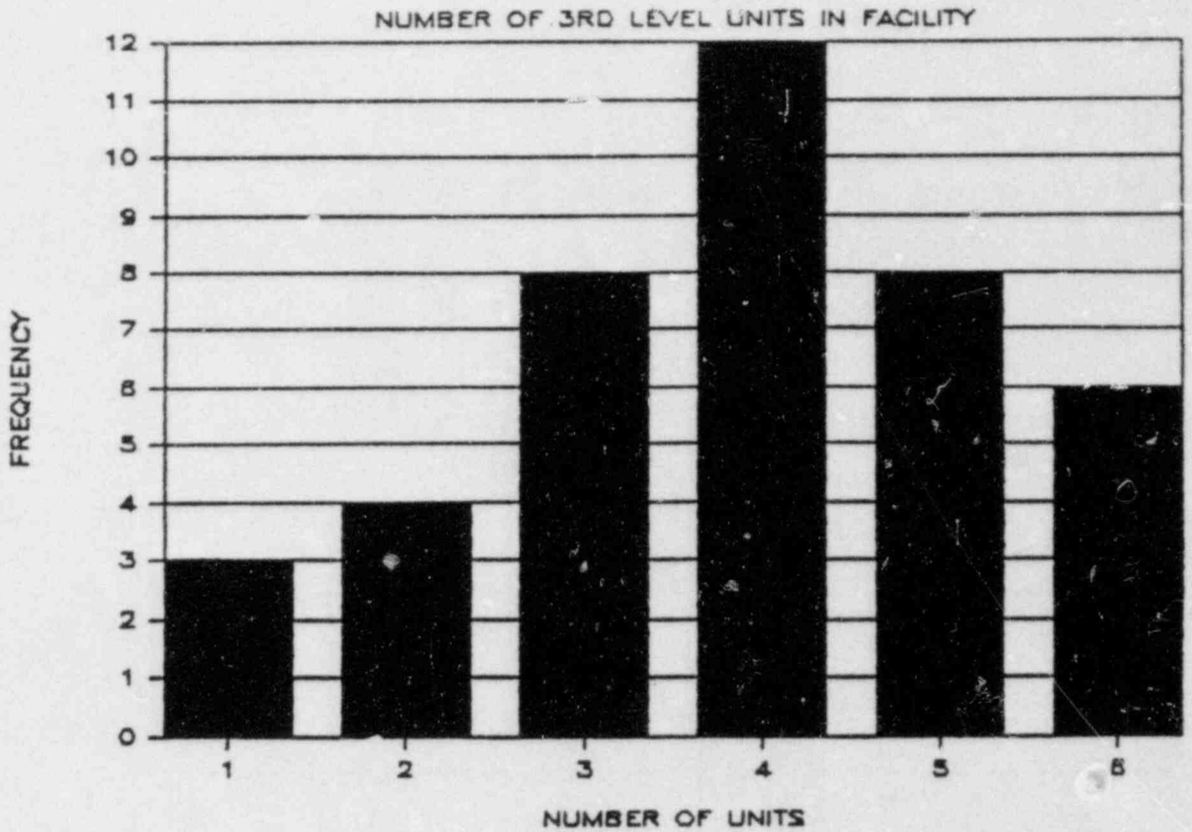
(c) Looking at the second level - the modal and mean number of units increases and there is substantial variance in the responses. Further, there is a tendency toward a bimodal distribution suggesting that different strategies may be employed in different parts of the industry.

FIGURE B-1



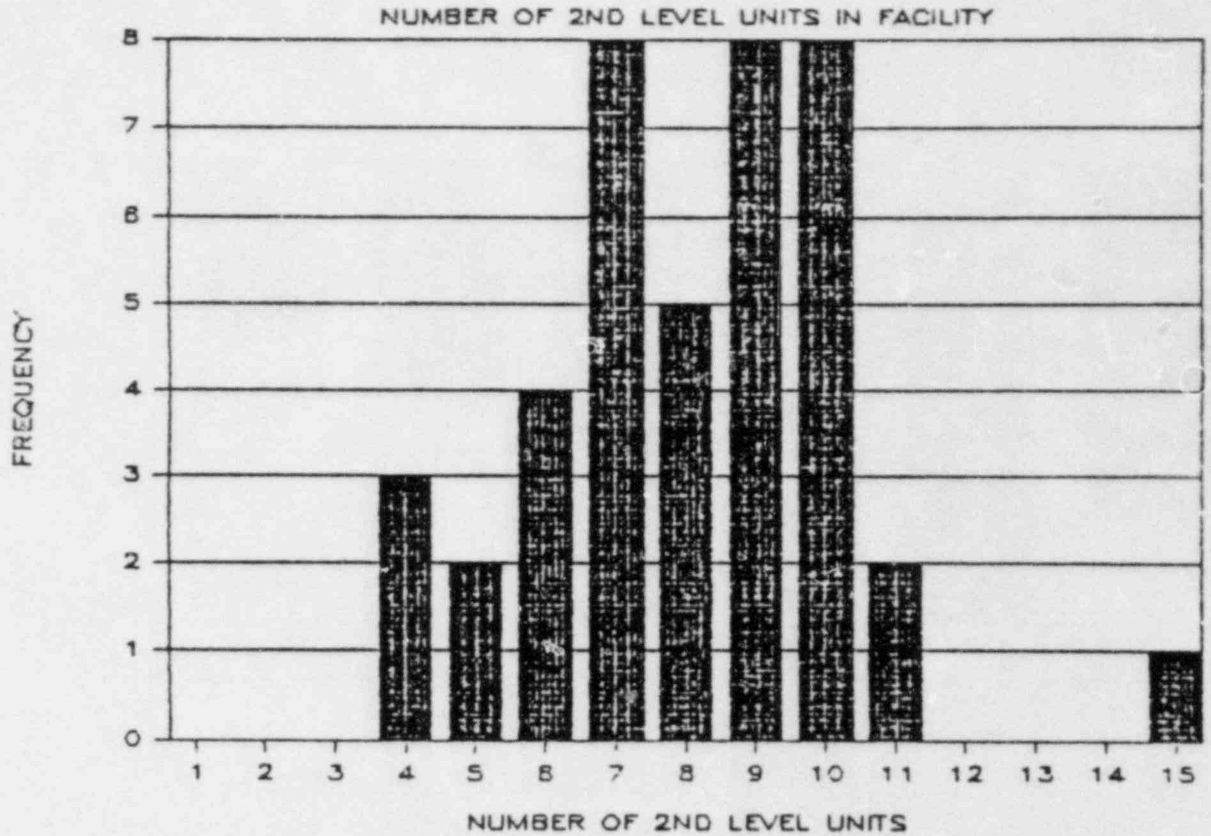
Mode = 4.0 $\overline{X}$ = 4.2 S.D. = 1.3

FIGURE B-2



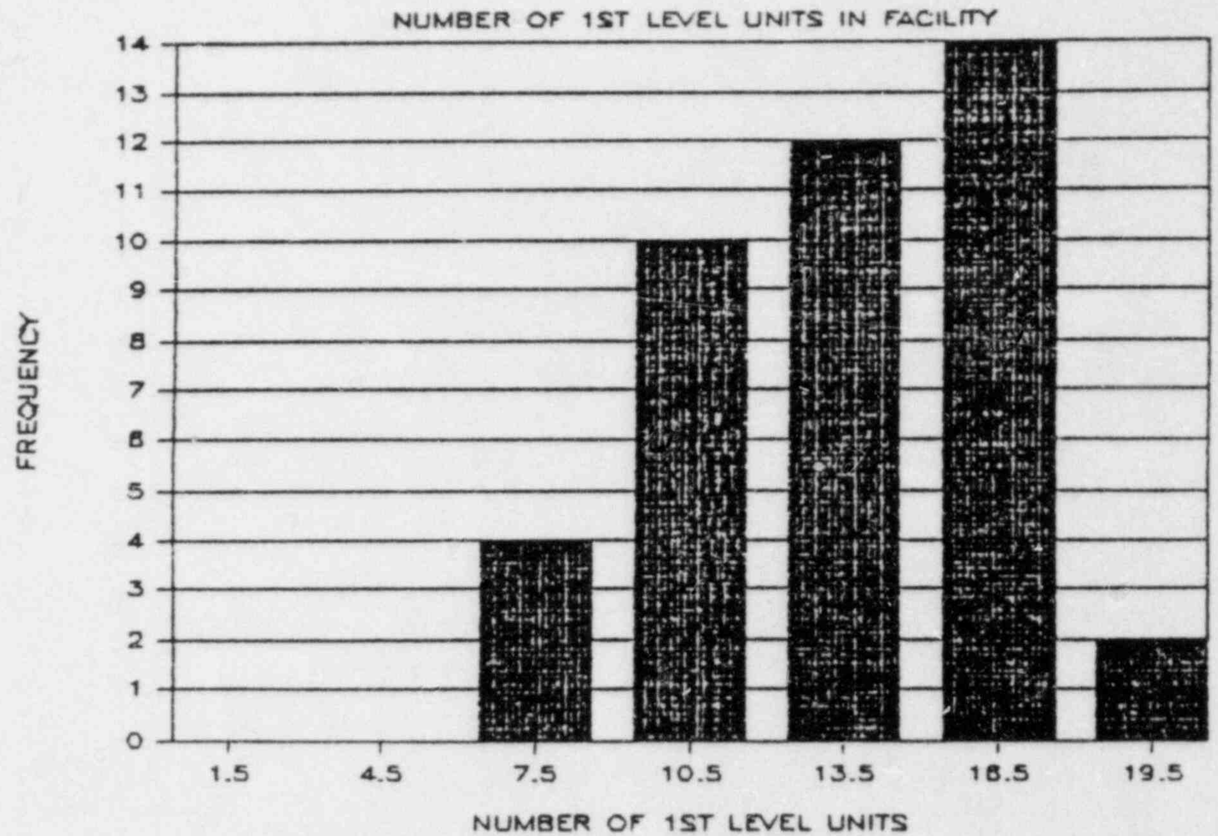
Mode = 4 $\overline{X}$ = 4.1 S.D. = 2.0

FIGURE B-3



Mode = 7 $\bar{X}$ = 9.0 S.D. = 6.6

FIGURE B-4



Mode = 17 $\overline{X}$ = 14.1   S.D. = 3.1

(d) For the number of first level units - the mean, mode and standard deviations are again larger.

2.2 Depth of the Facility Organization

Figures B-5 and B-6 show data for two important depth indicators. The first is the number of ranks (levels of management) in the longest chain of command. The modal response is 5, with a mean of 4.8 and a comparatively small standard deviation of .8. The second indicator (Figure B-6) is the average number of ranks. As expected, the average is lower than for the first indicator with a mode for the sample of 3.5. The mean is 3.4 with a comparatively small standard deviation of .5.

2.3 Spans of Control in the Facility Organization

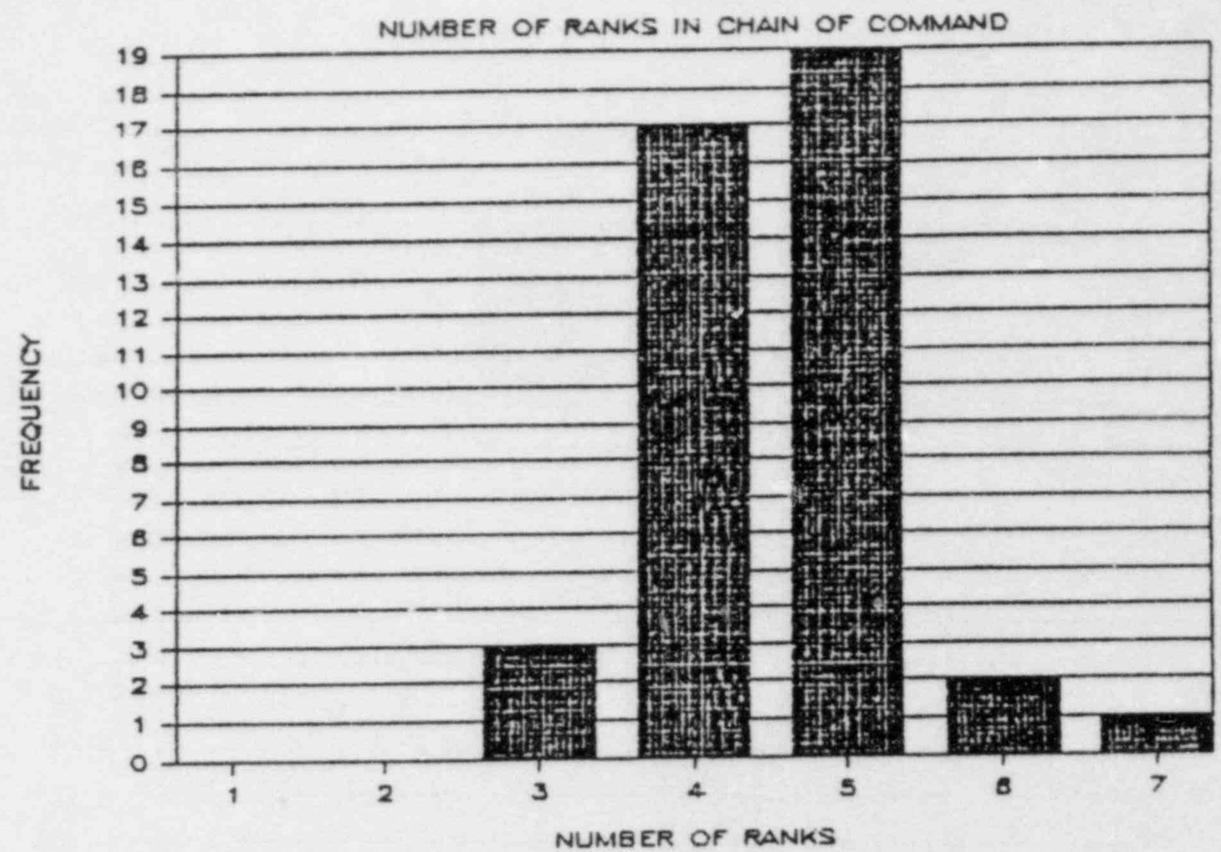
Figures B-7 through B-11 show average span of control for line versus staff units and by three levels of supervision within the plant. An easy way to interpret these is in terms of eader/lead ratios, with smaller averages suggesting more "chiefs" and comparatively fewer "Indians." For line units (Figure B-7), the leader/lead ratios fall within a comparatively narrow range with a modal response of 4.6, mean of 5.4, and a standard deviation of 1.6. For staff units, leader/lead ratios are considerably greater (more followers) and the variation is also greater (Figure B-8). Specifically, the mode is 6.5, the mean is 7.3, and the standard deviation is 3.2.

The leader/lead ratios by organizational level show a typical pattern with increasing averages below the facility manager as one moves down the hierarchy. Facility manager span (Figure B-9) has a mean of 4.5, third level supervisors (Figure B-10) have a mean of 2.4, and second level supervisors (Figure B-11) a mean of 3.4. Since data are not available on the numbers of direct workers, first level supervisory spans cannot be calculated. However, we would expect an even broader supervisory ratio at this level. Further, the pattern suggests that some facility managers have additional specialized support personnel.

2.4 Consistency in Spans of Control in the Facility

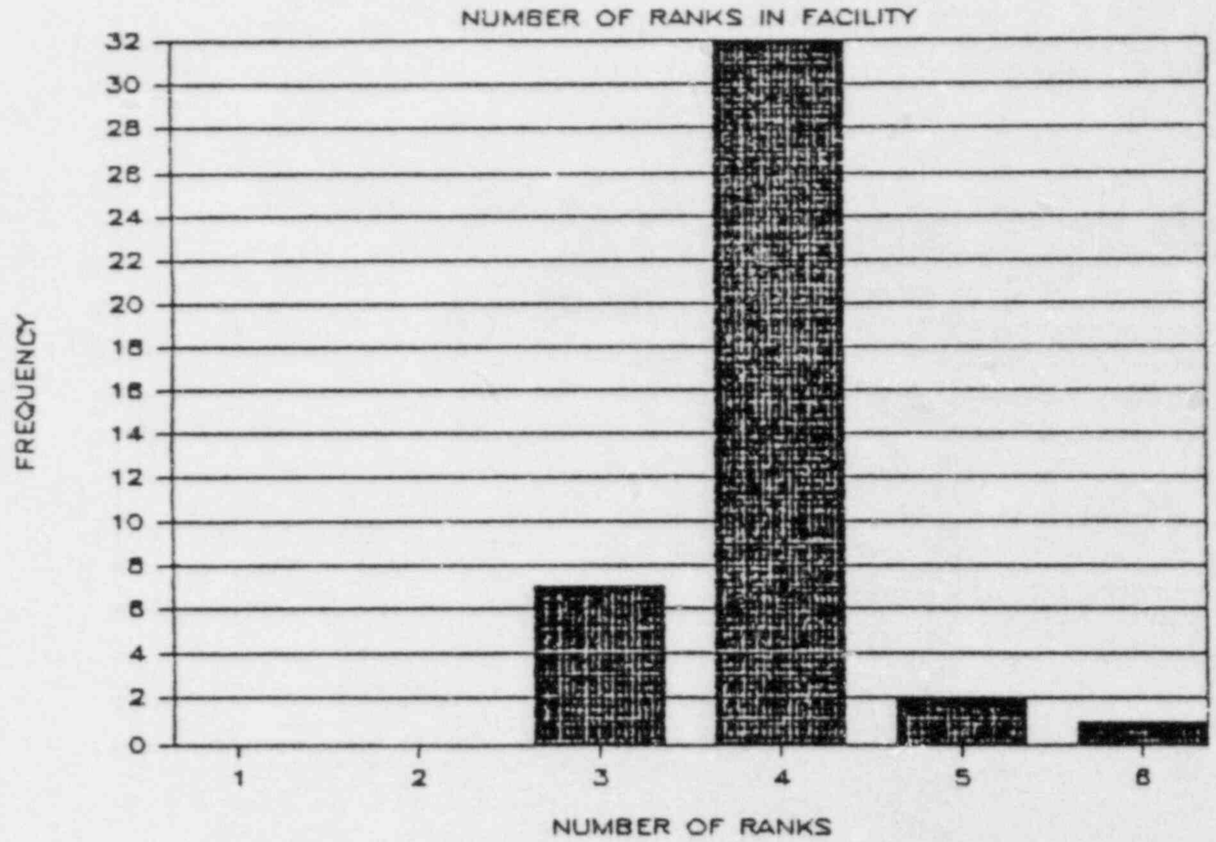
Figures B-12 and B-13 show the range in the deployment of leaders to followers for line and staff units respectively. The range in leader/lead ratios is substantial. For the line, the typical figure is 9, with a mean of 6.9 and a standard deviation of 3.3. The range for the deployment of leaders in staff units is even greater with a mode of 14, a mean of 12.4, and a standard deviation of 7.3. These two figures suggest that utilities may be relying upon quite different strategies in leader deployment. Some provide similar leader support across the system while others allow substantial variation.

FIGURE B-5



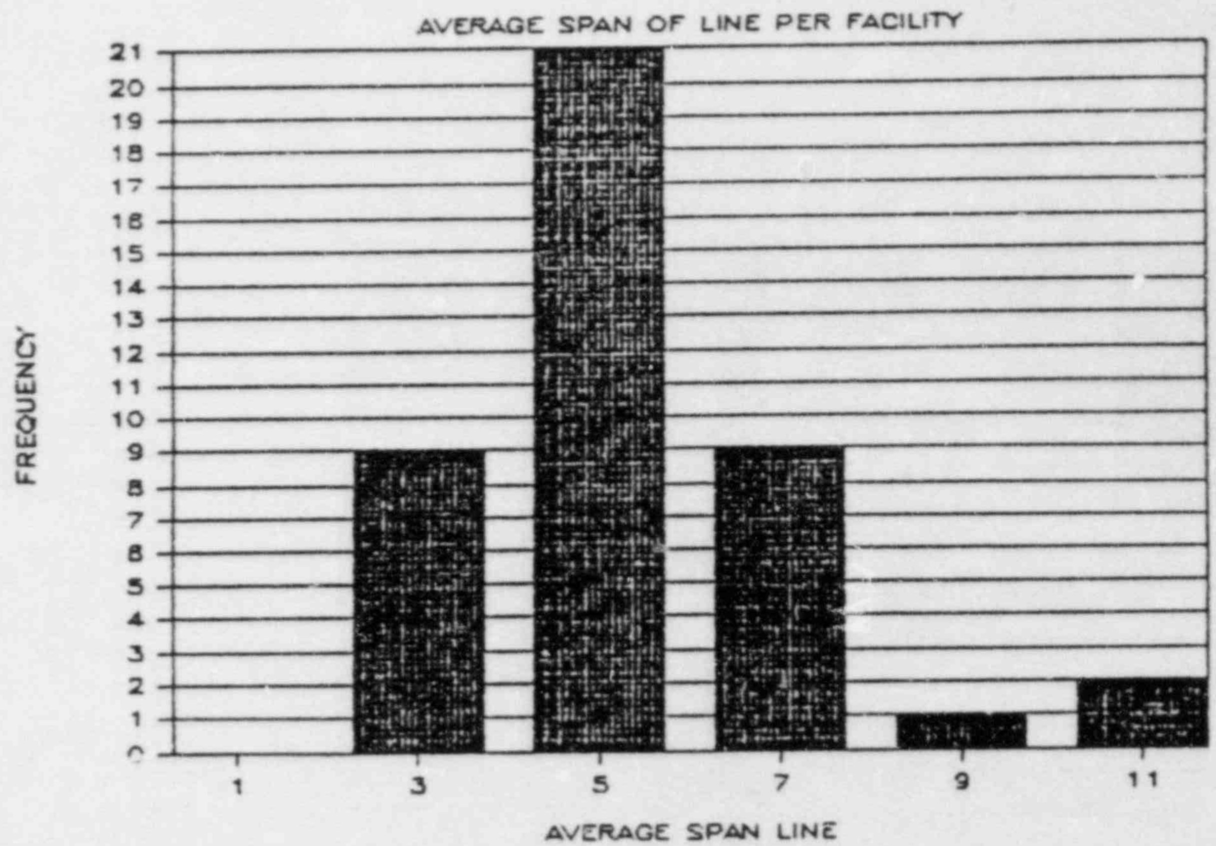
Mode = 5 $\bar{X} = 4.8$ S.D. = .8

FIGURE B-6



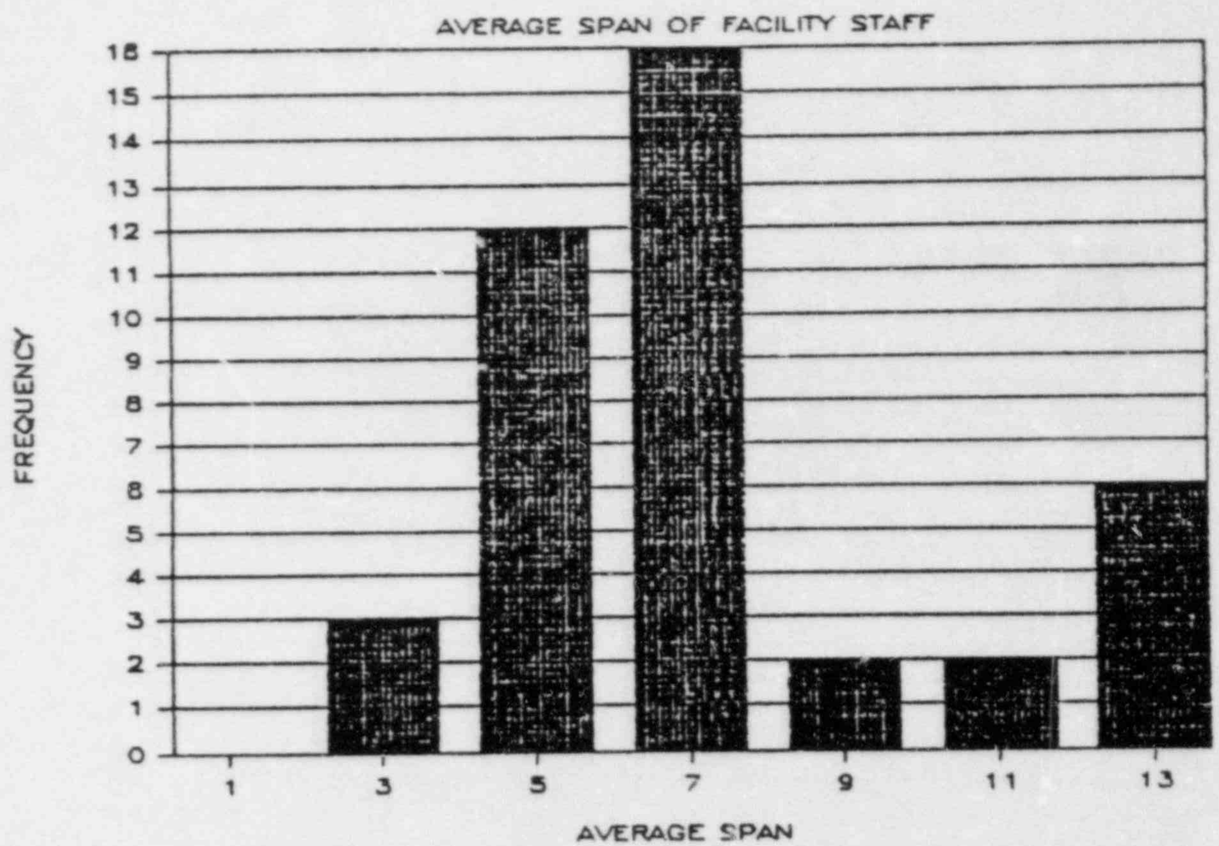
Mode = 3.5 $\overline{X}$ = 3.4 S.D. = .5

FIGURE B-7



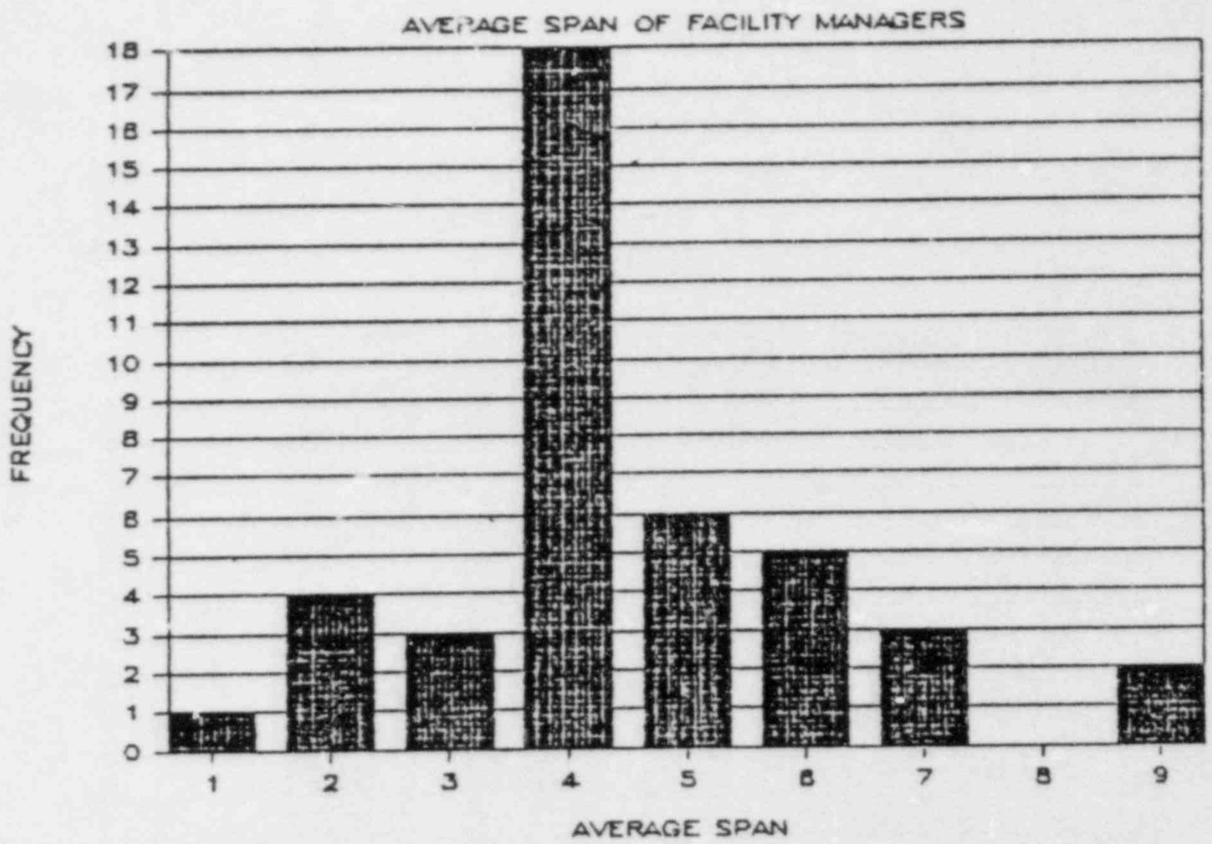
Mode = 4.6 $\overline{X}$ = 5.4 S.D. = 1.6

FIGURE B-3



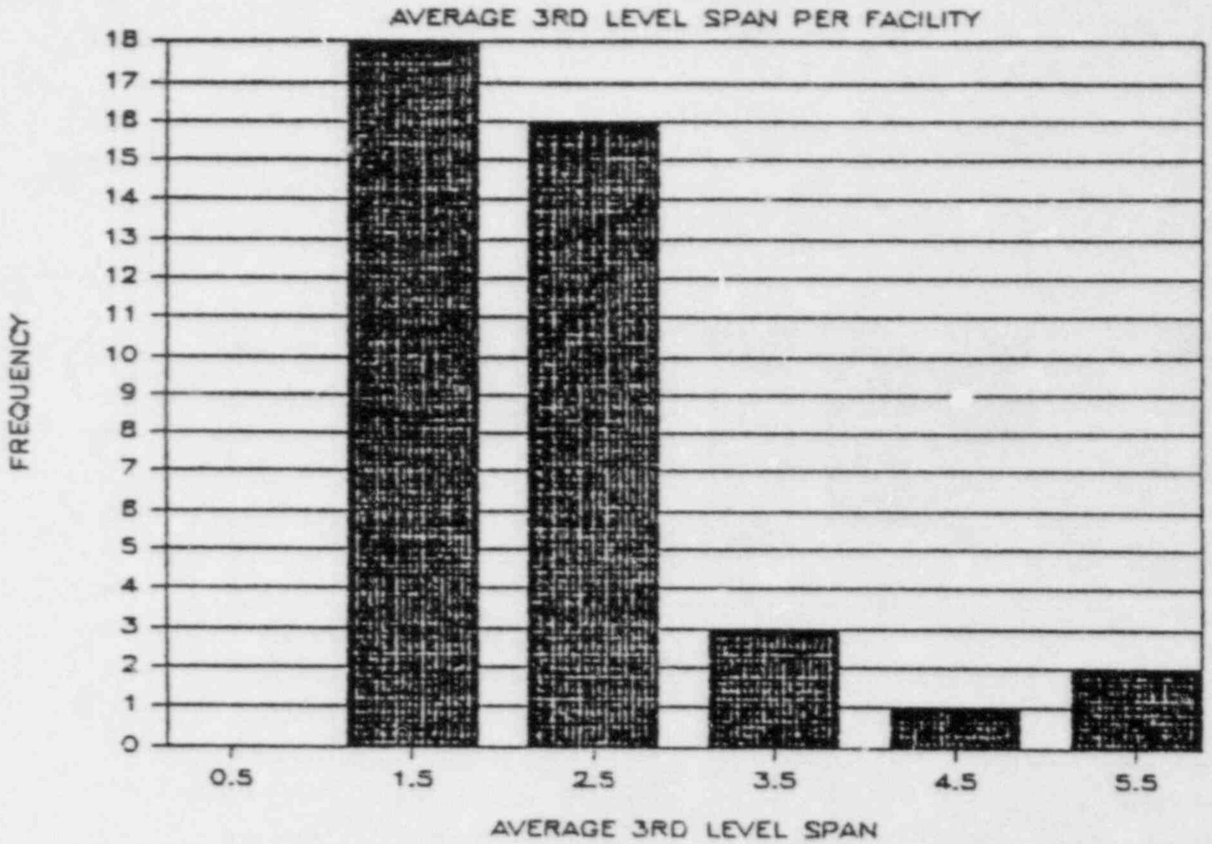
Mode = 6.5 $\overline{X}$ = 7.3 S.D. = 3.2

FIGURE B-9



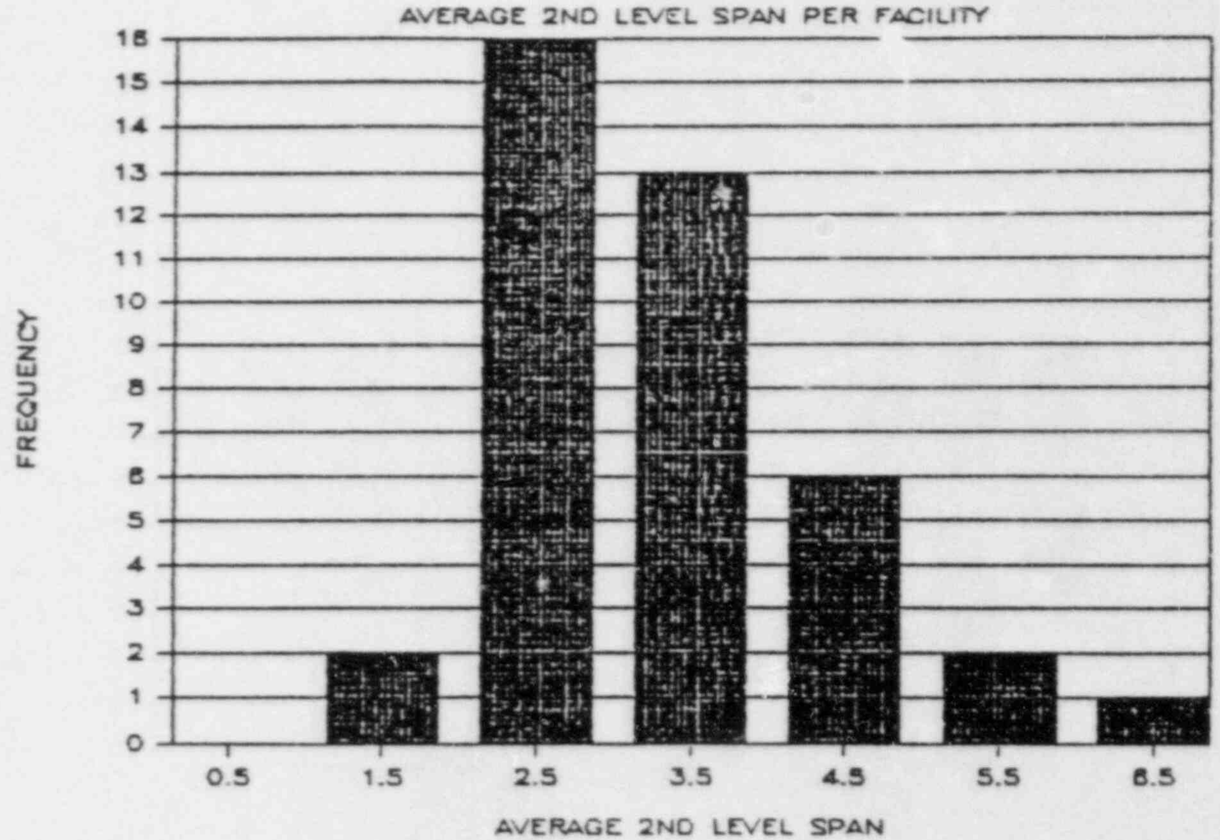
Mode = 4.0 $\overline{X}$ = 4.5 S.D. = 1.7

FIGURE B-10



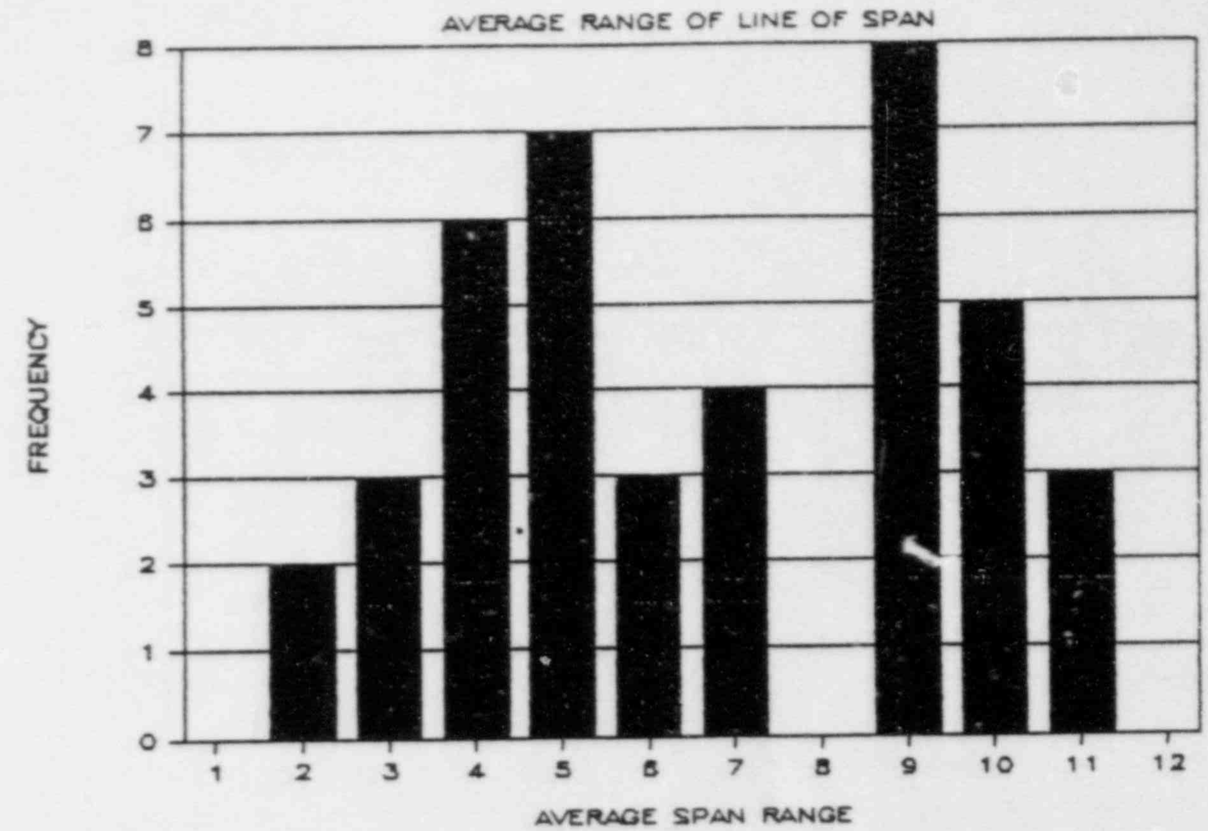
Mode = 2.0 $\overline{X}$ = 2.4 S.D. = .9

FIGURE B-11



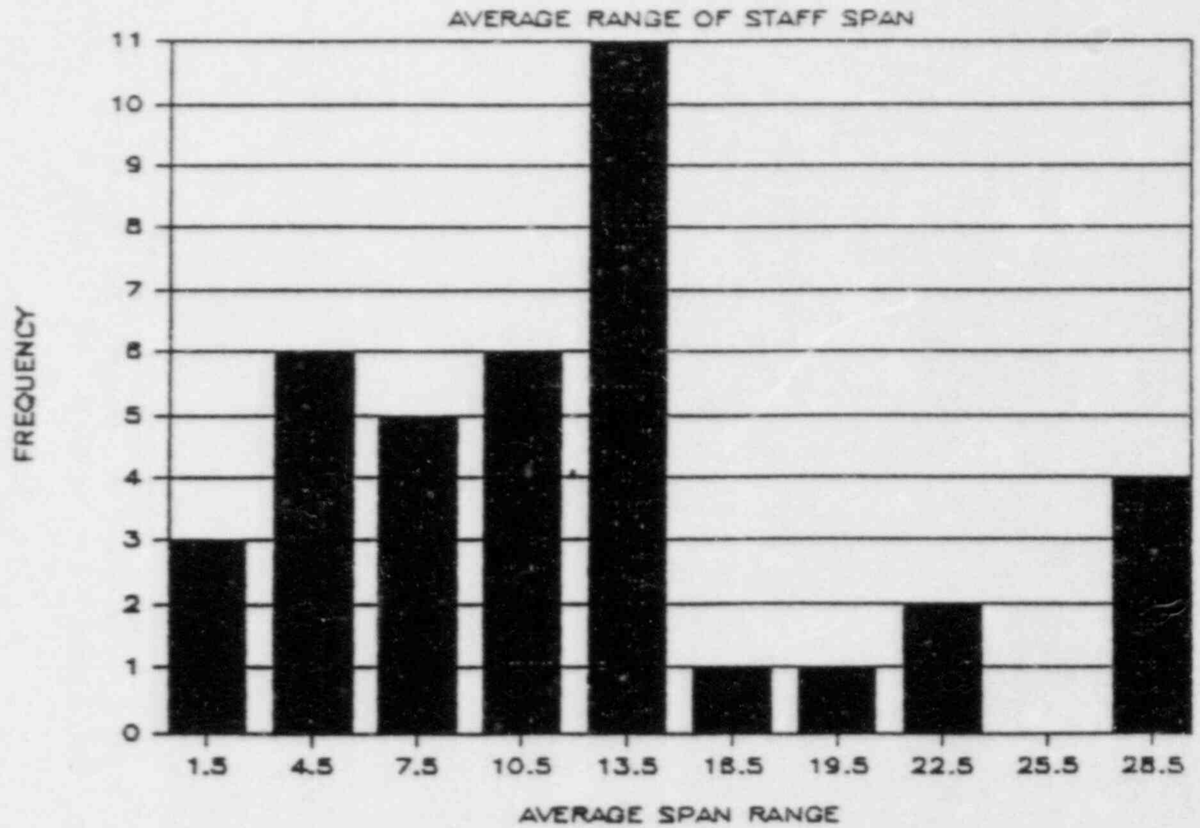
Mode = 3.3 $\overline{X}$ = 3.4 S.D. = 1.1

FIGURE B-12



Mode = 9 $\overline{X}$ = 6.9 S.D. = 3.3

FIGURE B-13



Mode = 14 $\overline{X}$ = 12.4 S.D. = 7.3

3. REACTOR ORGANIZATION

3.1 Breadth of Reactor Organization

Figures B-14 through B-17 provide data for questions of breadth for the reactor unit. In each case there is a substantial difference between the modal response of one and the mean response of about two. A basic difference, then, among the organizations in this sample is whether or not they decide to add staff units to the operations area. In general, however, the data suggest a very narrow formal organization for most of the sample plants with considerable consistency across the sample itself.

3.2 Depth of Reactor Organization

Figures B-18 and B-19 provide relevant data pertaining to the number of ranks in the chain of command. Both the longest and average number of ranks have the same mode of three levels. The means are close to the mode and there is comparatively little variation across the sample units. In short, the plants tend to organize the operations area in much the same way. And operations provides the greatest number of ranks to the systems as a whole.

3.3 Spans of Control and Consistency in Reactor Organization

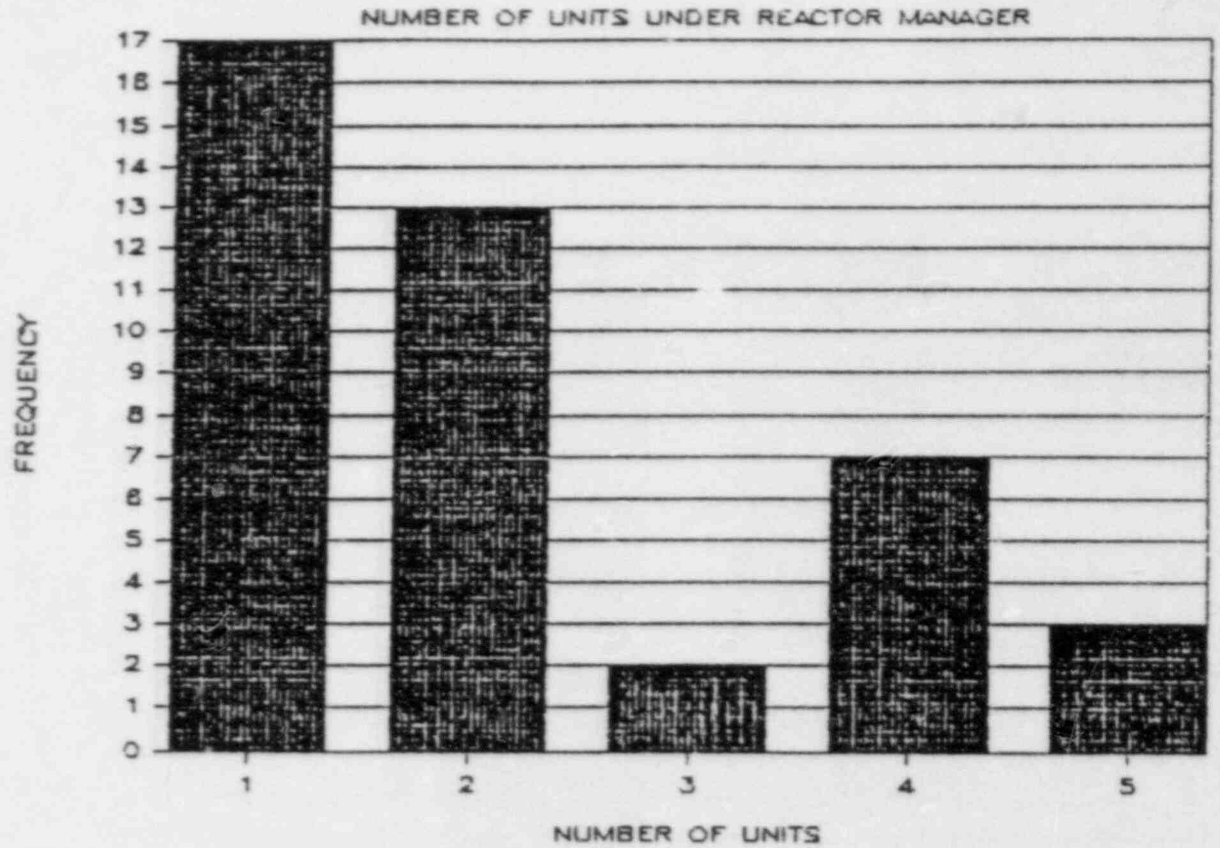
Figures B-20 through B-24 provide relevant data for leader/lead ratios. While the modal responses reflect an expected pattern of ever broader spans of control down the hierarchy, the averages and standard deviations suggest something slightly different. There appear to be two choices -- the inclusion of staff or limiting operations to operators and their immediate superiors.

Figures B-25 and B-26 complete the figures for this section. Essentially, they confirm the option of utilities to either include or not include staff units in their operations area.

4. SUMMARY

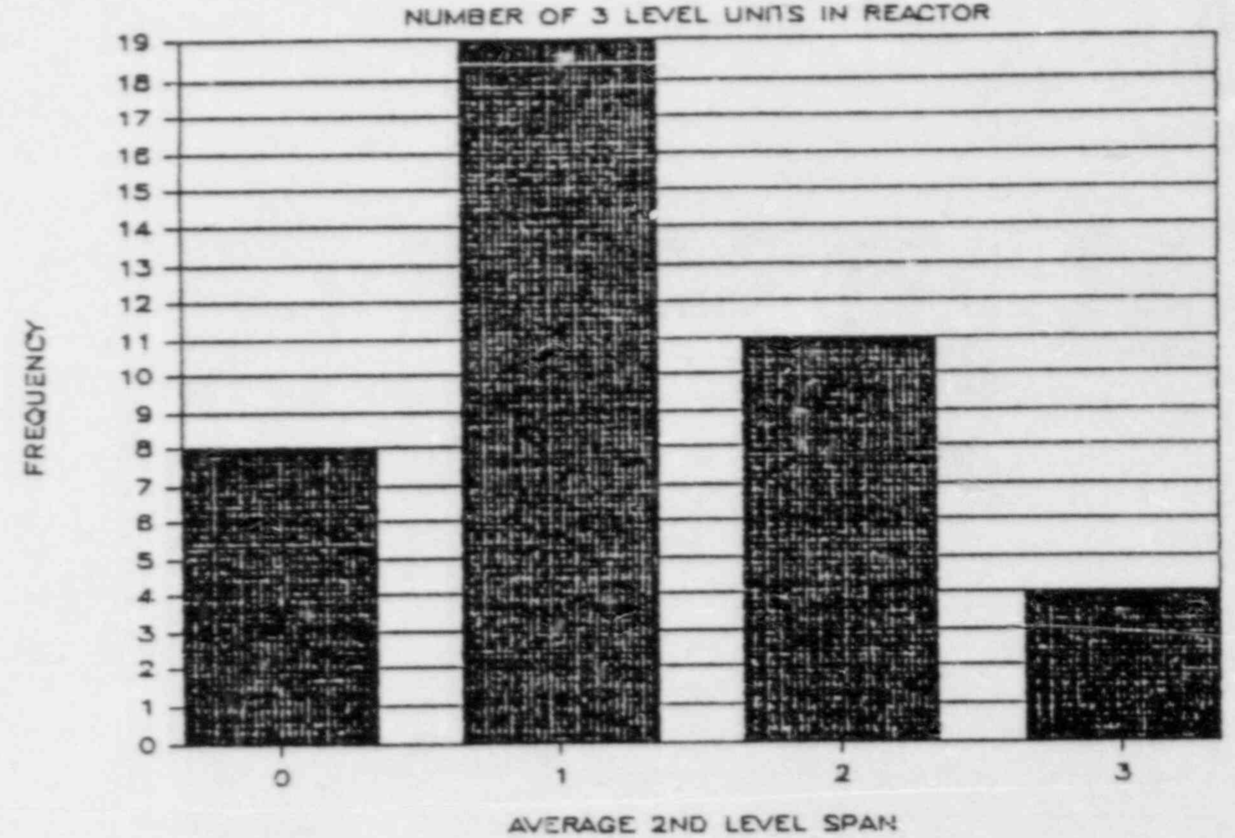
The descriptive data, when analyzed as a set, suggest an interesting series of patterns for formal organizaton. The reactor operations parts of the plant appear narrow and deep and there is relatively little variation across the sample. Leader/lead ratios suggest comparatively few employees per manager, particularly for operations. In contrast, the formal organization for staff components appears broader, not as deep, and with substantial variation across the sample. Further, there also appears to be a strategic design decision regarding the placement of staff units within the operations area. While the typical plant does not contain staff components within reactor operation, a few plants do.

FIGURE B-14



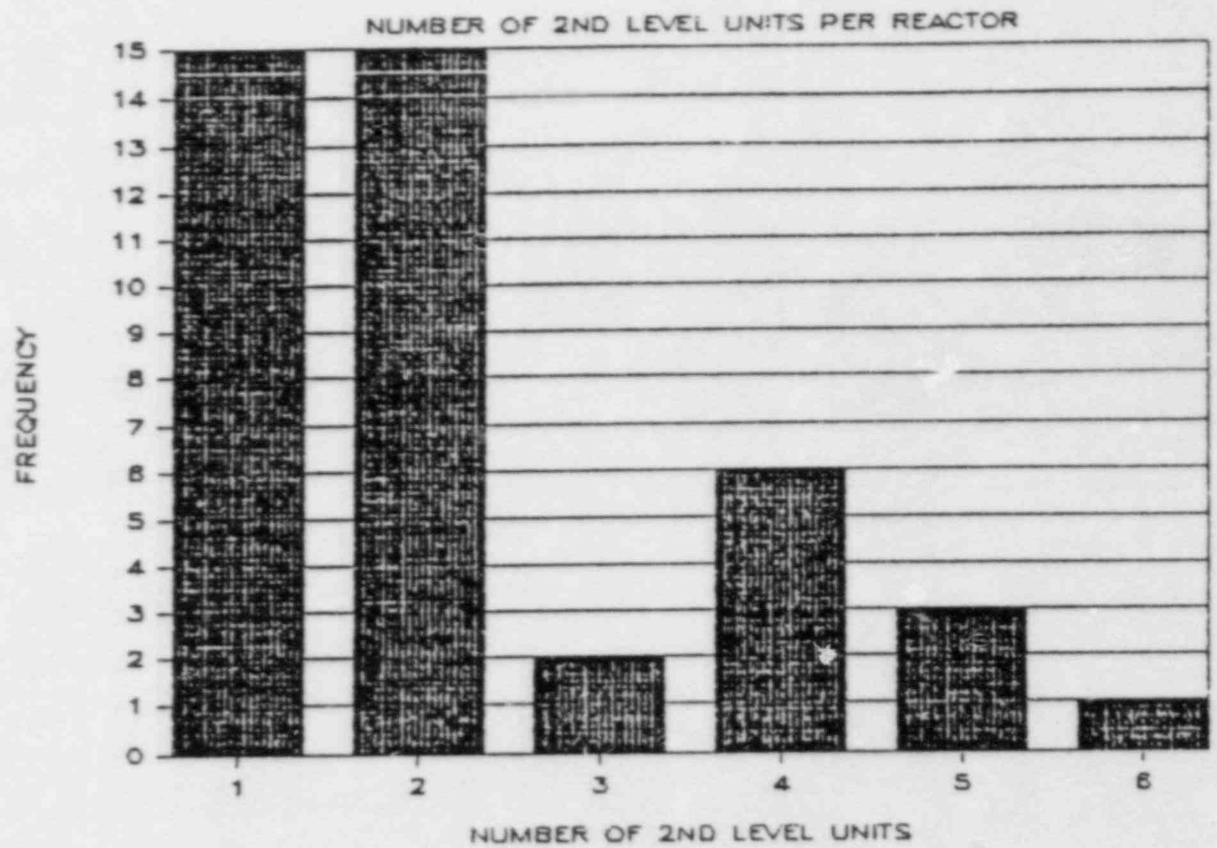
Mode = 1.0 $\overline{X}$ = 2.2 S.D. = 1.3

FIGURE B-15



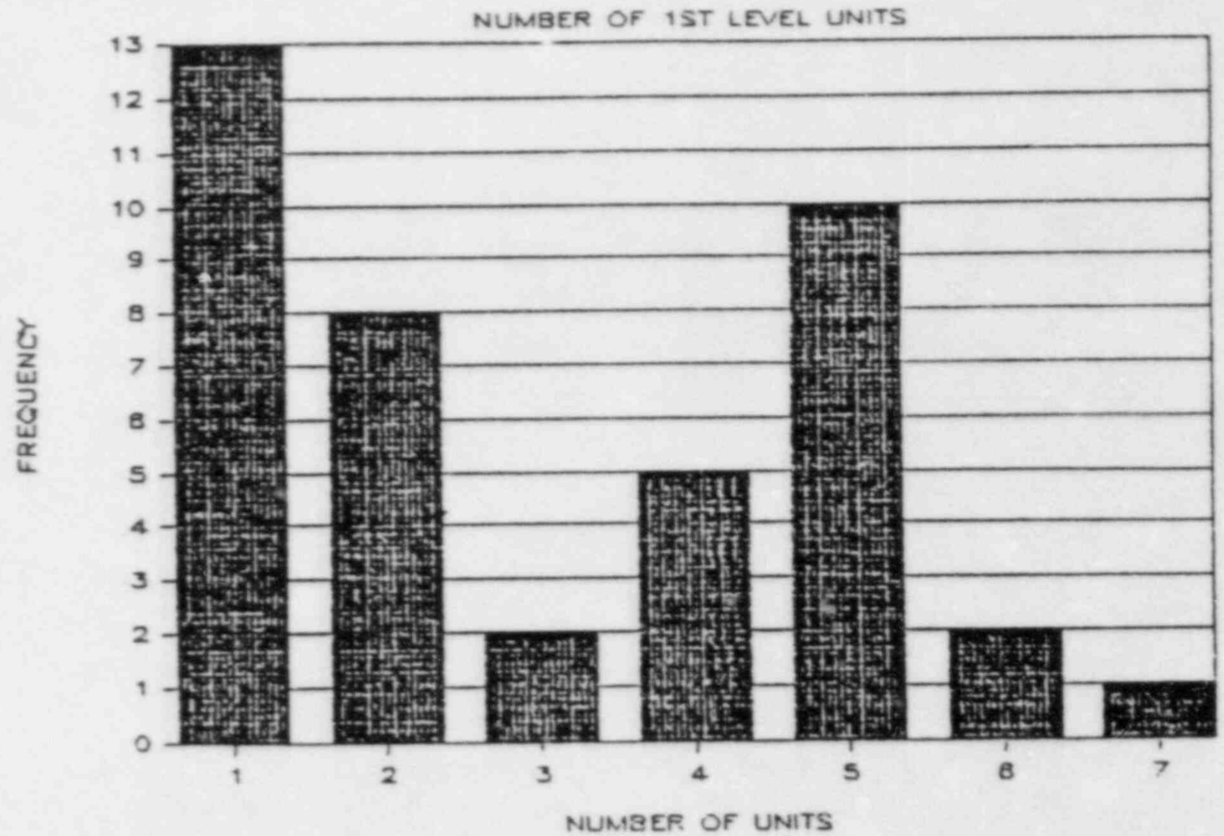
Mode = 1.0 $\overline{X}$ = 1.2 S.D. = .9

FIGURE B-16



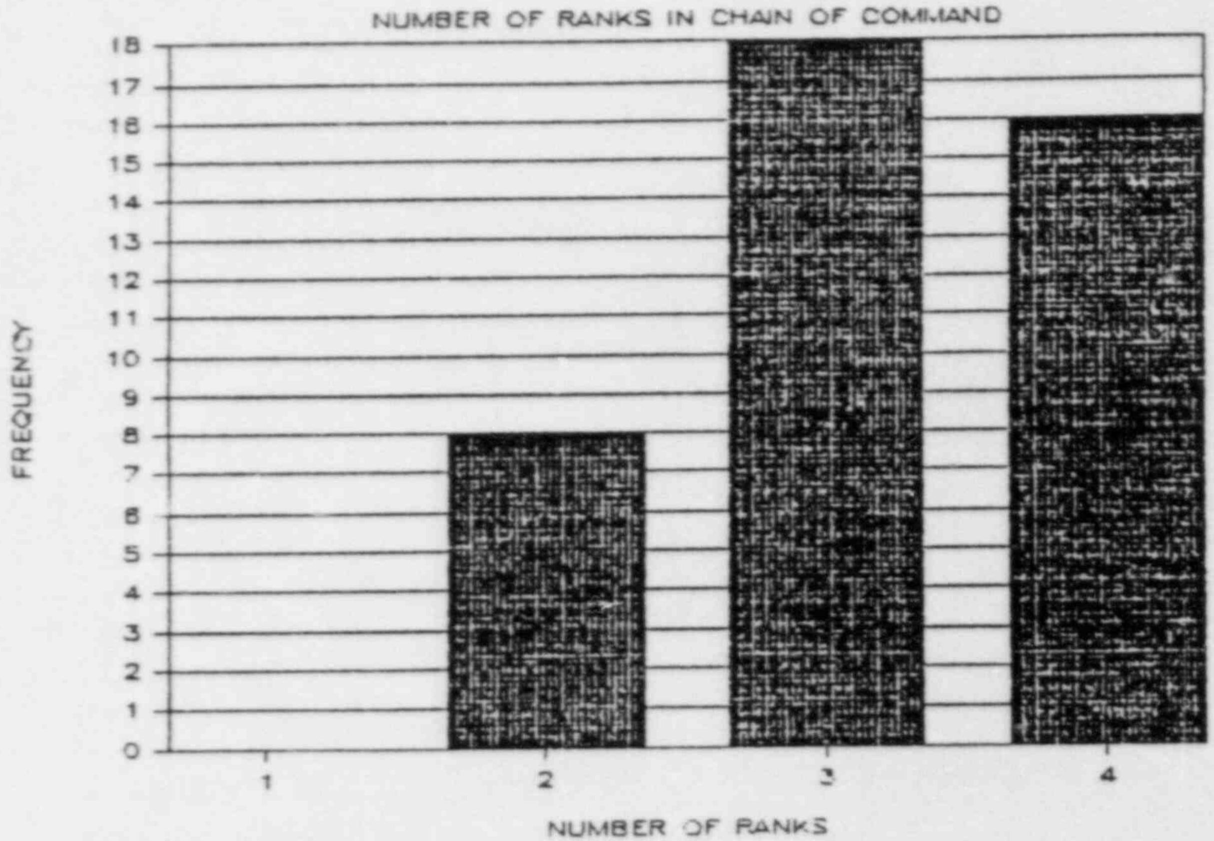
Mode = 1.0 $\overline{X}$ = 2.2 S.D. = 1.4

FIGURE B-17



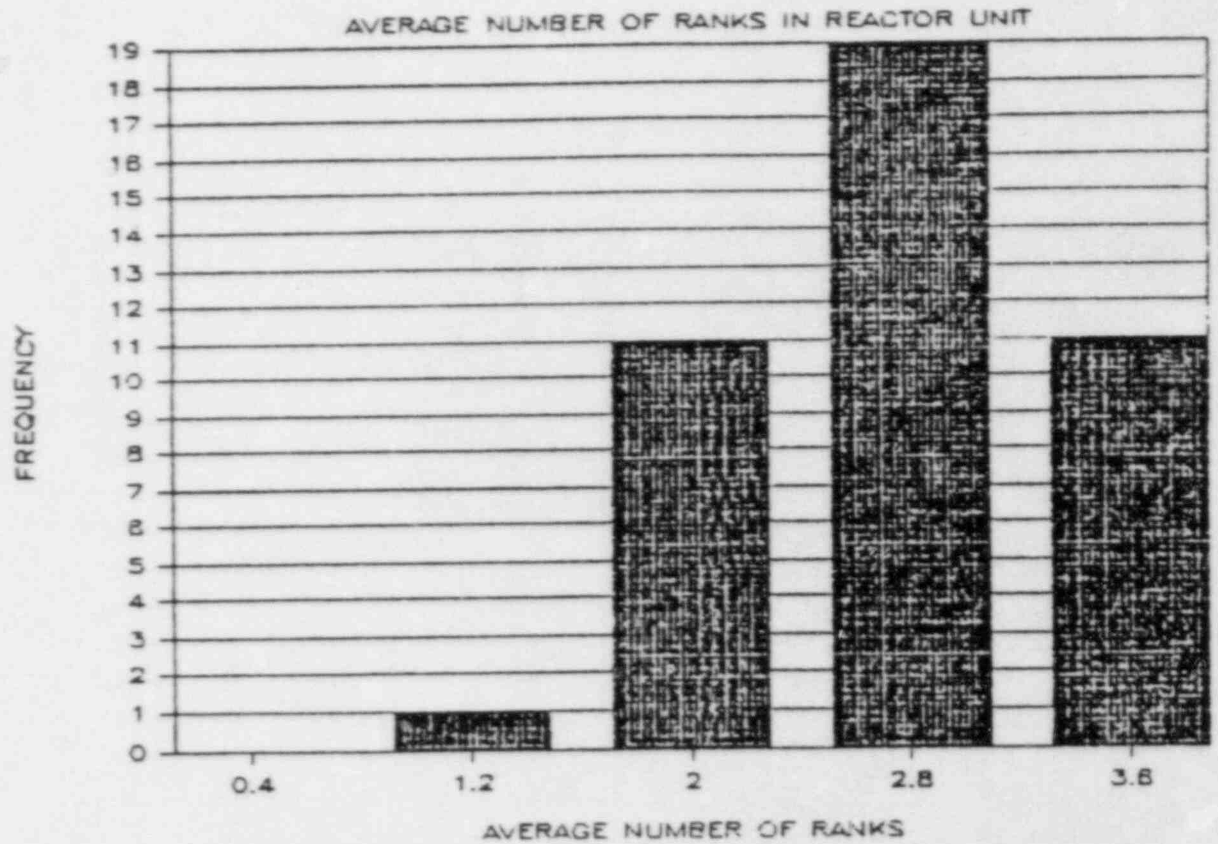
Mode = 1.0 $\overline{X}$ = 3.2 S.D. = 2.4

FIGURE B-18



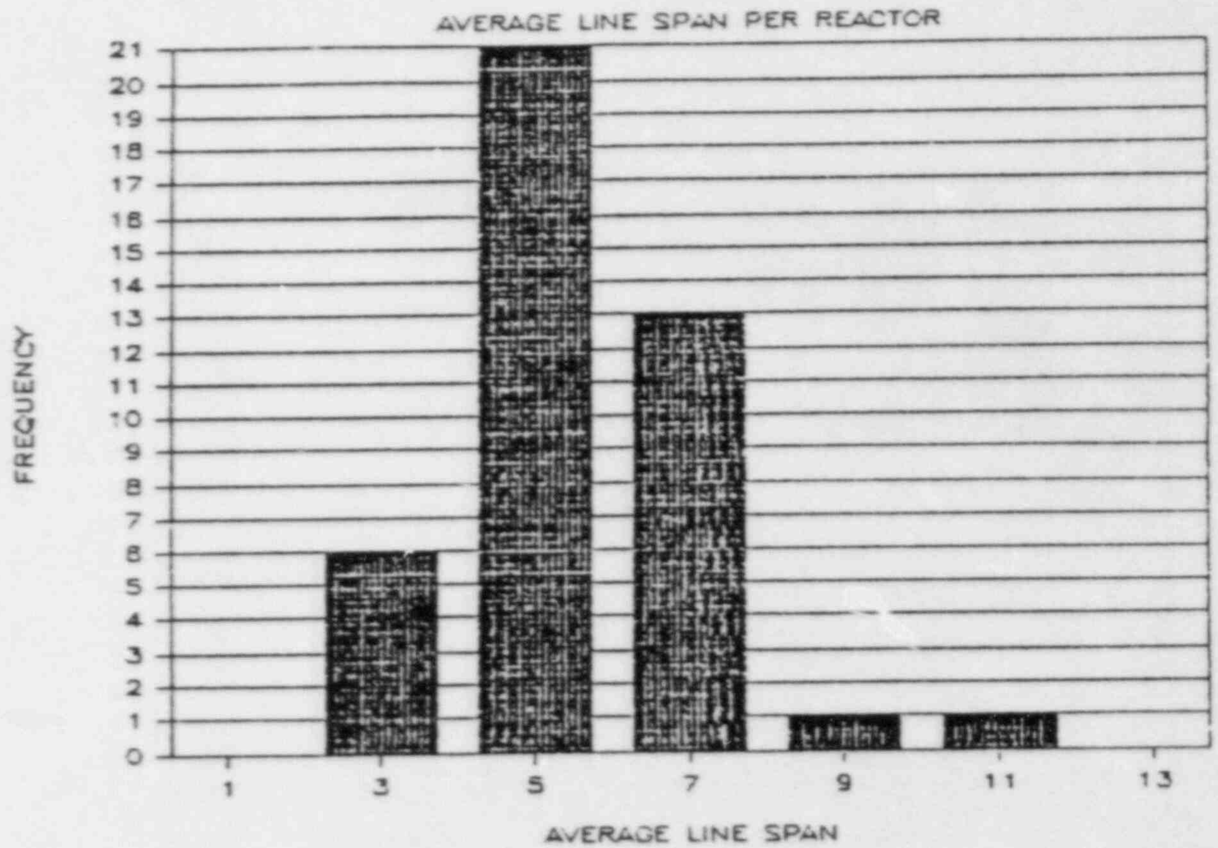
Mode = 3.0 $\bar{X}$ = 3.1 S.D. = .7

FIGURE B-19



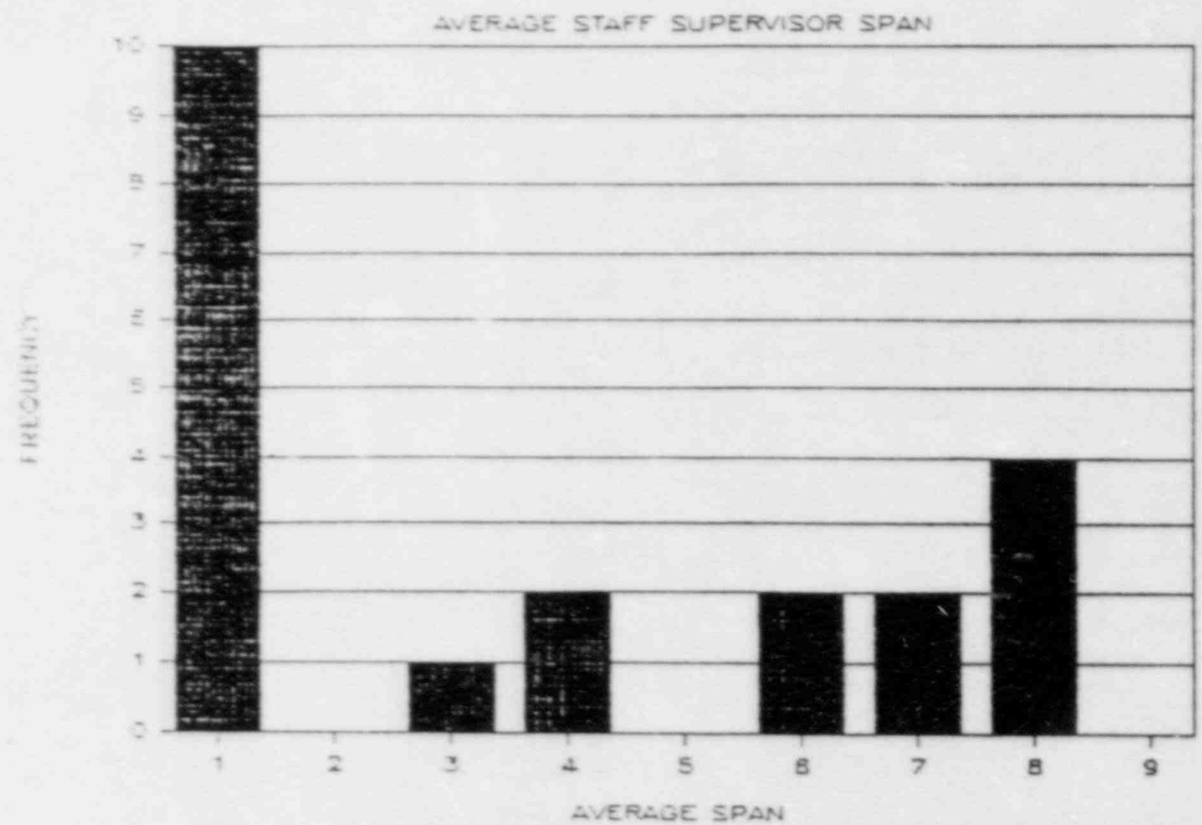
Mode = 3.0 $\overline{X}$ = 2.8 S.D. = .6

FIGURE B-20



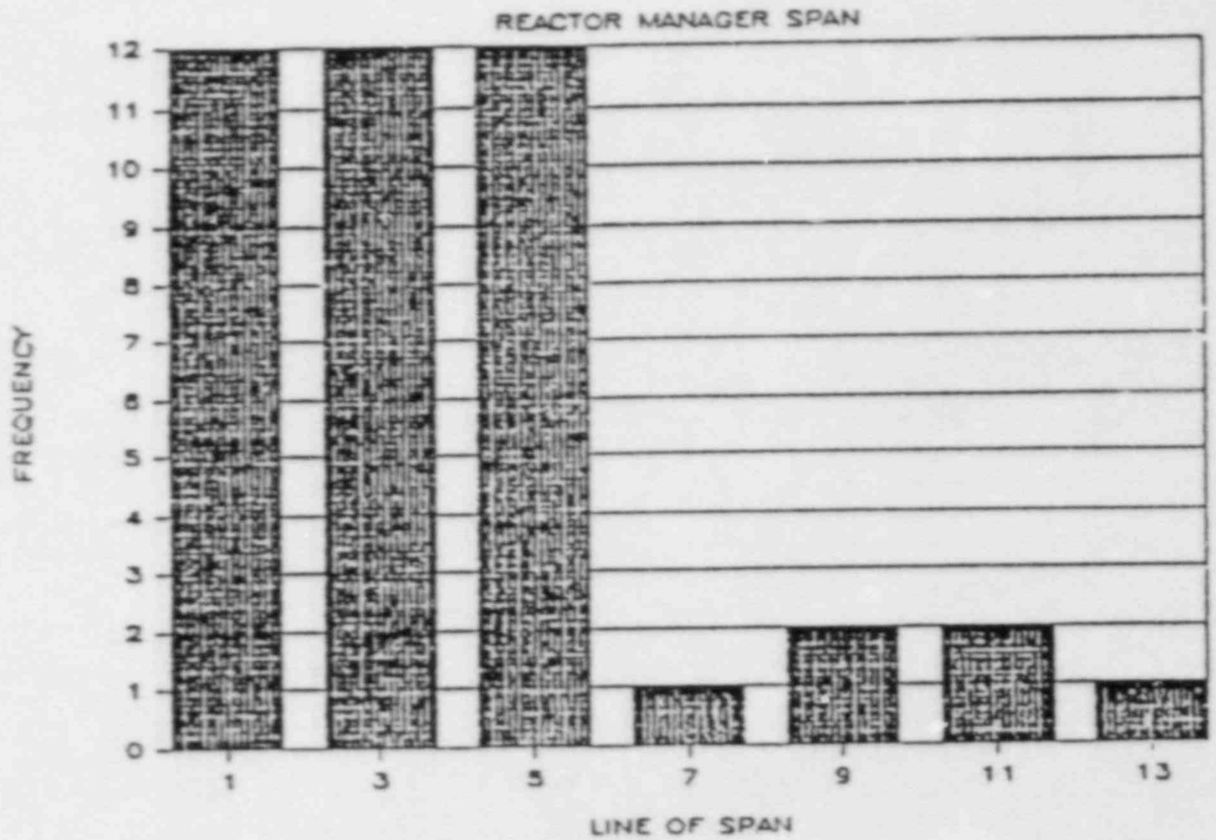
Mode = 4.8 $\bar{X}$ = 5.6 S.D. = 1.5

FIGURE B-21



AVERAGE STAFF SUPERVISOR SPAN
FREQUENCY
0
1
2
3
4
5
6
7
8
9
10
AVERAGE SPAN
1
2
3
4
5
6
7
8
9

FIGURE B-22



Mode = 2.0 $\overline{X}$ = 4.6 S.D. = 3.0

FIGURE B-23

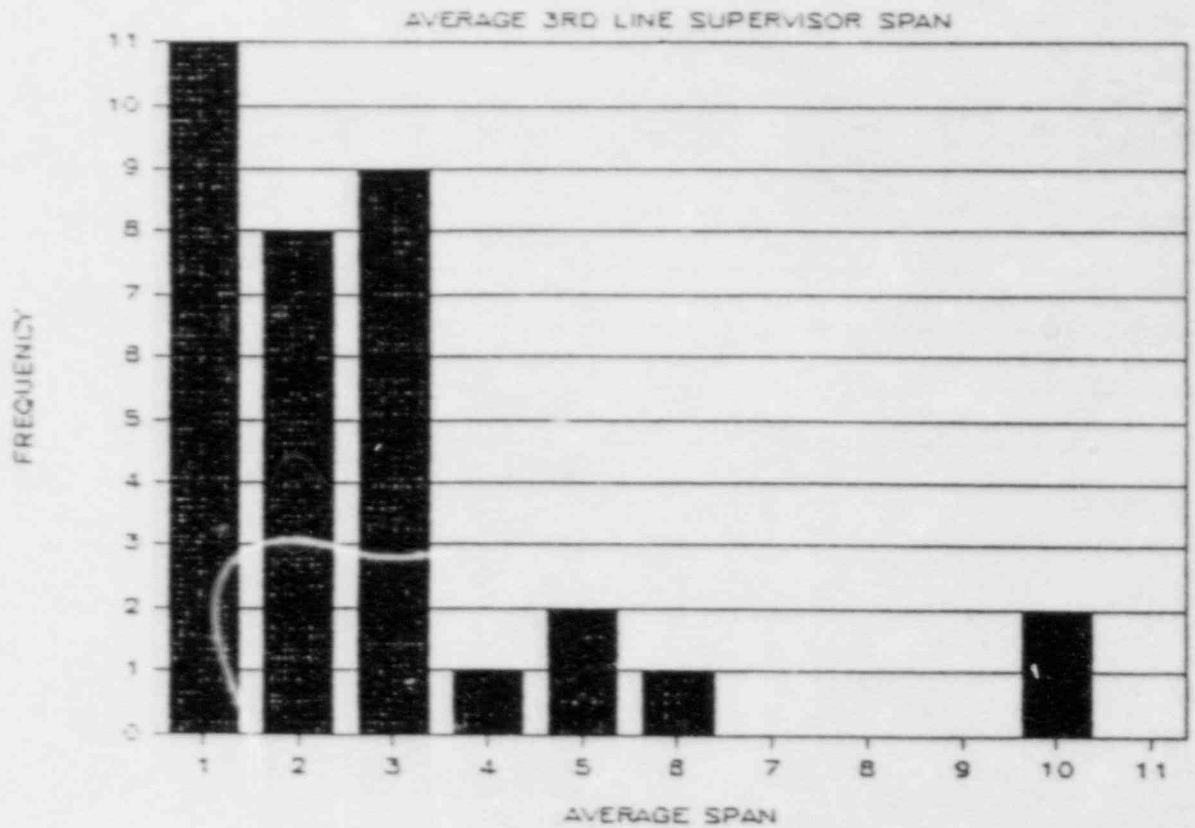
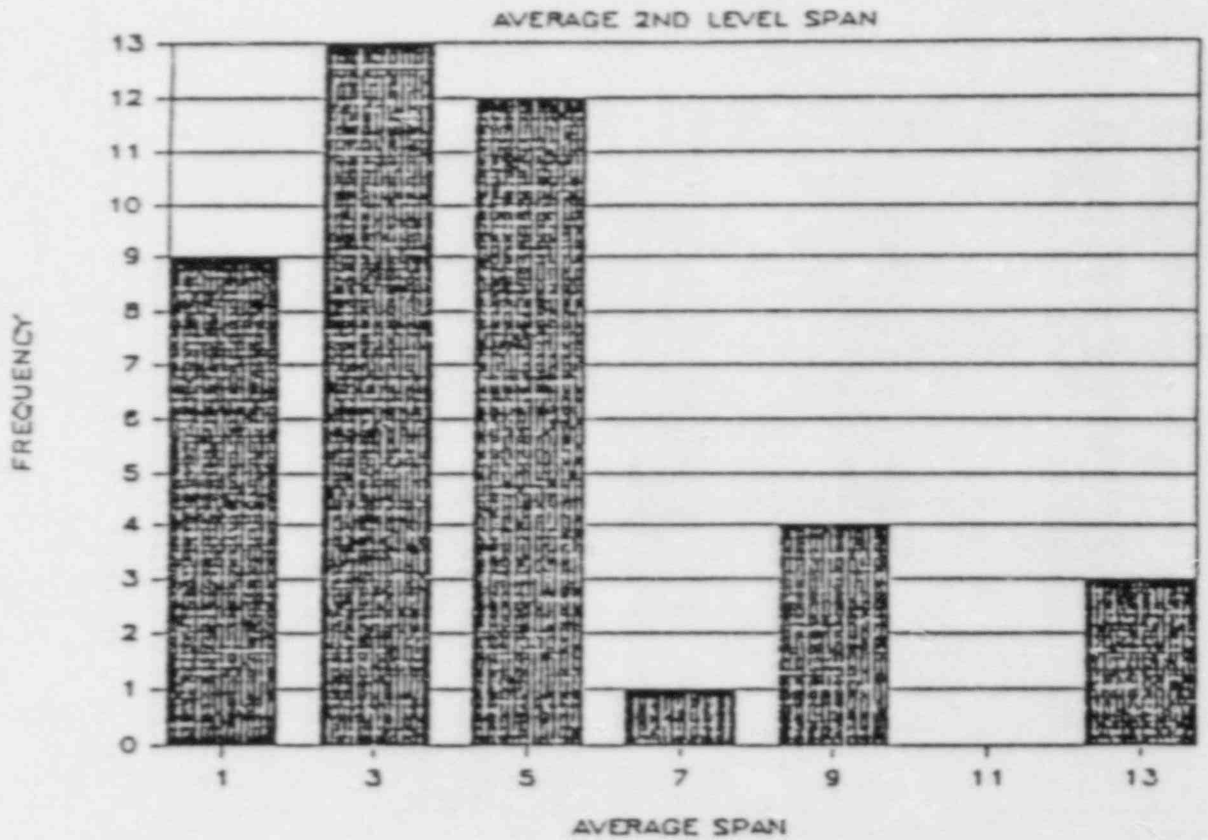
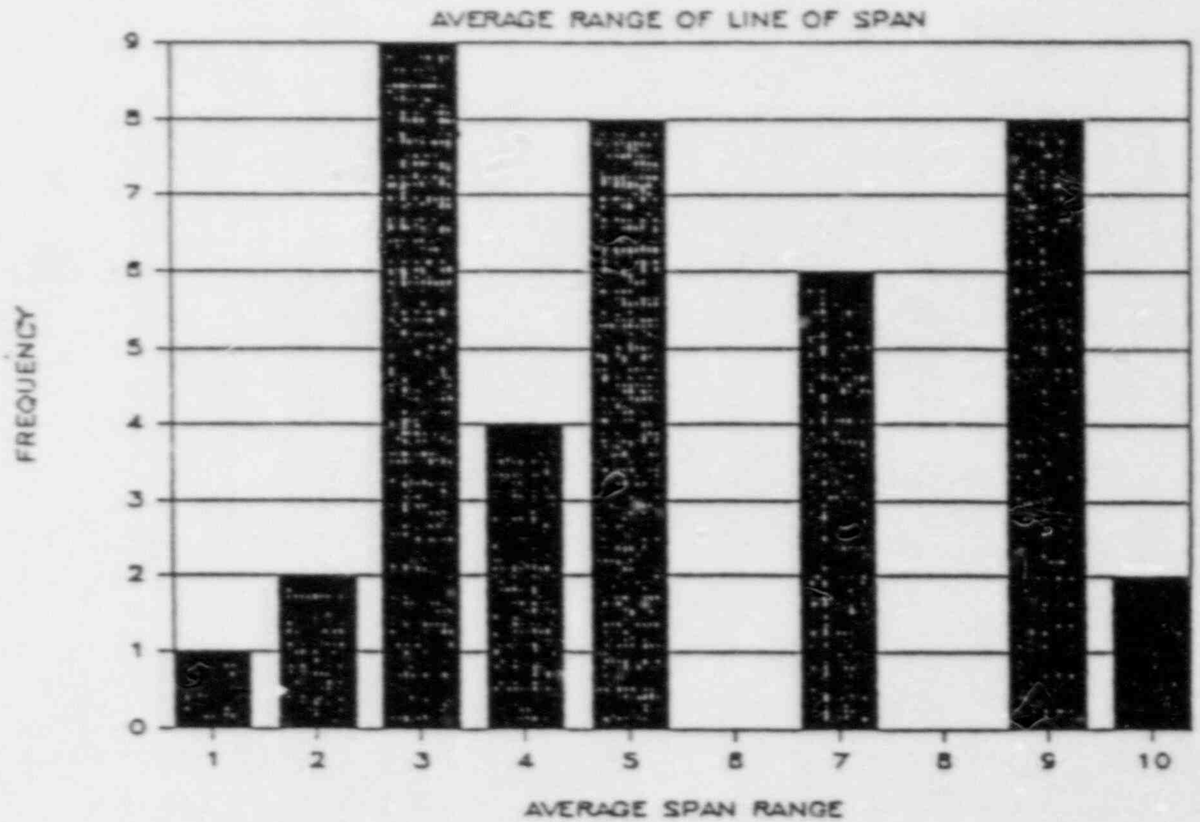


FIGURE B-24



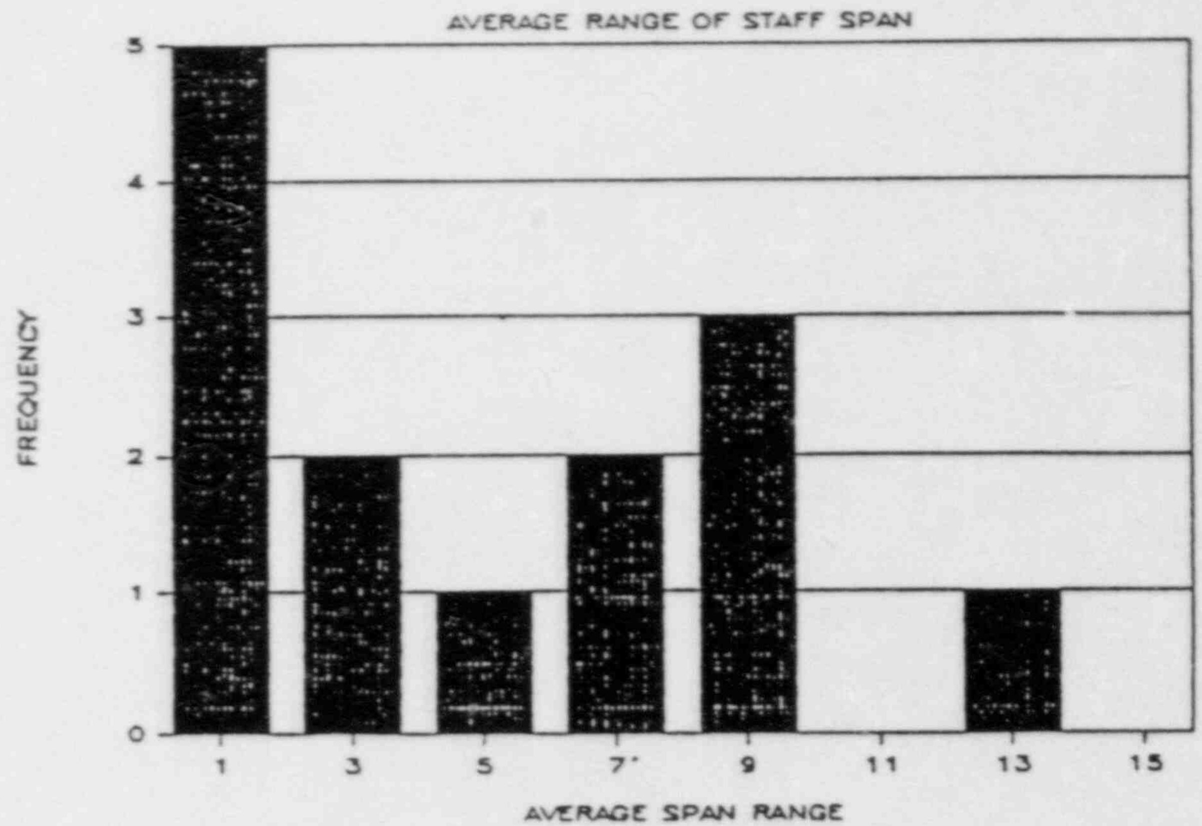
Mode = 5.0 $\overline{X}$ = 4.8 S.D. = 3.3

FIGURE B-25



Mode = 3.0 $\overline{X}$ = 5.8 S.D. = 3.2

FIGURE B-26



Mode = 0 $\overline{X}$ = 5.0 S.D. = 4.6

There also appear to be major differences across the sample plants in the number of individuals placed within either the tall, narrow structure of the line components, or the broader structure of the staff components. This partially accounts for the substantial variation in leader/lead ratios.

The data are consistent with the following dominant profile. It appears that the reactor components are bureaucratically oriented with comparatively few administrative units to levels (tall and narrow). In slight contrast, staff units appear somewhat less bureaucratically structured with more administrative units to levels (comparatively flat and broad).

When comparing the facility as a whole with the reactor operations area, it is quite obvious there is more consistency in the formal structure for operations. Standard deviations are considerably smaller. It is also quite obvious that one of two patterns is selected in the operations area. Either staff are included or they are not. This strategic choice accounts for the fact that mean values for the reactor operations area statistics are generally higher than modal responses.

The pattern suggested above is quite unusual for a highly automated, continuous process manufacturing operations staffed by professionals and highly trained technicians (e.g. chemical factory). One would have expected more units (particularly more first level units), fewer levels of command, broader spans of control, and less diversity in the treatment of line and staff units. The pattern is more similar to large-scale, quasi-automated batch processing organizations (e.g. auto manufacturing). Further, it is quite interesting to note that the breadth of the structure and its depth do not appear sensitive to an apparent increase in the number of personnel. That is, breadth and depth were similar even where there was wide variation in leader/lead ratios. This suggests that some aspects of the formal structure employed to organize the work are not particularly sensitive to increases in staff size.

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16. ABSTRACT (200 words or less)

This report contains an analysis of the relationship between selected aspects of organizational structure and the safety related performance of nuclear power plants. The report starts by identifying and operationalizing certain key dimensions of organizational structure that may be expected to be related to plant safety performance. Next, indicators of plant safety performance are created by combining existing performance measures into more reliable indicators. Finally, the indicators of organizational structure were related to the indicators of plant safety performance using correlational and discriminant analysis. The overall results show that plants with better developed coordination mechanisms, shorter vertical hierarchies, and a greater number of departments tend to perform more safely.

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AN INITIAL EMPIRICAL ANALYSIS OF NUCLEAR POWER PLANT ORGANIZATION AND ITS EFFECT ON SAFETY PERFORMANCE

NOVEMBER 1984