WNR

Board Notification

LICENSEE REGULATORY PERFORMANCE EVALUATION

February 1979

U. S. Nuclear Regulatory Commission

8002100090 P

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing
Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Members of the Boards:

Enclosed is information relating to Staff efforts with respect to licensee regulatory performance evaluation.

Copie this same information will be provided to the parties in approp cases.

Sincerely,

William D. Paton

William D. Paton
Counsel for NRC Staff

Enclosures as stated to
Each Member of the Licensing Board Panel
Each Mem er of the Appeal Board Panel

CONTENT

1. Two-page letter dated November 6, 1978 from G.C. Gower to D.B. Vassallo;

2. One-page letter dated November 1, 1978 from H.D. Thornburg to G.C. Gower;

3. Ten-page "Policy Session Item" from J.G. Davis to the Commissioners dated October 25, 1978;

4. Draft Report "An Evaluation of the Nuclear Safety-Related Management Performance of NRC Operating Reactor Licensees During 1976" dated February 1977;

5. Memorandum for Ernst Volgenau from E. Morris Howard dated September 26, 1977;

6. "Individual Site Ratings" from the "IE Employee Survey on Evaluation of Licensees" dated April 1978;

7. "Licensee Pefrormance Evaluation" Teknekron, Inc.;

8. Memorandum for D.B. Vassallo from S.E. Bryan dated December 1978.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 0 6 1978

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: George C. Gower, Acting Executive Officer for Operations Support, IE

SUBJECT: INFORMATION TO BE CONSIDERED FOR BOARD NOTIFICATION - LICENSEE REGULATORY PERFORMANCE EVALUATION (LRPE)

The enclosed information is being forwarded for consideration and possible Board notification. Mr. Thornburg's memorandum, also enclosed, provides information on the various aspects of the LRPE efforts and discusses current plans for a trial program. Questions on LRPE should be discussed with Mr. Thornburg.

Please note that the enclosed information is to be treated as predecisional, that is, withheld from public disclosure. Steps are being taken to obtain clearance from the Commission for release of this information. Prior to taking any action that would in effect release this information to the public, we request that you contact this office for the current status of the clearance effort now in progress.

We also request to be informed whether or not this matter is sent forward to the Boards.

G. C. Gower
George C. Gower
Acting Executive Officer
for Operations Support, IE

Enclosures:
1. Memo HDThornburg to GCGower dtd 11/1/78
2. SECY-78-554
3. Draft Report - An Evaluation of the Nuclear Safety-Related Management Performance of NRC Operating Reactor Licensees During 1976 dtd 2/77
4. Memo EMHoward to EVolgenau dtd 9/26/77 - Draft Report - Licensee Inspection and Enforcement Indicators

5. Report - Individual Site Ratings
   from the IE Employee Survey
   on Evaluation of Licensees
   dtd 4/78
6. Report - Licensee Performance
   Evaluation by Teknekron, Inc.
   dtd 5/78

cc w/o enclosures:
J. G. Davis
H. D. Thornburg
N. C. Moseley
IE Files

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 1 1978

MEMORANDUM FOR: G. C. Gower, Acting Executive
Officer for Operations Support

FROM: H. D. Thornburg, Director
Division of Reactor Construction Inspection

SUBJECT: BOARD NOTIFICATION - LICENSEE REGULATORY PERFORMANCE EVALUATION

In accordance with Manual Chapter 1530, the enclosed information is forwarded for notification of the appropriate NRC Hearing Boards. We believe that the information will be of interest to the Hearing Boards.

The boards should be made aware of the past efforts in Licensee Regulatory Performance Evaluation (LRPE), future plans, and limitations of the methods attempted as outlined below and as contained in the enclosed documents.

IE has been working to develop techniques for evaluating the regulation performance of licensees for several years. Three general approaches have been made to the problem. One employed statistical treatment of weighted noncompliance and LER data. The second employed trend analysis of LER data while the third involved a compilation of judgments made by regional managers and inspectors. These matters have been described to the Commission along with IE plans to develop an approach to Licensee Regulatory Performance Evaluation (LRPE) using the best elements of the above three general approaches in an integrated fashion in SECY 78-554 dated October 25, 1978. A copy of SECY 78-554 is enclosed along with copies of the documents referenced in that paper.

It should be noted in SECY 78-554 that each of the methods for LRPE attempted today by IE has technical shortcomings and that there is not staff unanimity regarding the usefulness of the results of the individual approaches. It is the position of IE management that the methods are imperfect but that they do provide insights into licensee regulatory performance. It is also the position of IE management that a system for licensee regulatory performance evaluation should contain a combination of quantitative and qualitative information and judgment. As indicated above, IE will initiate a trial program for LRPE upon receipt of Commission approval. The staff has been treating this information as predecisional pending the Commission's decision in this matter.

Harold D. Thornburg

Harold D. Thornburg, Director
Division of Reactor Construction Inspection

Enclosures: As stated

October 25, 1978

SECY-78-554

UNITED STATES
NUCLEAR REGULATORY COMMISSION

POLICY SESSION ITEM

For: The Commissioners

From: John G. Davis, Acting Director
Office of Inspection and Enforcement

Thru: Executive Director for Operations TAR for L.V.G.

Subject: LICENSEE REGULATORY PERFORMANCE EVALUATION

Purpose: The purpose of this paper is to inform the Commission regarding the status of efforts by the Office of Inspection and Enforcement in licensee regulatory performance evaluation and to obtain Commission approval of a two year trial program.

Discussion: IE has been working to develop techniques for evaluating the regulatory performance of NRC licensees for several years, with intensified effort over the last two years. "Regulatory performance," is meant to convey the ability of the licensee to meet regulatory requirements and to avoid reportable events that appear to be directly under the control of the licensee. "Regulatory performance" does not involve reliability, availability, earnings, or other measures which may be used to measure performance.

Licensee Regulatory Performance Evaluation (LRPE) is the effort to evaluate the regulatory performance of licensees on a national basis. It has as its objectives:

. Identification of factors that lead to different levels of regulatory performance.

. Effective and efficient use of NRC inspection resources.

Information from the evaluation process also can be used to evaluate aspects of the NRC inspection program.

Contact:
H. D. Thornburg, RCI
49-28484

The basic method of LRPE is to identify licensees whose regulatory performance is most different from the majority of licensees in the same class. These "different" licensees are examined on a case-by-case basis to identify the characteristics that lead to the differences. Actions then can be taken, if needed, to upgrade licensee regulatory performance. The thrust of LRPE is an upgrading, as appropriate, of performance.

The enclosed paper, entitled "Licensee Regulatory Performance Evaluation," defines the concept of licensee regulatory performance, describes why IE wants to evaluate it, and suggests the uses that may be made of the results. The paper also describes the evaluation approaches that IE has considered and offers some ideas how IE may develop and use an "integrated methodology" that incorporates selected aspects of each of the three methods considered to date. Finally, the paper provides a summary of value-impact considerations and plans and schedules for future actions.

Defining and agreeing upon the reasons for LRPE and suitable methods for its conduct have been difficult. Concepts and positions have been modified as new insights are developed. Staff agreement still has not been achieved. The results of efforts in LRPE have not been made public. No public nor industry comments have been requested. IE management believes that the potential benefits--resource management and performance upgrading--are sufficient to move forward into a trial program of LRPE.

IE proposes to implement a trial program for evaluating the operating reactor licensees on the basis of 1978 and 1979 data. As the program proceeds, IE will monitor its results to identify changes which may be needed. An interoffice steering group will be appointed for the trial program in December 1978. The trial is scheduled for completion in December 1980. By March 1981, IE will evaluate the trial and report to the Commission with recommendations for adopting LRPE as an ongoing programmatic effort, modifying the trial program, or abandoning this approach to evaluation.

The documents upon which this staff paper is based (these are listed in Attachment 1 to the enclosed paper) have been treated as predecisional information. Upon Commission approval of the trial program, IE recommends that these documents be released to the Public Document Room. The necessary logistics probably will take about 10 days.

Coordination: The Office of Management and Program Analysis and Standards Development concur. The Office of Nuclear Material Safety and Safeguards has no objection to the proposed program.

NRR concurs with the intended objectives of the trial program. However, because the mechanism by which these objectives are to be achieved has not yet been developed, NRR cannot offer a view as to the overall acceptability. Accordingly, NRR recommends that the overall program be subjected to periodic program office review.

The Executive Legal Director has no legal objections.

John G. Davis
Acting Director
Office of Inspection
and Enforcement

Enclosure:
"Licensee Regulatory
Performance Evaluation
Paper"

This paper is scheduled for consideration at an Open Meeting during the Week of October 23, 1978. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

DISTRIBUTION
Commissioners
Commission Staff Offices
Exec Dir for Operations
Regional Offices
Secretariat

LICENSEE REGULATORY PERFORMANCE EVALUATION
A REVIEW OF PAST EFFORTS, STATUS, AND FUTURE PLANS

Introduction

By the term "licensee regulatory performance" the Office of Inspection and Enforcement (IE) means the ability of a licensee to meet regulatory requirements and to avoid events whose occurrence appear to be directly controllable by the licensee. This does not include availability, reliability, earnings, or other measures sometimes used to evaluate the performance of utilities.

The Office of Inspection and Enforcement (IE) has been working to develop techniques for evaluating the regulatory performance on a nationwide basis since early 1976.[1] Studies of various techniques have revealed draw backs that have precluded adoption of any one technique. Yet, IE management believes that the ability to distinguish between various levels of licensee regulatory performance will give NRC a better basis for managing IE's inspection resources, by focusing inspection effort where it is most needed, and for identifying licensees whose performance should be examined. IE believes that a trial program should be initiated to further develop an acceptable technique and to test the technique.

This paper defines the concept of licensee regulatory performance, describes why IE wants to evaluate it, and suggests a number of uses that may be made of licensee regulatory performance evaluation (LRPE) results. The paper also describes the LRPE approaches that have already been considered by IE and offers some ideas of how IE may develop and use an "integrated methodology" that includes, but may not be limited to, selected aspects of each of the three methods considered to date. Finally, the paper provides a summary of the costs and benefits of LRPE and a schedule for completion of identified milestones.

---

[1] A review of the regulatory practices of other agencies has been conducted by Teknekron, Inc. under contract to IE. An initial survey of inspection and enforcement programs of twenty agencies revealed the following:

- 16 identified some kinds of criteria that could be used to assess their own effectiveness

- 7 have an assessment process that was a clearly defined element of program policy.

- 4 compare regulated facilities in terms of performance

- 7 use ratings in absolute terms.

Ref: NUREG/CR-0051 Vol. 1.

Purpose and Objectives

Licensee Regulatory Performance Evaluation (LRPE) is an attempt to systemize, on a formal basis, a method of evaluating the performance of licensees, in a regulatory sense, on a nationwide basis.

The objectives of LRPE are:

- Identification of factors that lead to different levels of regulatory performance

- Effective and efficient use of NRC inspection measures

Information from the evaluation process also can be used to evaluate aspects of the NRC inspection program.

Conceptually, the results of LRPE could be general groupings of licensees according to their performance. Most probably there will be three groupings (1) a "majority grouping"of licensees that include the average performance (2) a "majority +" grouping that performs better than the majority grouping and (3) a "majority -" grouping that does not perform as well as the majority grouping.

If LRPE is successful, it would enable IE to identify on a national basis:

1. A group of licensees that appear not to perform as well as most others. These licensees then could be examined to determine:

   - Whether, in fact, their performance is not as good as others.

   - Whether the level of performance is general within that plant's operations or specific to certain areas of the plant operations.

   - Causes for the level of performance.

   - Corrective actions to improve performance.

2. A group of licensees that appear to perform better than others. These licensees then could be examined to determine:

   - Whether, in fact, their performance is better.

   - If it is better, what are the factors that influence or cause the performance.

If the technique proves successful, LRPE could be used in several ways:

1. Managing of IE resources by directing various levels of inspection attention according to groupings.

2. Identifying the characteristics of the "majority +" performing [illegible]ees so that the industry could have access to these characteristics (if not proprietary) for improvement.

3. Identifying causes of "majority -" performance and focusing on causes so that improvement could be realized.

4. Informing the public and licensees, in a summary fashion, on a periodic basis of the licensees' regulatory performance.

5. Serving as a basis for periodic meetings between NRC regional management and licensee management for discussions of licensee performance.

In addition, LRPE will give IE management the ability to manage this "further examination" rather than rely to a high degree on regional judgments which by their very nature lack a national perspective.

Background

Over the years, a form of licensee regulatory performance evaluation has been done on a individual licensee basis. The manner in which a plant has performed against regulatory requirements has been reviewed, on a case-by-case basis, as a part of the routine inspection effort. Differences in inspection attention given by IE to licensees has been determined largely by the "problems" the licensee encounters and usually has been done on a regional rather than national basis. There has been no formal program for considering licensee performance on a national basis, and little program for reacting to licensee performance other than specific reaction to identified areas when IE believes improvement is needed.

In trying to systemize a method to evaluate the regulatory performance of NRC licensees, IE has undertaken three separate efforts, each involving a distinct approach. The first, which can be described as the "Statistical Method," produces single-valued dimensionless ratings (or Z-scores) for each licensee in a given class (in this case, operating reactors) that reflect relative numbers and types of noncompliances. The numbers of those Licensee Event Reports (LERs) attributable to personnel and procedural errors and the extent of personnel exposures and effluent releases attributed to each individual licensee are also considered.

The second approach, which can be characterized as a "Trend Analysis Method," involves detailed examination of licensee events, identification of those events that are repetitive or "causally-linked," and an evaluation of the responsiveness of each licensee's management in reacting to such events.

The third approach, the "Regional Survey Method," is more subjective; it involves a compilation of the qualitative judgments of regional managers and inspectors on a number of factors associated with the safety and security of licensed facilities. Work on these three approaches has been accomplished both in-house and under contract and reports developed.[2] More detailed descriptions of each of these methods are provided below. Although the basic data used for the "Statistical Method" and "Trend Analysis Method" are available in publicly available records, the reports themselves have previously been treated as "predecisional" information.

Licensee Regulatory Performance Considerations

Experience, thus far, shows us that the data and other influences make performance evaluation and the attendant assignment of licensees to any groupings imprecise. The concept of performance, like the concept of safety itself, is elusive. Consequently, any grouping, particularly at this stage of development of LRPE, should be considered, at best, a "director of attention," pointing IE's attention at a group of licensees worthy of more specific examination.

A hazard of proceeding into LRPE is that the groupings would be considered to sharply distinguish between the safety of operations of plants. Our efforts thus far do not support this. The fact that a licensee appears in the "majority -" grouping does not mean in a quantifiable sense that the licensee is less safe than licensees in the "majority" and "majority +" groupings. The groupings give IE management the ability, on a national level, to identify licensees for further examination aiming at improvement if necessary.

Each plant is subjected, on a plant-by-plant basis, to a formally described and conducted inspection program and continuing review by the Office of Nuclear Reactor Regulation. The plant is evaluated, on a continuing basis, as to its ability to operate with regard to safety. The NRC is charged with the protection of the public. Hence, the continuation of authority to operate a plant attests to the judgment of the NRC that the plant is operating with adequate safety. LRPE does not change that judgment.

---

[2] A list of License Performance Evaluation reports is provided in Attachment 1.

The tendency in an approach such as LRPE is to focus on the "majority -" group. However, IE has a strong interest in the "majority +" group. IE intends also to examine those licensees on an individual basis to determine whether their grouping is appropriate. If so, IE hopes to identify the characteristics, within these operations, which contribute to these "majority +" regulatory performances. If these factors can be identified they should be publicized (unless proprietary) for the benefit of the industry.

A second hazard of LRPE is that it could -- because it involves comparative grouping rather than absolute assessments -- become a constant "ratcheting" technique. Comparatively, some group of licensees could always appear "majority -". As experience is gained in LRPE, an attempt will be made to identify a "threshold" above which no special actions would be taken by IE. The goal of LRPE and the IE actions would be to achieve an industry-wide condition where all licensees remain above such a threshold.

Summary of Licensee Performance Evaluation Methods

The Statistical Method is a technique developed in-house that was applied to the evaluation of operating reactor licensees. The analysis is based upon four measures of performance: numbers of noncompliance findings, numbers of licensee-controllable events, amount of effluent releases, and amounts of personnel exposures. For each of these measures, each licensee's performance is described relative to that of the other licensees in the same class. This relative performance is then converted to dimensionless ratings, or Z-Scores, for each licensee. An overall rating (Z-Score) is obtained by computing a subjectively weighted sum of ratings for each of the four factors. The methodology accommodates different severity levels of noncompliance and adjusts noncompliance ratings to account for differences in the amount of NRC inspection time required to identify the noncompliance in each case.

The Statistical Method resulted in each licensee receiving a numerical score. Licensees could be given a relative ranking based on these scores. This was not the intent of this method; however, the ability to rank licensees relative to their peer group is inherent in a statistical approach to licensee evaluation. The assignment of a level of precision, which could lead to such a ranking is neither supported by the technique nor the data used in the calculations.

Concerns about this methodology expressed by various staff members are:

1. One product of the evaluation, a single-valued ranking of licensees, may not be warranted by the precision of the data and is affected by the subjective weighting of factors.

2. Numbers of items of noncompliance may not adequately describe the level of safety or security of a licensed facility. Variations among licensees in the significance of noncompliance will affect the quality of the Z-scores.

3. Inspection differences between regions and individual inspectors may mask the relative performance of the various licensees or be inseparable from licensee performance.

4. Requirements for the various licensees (i.e., technical specifications) may vary significantly enough to render the number of items of noncompliance an inadequate measure of performance.

5. Other exogenous variables may make it difficult to isolate the impact of LRPE on licensee performance, e.g. Revised Inspection Program, pending increase in civil penalty authority.

Each of these concerns involves judgment and differences of "degree"; each has been considered at length by staff. Despite these differences of opinion, some aspects of the Statistical Method should be considered in any LRPE method. First, noncompliance findings are a direct output of NRC's regulatory program; no LRPE method is complete without some consideration of noncompliance findings. If there are some regulatory deficiencies that detract from the meaningfulness of noncompliance findings (e.g., nonuniformity, variations in safety significance), then these regulatory weaknesses should be corrected or acknowledged as impacting LRPE accuracy. Numbers of noncompliance findings, are believed to be reasonable indicators of licensee regulatory performance.

The Trend Analysis Method is an approach developed by Teknekron, Incorporated under contract to IE. This method involves detailed subjective analysis of LERs for the purpose of categorizing them as "facility" problems reflecting reliability or similar problems beyond the direct control of the licensee, or as "personnel" or "management" problems that reflect human failure. By separating all LERs as to the reactor subsystem in which they occur and by analyzing patterns of LERs for each subsystem, Teknekron believes it is possible to identify trends of repetitive or "causally-linked" LERs that characterize a marginal performer and may allow NRC to predict the occurrence of actual incidents.

Staff concerns about this Trend Analysis Approach are that its predictive capability has not been established because Teknekron has conducted only a limited number of case studies based only on historical data, that it may be costly in terms of manpower required to conduct such analyses on a routine basis for all major NRC licensees, and that the NRC automated data base may not be complete enough to support the analysis at present.

The main advantage of the method is that it is based on analysis of actual safety or security related events. Some treatment of these potentially significant events, at an appropriate level of detail, should be considered in any LRPE approach taken in the future.

The Regional Survey Method involves the assessment of each facility by NRC inspectors and regional management. The judgments of other NRC staff members familiar with the facilities may be appropriate for future efforts. In the only effort of this type undertaken to date, IE obtained the assistance of Hay Associates, in developing a questionnaire and conducting a survey of those employees involved in inspection of operating reactors. Each survey recipient was asked to assess the "importance to safety" of a number of potential rating factors. Then, each inspector and regional manager was asked to rate each of the operating reactor sites he was familiar with in terms of its: (1) overall safety (on a scale of "acceptable" to "exceptional"), (2) site safety in specific areas of operation, and (3) the stringency of its requirements. Each recipient was encouraged to offer narrative comments on the safety of each site. In many cases there was a significant variation in the rating of a given facility by individual inspectors.

Subjective judgments of selected NRC staff members are an important element in any LRPE program, because the people who work with plant employees and facilities on a frequent basis often have insights into performance that are not immediately apparent in an isolated review of noncompliance and licensee event data. Yet, the Regional Survey Method should be recognized for what it is -- collected opinions. As any opinion survey, care must be exercised in its use. The opinions are subjective and may be affected by the make-up of the individual. They may not be clearly supportable by fact. Also, the judgments may be unduly influenced by the "last contact" with the licensee and the personality of licensee representatives. Even with those concerns judgments of qualified NRC employees are highly valued by NRC and IE management in making operating and program decisions; a systematic and explicit compilation of these judgments will be a valuable component of any LRPE program.

As indicated above, each of the three LRPE methods had strengths and shortcomings in the view of the IE Staff. The results of the Trend Analysis Method especially with the present limited sample cannot be compared with the results of the remaining two methods, the Statistical Method and the Regional Survey Method. The results of the latter two methods did not agree completely. For these reasons it was apparent that a method should be adopted that takes advantage of the strengths and compensates for the shortcomings of the methods attempted to date.

Accordingly, IE believes that an Integrated Approach is needed for Licensee Regulatory Performance Evaluation. This method could include what IE considers the best portions of the Statistical Method, the principles of the Trend Analysis Method, the general approach of the Regional Survey Method and other techniques to be developed. A licensee's regulatory performance might best be described by a combination of factual and interpretive information. The factual component of performance could include the licensee's noncompliance history over the period, a description of significant licensee events, any escalated enforcement sanctions taken by NRC against the licensee, a description of management meetings held between NRC and licensee management, and any other information considered pertinent to the licensee's performance. It should be noted that enforcement history would be factored into LRPE. Future enforcement action would not be predicated on LRPE, but rather would remain based upon the specifics of noncompliance at issue. The interpretive component of the performance evaluation could include the Region's assessment of the significance of all the factual information and a general description of inspection activity planned during the next year by the Region to cause improved regulatory performance. Included in the Region's assessment could be an assessment of the significance of the licensee's noncompliance.

During development of the Integrated Approach a foremost concern will be whether the results provide a true measure of licensee performance. Qualifications of the validity of results will be articulated.

These integrated analyses would be documented in a report that would be made available to licensees, the NRC staff, and the public. The results of these analyses will be used as a basis for periodic meetings with selected licenses.

Value-Impact Considerations

Licensee performance evaluations have been performed in the past by both Headquarters and Regional staffs using a variety of techniques. By consolidating these fragmented efforts, IE will be able to systematically conduct these necessary evaluations within existing resources. An estimated 3 man years per year will be used to develop, conduct and evaluate the trial program.

IE does not believe that the adoption of a systematic LRPE process will have any direct resource impacts on licensees, excluding possible costs to the licensee to upgrade his performance.

IE believes that the benefits of being able to evaluate licensee regulatory performance could provide a means for improved management of inspector resources and for identifying factors to be used for upgrading of regulatory performance as appropriate.

Plans and Schedule

If approved, IE intends to move promptly to develop the Integrated Approach to Licensee Regulatory Performance Evaluation. The integrated approach will serve as a basis for the Trial Program using 1978 and 1979 data. As the Trial Program proceeds, its progress will be monitored and modifications made as appropriate. By March 1981, a report to the Commission will present an evaluation of the Trial Program and recommendations concerning LRPE.

IE will document the findings of the Trial Program in three reports -- one for the 1978 analysis of operating reactors, one for the 1979 analysis, and one assessing the LRPE Trial Program. Each of these will be made available to the public.

Milestones associated with these plans are:

Before December 1978	Release existing LRPE reports to the PDR
December 1978	Appoint interoffice steering group for Trial Program
February 1979	Initial Trial Methodology for Integrated Approach complete
April 1979	First report (for 1978 data) complete
April 1980	Second report (for 1979 data) complete
December 1980	Assessment of Trial Program complete
March 1981	Report to Commission on LRPE

Attachment 1

The Licensee Regulatory Performance Evaluation Reports prepared by and for IE and listed below:

1. Draft Report - An Evaluation of the Nuclear Safety-Related Management Performance of NRC Operating Reactor Licensees During 1976 - February 1977, E. Morris Howard, Project Director.

2. Update of Draft Report - September 26, 1977, E. Morris Howard to E. Volgenau.

3. Individual Site Ratings from the IE Employee Survey on Evaluation of Licensees - April 1978, S. K. Conver, IE Study Group.

4. Licensee Performance Evaluation, Phase I Report, NUREG/CR-0110, Teknekron, Inc. - May 1978.

Draft Report

AN EVALUATION OF THE NUCLEAR SAFETY-RELATED MANAGEMENT PERFORMANCE OF NRC OPERATING REACTOR LICENSEES DURING 1976

(Licensee Management Performance Indicators)

February 1977

E. Morris Howard, Project Director
Stephen K. Conver
Robert G. Easterling
Walter S. Schwink

TABLE OF CONTENTS

Chapter I

INTRODUCTION

A. Background

"Licensee Management Performance Indicators" is the term used to describe the efforts of the NRC Office of Inspection and Enforcement to evaluate the nuclear safety-related management performance of its licensees. This draft report addresses the management performance of NRC's operating reactor licensees during the first half of calendar year 1976 (FH76). A final version of the report will include data for the full year. This report is the culmination of an effort initiated in April 1975 to develop and test a methodology for evaluating licensee management performance.

This evaluation is made on the basis of four factors that are considered to reflect the degree of success licensee management has achieved in carrying out its responsibilities for safe operation and protection of the public. These factors indicate each licensee's compliance with NRC rules, the numbers of occurrences at a licensee's facility with potential safety implications, and the extent to which the licensee limits releases of effluents and radiation exposures at his facility. Each of these factors is analyzed using standard statistical techniques that permit comparisons of licensee performance both in specific areas and from an overall perspective.

The choice of measures of licensee management performance reflects the concerns that licensees be measured objectively, using measurable and collectable statistics that apply uniformly to all operating reactor licensees. It is also important that the measures of performance include only items that are controllable by the licensee.

The use of statistical analysis to develop performance measures or indices has many precedents. Economic indicators, such as the Dow Jones averages, are commonly used to give the public an appreciation of the overall state of the economy. The "Quality of Life" index published by the Midwest Research Institute is a similar effort that ranks American cities on the basis of weighted sums of a number of indicators. Overall product rankings of Consumer's Reports and NFL quarterback rankings are other examples of the process of ranking individuals, groups, or objects according to some function of selected attributes or statistics.

In all these cases, there is some "latent variable" that is of interest, but which cannot be measured directly - economic health, quality of life, product quality, or athletic ability. By carefully choosing and analyzing data, one hopes to develop useful indicators of the latent variables. The various indicators are not equivalent to the latent variables; however, as the measured indicators are improved (numbers of libraries, interceptions, etc.), often the latent variables (quality of life, athletic performance, etc.) are also improved.

This effort to rank NRC licensee has similar objectives. While we recognize that "safety" cannot be measured directly, we hope to improve it by evaluating the success of licensee management in controlling several safety-related indicators of performance.

B. NRC and Licensee Responsibilities

Direct responsibility for conducting nuclear operations in a manner that protects public health and safety lies with the licensee. One of the ways that the licensee satisfies this obligation is by complying with NRC rules and regulations.

NRC shares this responsibility for protecting the public with the licensees. NRC responsibilities, as described in law, are to generate rules to insure safe operations and to verify that those rules are being followed. The NRC Office of Inspection and Enforcement (IE) is the arm of NRC charged with conducting this verification. IE uses its inspection force to insure compliance with the rules. Another important function of IE is to identify existing rules that need improvement or new rules that are needed.

Compliance with NRC rules is a function of licensee management. In general, a low level of noncompliance indicates that licensee management is doing a good job of carrying out its responsibilities to NRC and to the public. On the other hand, a high level of noncompliance may indicate poor management performance in this regard. The performance of licensee management is similarly evident in trends for the other indices - LERs, effluents, and exposures. The present effort proposes a method to evaluate licensee management performance on a systematic and objective basis so that negative trends can quickly be identified and so that "management breakdown" can be prevented.

C. Why Licensee Management Performance Indicators?

The NRC practice of focusing inspection attention on "poor performers" is well established and generally accepted. This evaluation of licensee management performance is designed to permit this allocation of IE inspection resources to be conducted more systematically than in the past. This effort should also allow a more objective allocation, because all licensees will be measured against a single set of performance standards. And because each licensee will be compared against the total population of similar facilities, the identification of poor performers and subsequent allocation of IE resources to these facilities (and to specific areas at a given facility) should be more uniform across the NRC Regional Offices.

A second reason for this evaluation is to permit the licensees to see the trends for themselves so that they can improve performance before it becomes serious enough to warrant enforcement action by NRC. This evaluation of performance could also be used by NRC to establish enforcement thresholds that could provide incentive for improvement without additional NRC action. If a licensee knew that his performance was approaching an enforcement threshold, he would have additional incentive to improve.

IE's mission of providing information to the public is a third reason for evaluating licensee management performance. The eventual publication of this evaluation information is consistent with IE's mission, which states that, "... the general public ... (is to be) informed on issues under the jurisdiction of this office." Since all of the performance indicator information is already in the public domain or readily available, this evaluation may preempt the efforts of those who would use it less responsibly and objectively.

D. Structure of the Report

The main body of this report is presented at a level of detail appropriate for IE and NRC management. Brief chapters summarizing the methodology (Chapter II) and results (Chapter III) are followed by a treatment of the major concerns about the evaluation that have been raised by the IE staff (Chapter IV). Recommendations (Chapter V) conclude the main report.

Additional technical detail is provided in three appendices to the report. These consist of more detailed descriptions of the methodology (Appendix A) and results (Appendix B), as well as the documentation of an analysis of the sensitivity of overall rating results to the choice of weighting factors (Appendix C).

Chapter II

METHODOLOGY

A. Introduction

This chapter describes the data elements that are considered in the evaluation, the analysis tools that are used, and the specific approach taken to analyze each distinct type of data. A detailed description of the methodology is provided in Appendix A.

B. Data Elements

Two basic types of data are considered in the analysis. "Counting data" involve numerical counts of events or occurrences. "Measurement data" describe physical characteristics of reactor operation. These data can be converted to "ranking data," which list the objects being evaluated in order of performance.

The analysis is based upon four measures of performance - noncompliance history of licensees, selected Licensee Event Reports (LERs), effluent releases, and personnel exposures. Each of these measures is discussed below.

Noncompliance items result from NRC regulation and inspection of licensee facilities. Noncompliance data consist of "counts" of NRC findings in a given time period. These noncompliance items are classified into three categories: in decreasing order of severity, these are violations, infractions, and deficiencies. The noncompliance data thus consist of number of violations, infractions, and deficiencies for each licensee considered. The data are further broken down to describe the licensee function or operation that is the source of each noncompliance. Six areas are used: (1) Administrative Control, (2) Operations, (3) Emergency Planning, (4) Radiological Protection and Control, (5) Safeguards, and (6) Quality Assurance. Individual counts of noncompliance data are presented for each of the three severity levels. Also, an overall measure on noncompliance is developed for each licensee that is a weighted sum of the numbers of findings in each of the three severity categories.

Licensee Event Report data are also stated in terms of "counts" for each licensee. LERs are reports submitted by reactor licensees when certain safety-related events occur at a facility. It is not appropriate to measure licensees by a count of all LERs submitted, because not all reportable events are controllable by the licensee, applicable to the total population of reactor licensees, or serious enough to warrant NRC enforcement action if not reported by the licensee. For this reason, only those LERs characterized as "personnel errors" and "procedural errors" are considered. Individual counts of LERs in both categories are presented, and a combined rating is obtained by adding the ratings of LERs in the two categories.

Effluent Release data are expressed in terms of licensee rankings. This is done because the actual effluent measurements may vary over several orders of magnitude for selected licensees. Effluent data are also categorized into five types: (1) noble gases, (2) halogens and particulates, (3) tritium, (4) mixed fission and activation products, and (5) solid waste. Overall rankings for effluent release are obtained by summing the licensee rankings in each of the five categories.

Personnel exposure data for each operating reactor are reported annually in the form of a table listing the numbers of persons exposed in various ranges of exposure (in rems). This evaluation of licensee management's success in limiting exposures is measured in terms of the percent of all people receiving measurable doses of radiation that received three rems or more in one year (rather than on a "ranking" or "counting" basis). Since these data are reported on a plant basis, each reactor at a multiple site is assumed to have the same exposure rating.

C. Analysis Tools

Three types of statistical techniques are used in this analysis - adjustment, normalization, and weighting procedures.

Adjustment of data is necessary because direct comparisons of licensee management performance are not always meaningful. For example, if one plant has twice as much inspection effort as another, a direct comparison of the noncompliance findings resulting from those inspections may not be meaningful, and it is necessary to make appropriate adjustments. The purpose of adjustment is to compensate for those measurement factors that are not under the licensee's control. This technique is used sparingly in the analysis to preclude elimination of actual licensee differences.

The specific techniques used to make these adjustments include linear regression, goodness of fit tests, and graphical techniques. These methods are used to identify and compensate for factors beyond the control of the licensees that can account for differences in their performance. For example, using the earlier example, Figure 1 depicts the number of infractions for each operating PWR reactor as a function of the hours of inspection devoted to each during 1976. This chart shows that infractions increase as inspection effort increases. Since this relationship also has an intuitive explanation (the more you look, the more you find), an appropriate adjustment is made.

The diagonal straight line that bisects the data in Figure 1 accounts for a significant portion of the differences in the "performance" of the various licensees. It shows that every 100 hours of inspection, on the average result in about 1.1 infractions. Thus, those licensees below the line are considered in this analysis to have better performance than those above the line, and for this measure, performance is essentially measured on a "rate" basis (infractions per hour of inspection).

The line representing 1.1 infractions per 100 hours can be obtained by one of several analytical methods - linear regression or graphical techniques. Goodness of fit tests can be used to assess whether the residual variations in licensee performance (the deviations of the individual points from the diagonal line in Figure 1) are random (what could be expected by chance). If these tests, such as the "chi square test," show that the residual variation (after adjustment) is random, this means that licensees are "homogeneous" with respect to the variable being measured (infractions) and that there is no need to look for further adjustments. A lack of randomness indicates either actual differences between licensees or the need to look for further adjustments. The approach in this analysis has been to make adjustments only when there is a logical cause and effect explanation for the relationships identified.

The techniques for adjustment of data explained in the preceding paragraphs enable comparisons of licensee performance for single measures of that performance. Another objective of this analysis is to combine the various performance measures so that an overall measure of licensee management performance can be obtained.

Normalization is one analytical technique that makes these overall comparisons possible. Its purpose is to transform each performance measure into a dimensionless quantity so that a sum of different measures is possible. The transformation used is a "Z-score," which is defined as the number of standard deviations that an observation differs from the mean of its group. That is, for any single measure, the performance of a single licensee can be expressed as the number of standard deviations that this performance varies from the group average. In this analysis, Z-scores are defined so that positive scores indicate better performance, and vice-versa.

The appropriate frame of reference for Z-scores is the standard normal distribution shown in Figure 2. This distribution has a mean of zero and standard deviation of one. When converted to Z-scores, licensee performance measures or any other statistics can be compared to this distribution. Using the standard normal, about two-thirds of the Z-scores would be expected to fall between plus and minus one, with one-sixth on either side of this interval. While these fractions are rarely achieved exactly, the Z-scores are still comparable. And, because they are dimensionless and comparable, Z-scores can be summed for various performance measures.

Weighting is the process by which Z-scores for various performance measures are summed in a manner that reflects the relative importance (weight) associated with each of the factors contributing to the overall score. The process of transforming raw data for noncompliances, LERs, effluents, and exposures to overall licensee management ratings is depicted in Figure 3. While the overall rating is of interest, the raw data and all intermediate results leading to that overall rating are significant results in their own right.

Figure 1: Adjustment of Sample Infraction Data for Inspection Time

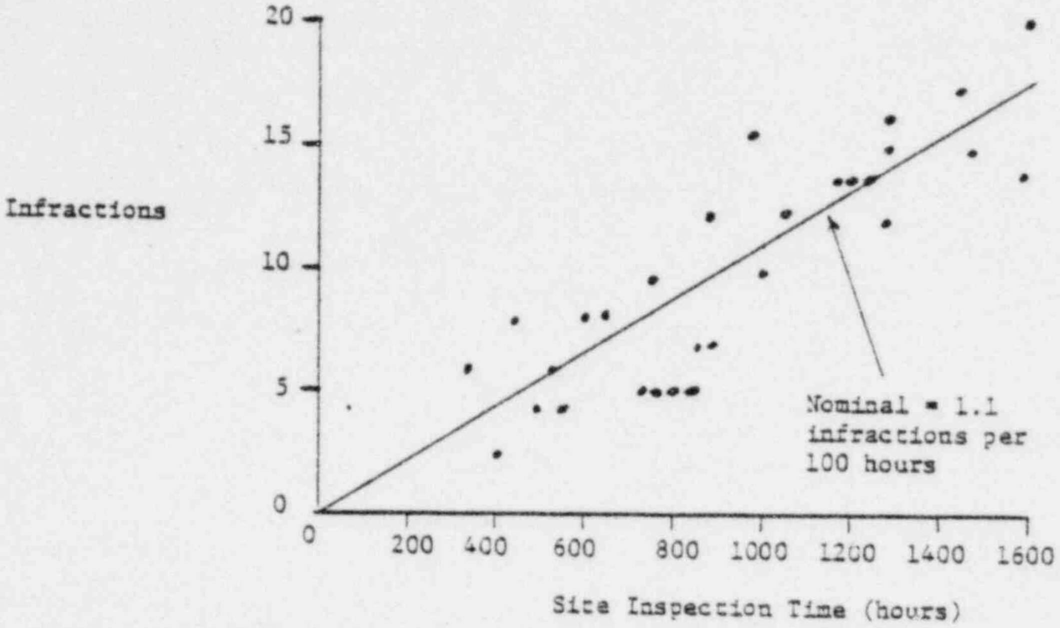


Figure 2: The Standard Normal Distribution

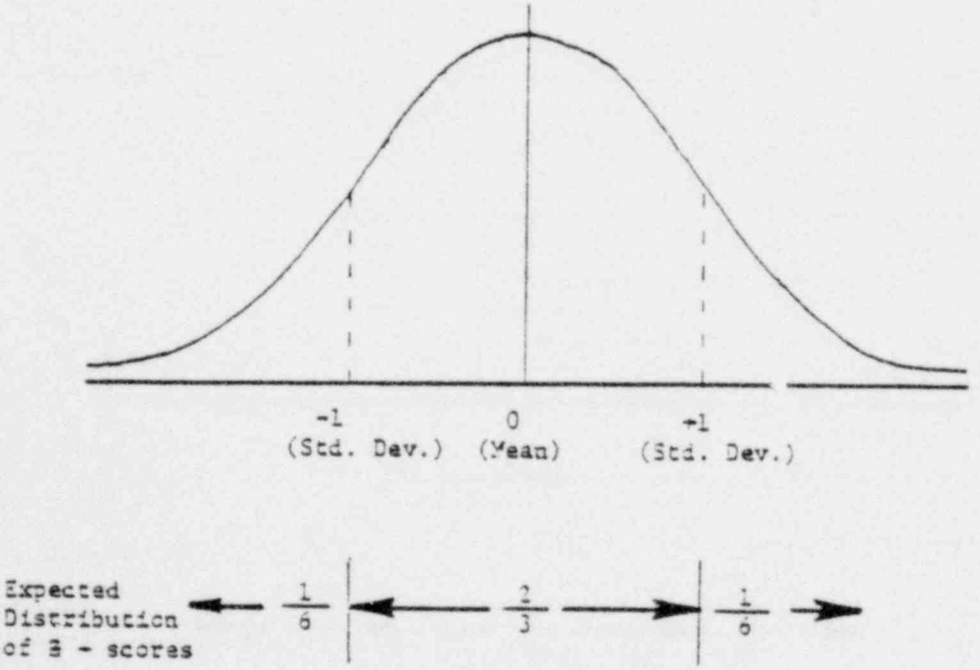
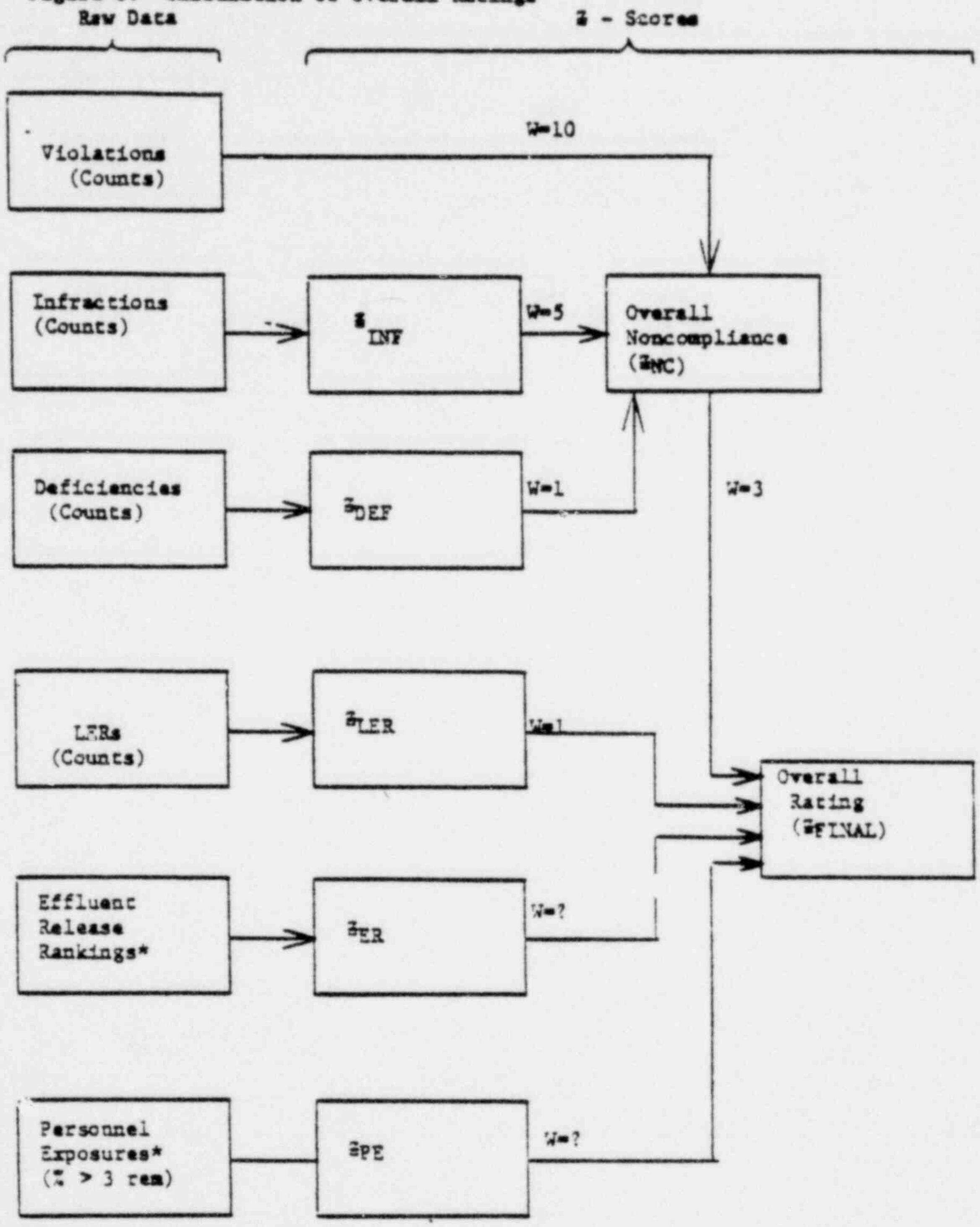


Figure 3: Calculation of Overall Ratings



* Data not yet available.

Because the process of weighting is inherently judgmental, a "sensitivity analysis" was conducted to assess the influence of a variety of alternative weightings on the overall ratings. The results of this analysis are presented in Appendix C.

In summary, the analysis process leading to overall evaluations of the performance of licensee management involves the following steps:

- Adjust data to remove factors beyond the control of licensee management.

- Normalize licensee performance in each measure to a dimensionless Z-score.

- Obtain overall ratings using weighted sums of the Z-scores.

Chapter III

ANALYSIS RESULTS (JANUARY - JUNE 1976)

A. Introduction

The methodology described in the previous chapter has been developed through analyses of 1974 and 1975 data. This chapter summarizes the results of applying these analysis methods to data obtained the first half of 1976. Details of the analysis are given in Appendix B.

The data analyzed are from reactors which went into commercial operation prior to 1 January 1976, with the exceptions of Indian Point 1 and Browns Ferry 1 and 2, which were shut down all of FH 76. This leaves thirty PWRs and 21 BWRs to be considered. At this writing, only data on noncompliances and LER's are available for FH 76. Effluent release and personnel exposure data will be included in a final report on 1976 licensee performance.

The preceding chapter and Appendix A describe how the raw data are first to be adjusted for the effect of any identifiable variable not under management control, and then normalized. Table 1 summarizes adjustments made for FH 76 data. Selection of the independent variables shown in Table 1 was based on patterns observed in the current and previous data and on the grounds that they were sensible. For example, one might expect that the more a reactor is inspected, the more noncompliances will be found, and the data support this hypothesis. The observed variation of performance with a variable such as date of commercial operation could reflect on aging effect or a systematic difference among reactor vintages (including license differences). In either case, it is considered appropriate to adjust the raw performance measures for this effect. There are many other candidate variables for use in adjusting performance measures. Although not all have been considered in the analysis of FH 76 data, several adjustments were included in this and previous years' analyses. The adjustments shown in Table 1 are considered meaningful, and as shown in detail in Appendix B, they do effect a considerable reduction in the variation among licensees.

Table 1

Summary of Adjustments: FH 76
(Adjustments are denoted by X)

		Independent Variable	
Reactor Type	Performance Measure	Inspection Effort	Age/Vintage
PWR	Noncompliances	X	
	LERs		X
BWR	Noncompliances	X	X
	LERs		X

The presentation of results in the remainder of this chapter is organized as follows. Section B gives the noncompliance data and Z-scores calculated from them. The order of presentation is first PWRs, then BWRs. For each type of reactor, the order of presentation of results is infractions, deficiencies, and then a combined Z-score for noncompliances. Section C presents the data and results for LERs, similarly ordered by PWR results, then BWR. Sections D and E will give the effluent release and personnel exposure data and analysis results when they become available. Section F provides overall results, in this case the Z-scores obtained by a weighted sum of the Z-scores for noncompliances and LERs with the weights in a 3:1 ratio. Section G summarizes the results of a sensitivity analysis of the weighting factors which is described in detail in Appendix C. These results show the effect on the overall Z-score of an incremental change in each of the performance measures.

B. Noncompliance Results

1. PWR's

Noncompliance data by type and severity level are presented in Table 2. Total on- and off-site inspection hours are also presented. After adjusting for plant inspection hours, the data are transformed to Z-scores. The Z-scores for infractions are shown in Table 3 and for deficiencies are shown in Table 4. Because the frequencies of some types of noncompliances are quite small, Z-scores are not calculated for these types, but rather the frequencies for three or four types are added and Z-scores obtained for the total. For infractions, Emergency Planning, Radiological Protection and Control, and Quality Assurance frequencies are combined. For deficiencies, these same three types plus Safeguards are added.

The Z-scores are defined so that the greater levels of noncompliances are expressed as more negative Z-scores. Thus, the poorer the relative performance, the lower the Z-scores. It should also be noted that the adjustment made is based on plant inspection hours (both on-site and off-site), not reactor inspection hours. Thus, for multiple reactor sites, the inspection hours are totaled across reactors. Noncompliance items, however, are not totaled, but are maintained on a reactor basis because, in a large majority of cases, a single event at a plant site results in noncompliance citations for each reactor at the site.

To provide an overall Z-score for noncompliance items, a weighted sum is computed, which weights violations, infractions, and deficiencies in the ratio (10:5:1). This choice of weights is based on judgments of the relative importance of the three severity levels and on the results of the

TABLE 2

FH76 NONCOMPLIANCES: PWR'S

Reactor	Admin. Control Severity* 1	2	3	Operations Severity 1	2	3	Emerg. Plan. Severity 1	2	3	Rad. Prot. & Control Severity 1	2	3	Safeguards Severity 1	2	3	Quality Assur. Severity 1	2	3	Total Severity 1	2	3	Insp. Hours
Yankee Rowe		2			1													3		3	3	434.5
San Onofre #1		1												5				1		6	1	286.5
Connecticut Yankee		5	2		3	2		1			2			2						13	4	519.0
Ginna					2									1				1		3	1	619.9
Indian Point #2		1	4		1	1		4			1	1		4						11	6	874.0
Turkey Point #3		3	2		2	1														5	3	284.1
Turkey Point #4		3	2		2	1														5	3	298.0
Palisades		5	2		4									6			2	1		17	3	690.5
Robinson #2		2	1		3										1			1		5	3	395.0
Point Beach #1						2					1			2						3	2	280.1
Oconee #1		1	1		4	2					2			3	1		1			11	4	376.9
Oconee #2		1			4	2					2			3	1		1			11	3	218.8
Surry #1		2	5		6						1			3	1		3			15	6	440.5
Surry #2		2	4		2						2			3	1		3			12	5	472.1
Prairie Island #1		2	3		3	3					1			5						11	6	336.0
Ft. Calhoun		2	1		1															3	1	349.0
Oconee #3		2			3	2					2			3	1		1			11	3	297.5
Three Mile Island #1		1	1		4	1					1			2	1		1	1		9	4	858.0
Zion #1		3	2		13	3		1		2	2								2	19	5	827.5
Point Beach #2			1		1	2								2						3	4	306.4
Zion #2		4	3		7	7					1						1			13	10	530.5
Kewaunee			1		1	2					2									3	3	470.5
Prairie Island #2			1		1	3					1			5						7	5	320.5
Maine Yankee		3	3		1	1								1	2			1		5	7	389.5
Rancho Seco #1					1	2												1		1	3	261.0
Arkansas #1		2	2		2													1		4	1	244.0
Cook #1		3	7		3	3			1											6	11	515.7
Calvert Cliffs #1														4						4		380.5
Millstone #2			3		4										3					4	6	672.5
Trojan			1		1	1					1	1					1	2		3	5	311.5
Total	0	50	52	0	80	41	0	6	1	2	22	2	0	54	12	0	14	13	2	226	121	

*Severity 1 = Violation
Severity 2 = Infraction
Severity 3 = Deficiency

TABLE 3

Z-SCORES FOR INFRACTIONS: PWR'S

Reactor	Admin. Control	Operations	Safeguards	Remaining Types*	Combined
Yankee Rowe	-0.9	0.6	1.0	0.9	0.8
San Onofre #1	-0.3	1.1	-5.1	0.8	-1.8
Connecticut Yankee	-3.3	-0.6	-0.6	-1.9	-3.2
Ginna	1.2	0.3	0.4	1.1	1.6
Indian Point #2	0.8	1.3	-1.2	-2.5	-0.8
Turkey Point #3	-1.3	0.2	1.2	1.1	0.6
Turkey Point #4	-1.3	0.2	1.2	1.1	0.6
Palisades	-2.5	-0.8	-3.3	-0.5	-3.5
Robinson #2	-1.0	-1.1	1.0	0.9	-0.1
Point Beach #1	1.2	1.5	-0.4	0.2	1.2
Oconee #1	0.8	-0.2	-0.5	-0.9	-0.4
Oconee #2	0.8	-0.2	-0.5	-0.9	-0.4
Surry #1	0.2	-1.2	-0.5	-1.6	-1.6
Surry #2	0.2	0.9	-0.5	-2.3	-0.9
Prairie Island #1	-0.3	-0.2	-2.6	0.3	-1.4
Ft. Calhoun	-1.2	0.3	0.9	0.8	0.4
Oconee #3	0.2	0.3	-0.5	-0.9	-0.5
Three Mile Island #1	0.8	-0.3	0.1	-0.2	0.2
Zion #1	0.2	-3.2	1.9	-0.1	-0.6
Point Beach #2	1.2	0.9	-0.4	1.1	1.4
Zion #2	-0.3	-0.6	1.9	0.5	0.7
Kewaunee	1.[illegible]	0.6	1.1	-1.1	0.9
Prairie Island #2	1.3	1.0	-2.6	0.3	-0.0
Maine Yankee	-2.1	0.4	-0.0	0.9	-0.4
Rancho Seco #1	0.8	0.0	0.3	0.7	1.2
Arkansas #1	-1.8	-1.0	0.8	0.7	-0.7
Cook #1	-1.5	-0.7	1.1	1.0	-0.0
Calvert Cliffs	1.0	1.2	-3.1	0.9	-0.0
Millstone #2	1.3	-0.8	1.3	1.2	1.5
Trojan	-0.2	0.2	0.9	-0.5	0.2

*Remaining Types are Emergency Planning, Radiation Protection and Control, and Quality Assurance infractions combined.

TABLE 4

Z-SCORES FOR DEFICIENCIES: PWR'S

Reactor	Admin. Control	Operations	Remaining Types*	Combined
Yankee Rowe	1.0	0.9	-2.9	-0.5
San Onofre #1	0.8	0.8	-0.9	0.4
Connecticut Yankee	-0.6	-0.9	0.9	-0.4
Ginna	1.2	1.1	-0.1	1.3
Indian Point 2	-1.2	0.6	0.3	-0.2
Turkey Point #3	-0.5	0.2	0.9	0.4
Turkey Point #4	-0.5	0.2	0.9	0.4
Palisades	-0.2	1.2	0.0	0.6
Robinson #2	-0.0	0.9	-1.8	-0.5
Point Beach #1	1.2	-0.8	0.9	0.8
Oconee #1	0.8	-0.2	0.3	0.6
Oconee #2	1.5	-0.2	0.3	0.9
Surry #1	-1.8	1.4	0.3	-0.1
Surry #2	-1.1	1.4	0.3	0.3
Prairie Island #1	-1.1	-1.5	1.0	-0.9
Ft. Calhoun	-0.1	0.8	0.7	0.8
Oconee #3	1.5	-0.2	0.3	0.9
Three Mile Island #1	0.8	0.5	-0.6	0.4
Zion #1	0.8	-0.1	1.4	1.2
Point Beach #2	0.4	-0.8	0.9	0.3
Zion #2	0.2	-2.6	1.4	-0.5
Kewaunee	0.2	-1.1	0.8	-0.0
Prairie Island #2	0.5	-1.5	1.0	0.0
Maine Yankee	-2.1	-0.2	-3.2	-3.2
Rancho Seco #1	0.8	-2.0	-1.0	-1.3
Arkansas #1	-1.8	0.7	-1.0	-1.2
Cook #1	-5.0	-2.0	-0.3	-4.2
Calvert Cliffs #1	1.0	0.9	0.8	1.5
Millstone #2	-1.0	1.2	-2.0	-1.1
Trojan	-0.2	-0.5	-3.7	-2.6

*Remaining Types are Emergency Planning, Radiation Protection and Control, Quality Assurance, and Safeguards deficiencies combined.

sensitivity study described in detail in Appendix C. Table 5 gives the overall Z-scores obtained for PWRs.

2. BWRs

Table 6 gives the noncompliance frequencies and inspection hours for the 21 BWRs under consideration. Adjusting infraction frequencies for plant inspection hours leaves considerable unexplained variability among the licensees. Further analysis, prompted by findings in the 1974 and 1975 analyses, indicates that part of this variation is assocated with the age or vintage of the reactor. This leads to a further adjustment of post-Dresden 3 reactor Z-scores. Details of this adjustment process are given in Appendix B. The Z-scores obtained from this analysis are given in Table 7. Table 8 gives the Z-scores for BWR deficiencies. As in the case of PWRs, some types of noncompliances occurred with quite small frequencies and so were summed and Z-scores then obtained for three or four types combined.

Table 9 gives the overall noncompliance Z-scores for BWRs. Violations are not shown in Table 9 because no BWR incurred a violation in FH 76.

C. Licensee Event Reports

1. PWRs

Table 10 lists the FH 76 licensee-reported personnel and procedural error frequencies for the 30 PWR's being considered. Analysis of the data indicates a dichotomy associated with the age or vintage of the reactors. Thus, nominal LER frequencies from which Z-scores are calculated are determined separately for reactors which began commercial operation prior to 1973 and for those which began in 1973 or after. The resulting Z-scores are also shown in Table 10.

2. BWRs

Table 11 gives the LER frequencies of BWRs. As in the case of PWRs, analysis of the data indicates a dichotomy related to the commercial operation date of the reactor. The division in this case occurs at a later date - 1975 instead of 1973. Thus, nominal values are determined separately for those BWRs which began commercial operation prior to 1975 and those which began during 1975. The resulting Z-scores are also shown in Table 11.

TABLE 5

COMBINED MEASURE OF NONCOMPLIANCE: PWR'S

Reactor	Viol.	Inf.	Z-Scores Deficiencies	Combined
Yankee Rowe	0	0.8	-0.5	0.7
San Onofre	0	-1.8	0.4	-1.7
Connecticut Yankee	0	-3.2	-0.4	-3.2
Ginna	0	1.6	1.3	1.8
Indian Point 2	0	-0.8	-0.2	-0.8
Turkey Point 3	0	0.6	0.4	0.7
Turkey Point 4	0	0.6	0.4	0.7
Palisades	0	-3.5	0.6	-3.3
Robinson	0	-0.1	-0.5	-0.2
Point Beach 1	0	1.2	0.8	1.3
Oconee 1	0	-0.4	0.6	-0.3
Oconee 2	0	-0.4	0.9	-0.2
Surry 1	0	-1.6	-0.1	-1.6
Surry 2	0	-0.9	0.3	-0.8
Prairie Isla ' 1	0	-1.4	-0.9	-1.5
Fort Calhoun	0	0.4	0.8	0.5
Oconee 3	0	-0.5	0.9	-0.3
Three Mile Island 1	0	0.2	0.4	0.3
Zion 1	2	-0.6	1.2	-4.3
Point Beach 2	0	1.4	0.3	1.4
Zion 2	0	0.7	-0.5	0.6
Kewaunee	0	0.9	0.0	0.9
Prairie Island 2	0	0.0	0.0	0.0
Maine Yankee	0	-0.4	-3.2	-1.0
Rancho Seco 1	0	1.2	-1.3	0.9
Arkansas 1	0	-0.7	-4.2	-1.5
Cook 1	0	0.0	-4.2	-0.8
Calvert Cliffs 1	0	0.0	1.5	0.3
Millstone 2	0	1.5	-1.1	1.3
Trojan	0	0.2	-2.6	-0.3

Table 6

NONCOMPLIANCES: BWR'S

Reactor	Admin. Control Severity* 1	2	3	Operations Severity 1	2	3	Emerg. Plan. Severity 1	2	3	Rad. Prot. & Control Severity 1	2	3	Safeguards Severity 1	2	3	Quality Assur. Severity 1	2	3	Total Severity 1	2	3	Insp. Hours
Dresden #1		2	2		1						1			4			1			9	2	325.5
Humboldt Bay														4			4	1		8	1	338.5
Big Rock Point			2		5									2	1		1			8	3	679.0
Oyster Creek		3	3		1	1					1	1		2	2		9	9		16	16	831.0
Nine Mile Point #1		2	2		3	2					2						4	5		11	9	810.5
Dresden #2		1	2			1								3				1		4	4	600.2
Millstone #1		3	4		3						3			2	1					11	5	821.0
Dresden #3			1		1									3						4	1	316.0
Quad Cities #1		1	1		3	1					1			2						7	2	500.5
Monticello		1	1											1				1		2	2	594.0
Quad Cities #2			2		4						2			2				1		8	3	303.0
Vermont Yankee		1			1									1	1					3	1	489.5
Peach Bottom #2		4	2		8										1					12	3	731.5
Peach Bottom #3		2	2		4										1					6	3	403.0
Pilgrim #1			1			1											1			1	2	779.0
Cooper		2	1		1												6	2		9	3	434.0
Hatch #1			2		6	1								1			1			8	3	526.2
Brunswick #2		1	2		5	6					2			2	2		2	1		12	11	606.5
Duane Arnold			1		4	3		2									2			8	4	768.0
Fitzpatrick			4			3			1		1			9	1		1	4		11	13	580.5
LaCrosse		1	1								2							1		3	2	577.5
Total	0	24	36	0	50	19	0	2	1	0	15	1	0	39	10	0	37	26	0	167	93	

*Severity 1 = Violation
Severity 2 = Infraction
Severity 3 = Deficiency

TABLE 7

Z-SCORES FOR INFRACTIONS: BWR'S

Reactor	Admin. Control	Operations	Safeguards	Remaining Types*	Combined
Dresden 1	-0.1	1.6	-0.5	1.1	1.1
Humboldt Bay	0.7	1.1	-3.4	-2.6	-2.1
Big Rock Point	1.0	-1.7	-0.2	0.9	-0.2
Oyster Creek	-1.6	1.1	0.1	-4.2	-2.3
Nine Mile Point 1	-0.7	-0.1	1.4	-1.9	-0.6
Dresden 2	0.6	2.1	0.1	2.1	2.4
Millstone 1	-1.6	-0.1	0.0	-0.1	-0.9
Dresden 3	1.4	1.6	0.1	2.1	2.6
Quad Cities 1	-0.6	-0.9	-0.8	0.3	-1.0
Monticello	-2.1	-0.6	-1.6	-0.6	-2.4
Quad Cities 2	0.3	-1.5	-0.8	-0.3	-1.2
Vermont Yankee	-1.5	-0.7	-1.0	0.1	-1.5
Peach Bottom 2	-1.7	-1.9	1.6	1.9	0.1
Peach Bottom 3	0.3	0.5	2.2	2.5	2.7
Pilgrim	-0.1	0.5	0.2	-0.2	0.2
Cooper	-1.2	0.9	1.5	-3.1	-0.9
Hatch	2.7	-1.3	2.1	2.4	3.0
Brunswick 2	1.3	-0.6	1.0	0.1	0.9
Duane Arnold	2.1	0.2	2.4	0.2	2.4
Fitzpatrick	2.3	2.8	-4.9	1.4	0.9
Lacrosse	-0.1	1.4	1.2	0.0	1.2

*Remaining Types are Emergency Planning, Radiation Protection and Control, and Quality Assurance infractions combined.

TABLE 8

Z-SCORES FOR DEFICIENCIES: BWR'S

Reactor	Admin. Control	Operations	Remaining Types*	Combined
Dresden 1	0.6	1.1	1.8	2.0
Humboldt Bay	0.9	0.6	-0.2	0.8
Big Rock Point	-0.2	0.8	0.5	0.6
Oyster Creek	-0.6	-0.2	-6.9	-4.5
Nine Mile Point 1	0.0	-1.3	-2.1	-2.0
Dresden 2	0.6	0.2	1.2	1.2
Millstone 1	-1.4	0.9	0.7	0.1
Dresden 3	0.9	1.1	1.8	2.2
Quad Cities 1	0.7	-0.2	1.4	1.1
Monticello	0.4	0.8	0.4	0.9
Quad Cities 2	0.0	0.9	0.7	0.9
Vermont Yankee	1.1	0.7	0.2	1.2
Peach Bottom 2	0.5	1.1	1.1	1.6
Peach Bottom 3	0.5	1.1	1.1	1.6
Pilgrim	0.7	-0.3	1.4	1.0
Cooper	0.1	0.7	-0.9	-0.1
Hatch	-0.6	-0.7	1.2	-0.1
Brunswick 2	-0.4	-6.9	-1.2	-4.9
Duane Arnold	0.7	-2.6	1.4	-0.3
Fitzpatrick	-2.1	-3.2	-3.8	-5.3
Lacrosse	0.4	0.8	0.4	0.9

*Remaining Types are Emergency Planning, Radiation Protection and Control, Quality Assurance, and Safeguards deficiencies combined.

TABLE 9

COMBINED MEASURE OF NONCOMPLIANCE: BWR'S

Reactor	Z-Scores Inf.	Z-Scores Deficiencies	Combined
Dresden 1	1.1	2.0	1.5
Humboldt Bay	-2.1	0.8	-1.9
Big Rock Point	-0.2	0.6	-0.2
Oyster Creek	-2.3	-4.5	-3.1
Nine Mile Point 1	-0.6	-2.0	-1.0
Dresden 2	2.4	1.2	2.6
Millstone	-0.9	0.1	-0.9
Dresden 3	2.6	2.2	3.0
Quad Cities 1	-1.0	1.1	-0.8
Monticello	-2.4	0.9	-2.2
Quad Cities 2	-1.2	0.9	-1.0
Vermont Yankee	-1.5	1.2	-1.2
Peach Bottom 2	0.1	1.6	0.4
Peach Bottom 3	2.7	1.6	3.0
Pilgrim	0.2	1.0	0.4
Cooper	-0.9	-0.1	-0.9
Hatch	3.0	-0.1	2.9
Brunswick 2	0.9	-4.9	-0.1
Duane Arnold	2.4	-0.3	2.3
Fitzpatrick	0.9	-5.3	-0.2
Lacrosse	1.2	0.9	1.4

Table 10

FH76 LER FREQUENCIES AND Z-SCORES: PWR'S

Reactor	Frequencies			Z-Scores		
	Personnel	Procedural	Total	Personnel	Procedural	Combined
Yankee Rowe	2	1	3	-0.6	-0.4	-0.7
San Onofre	0	0	0	1.1	0.8	1.3
Connecticut Yankee	1	1	2	0.3	-0.4	-0.1
Ginna	2	1	3	-0.6	-0.4	-0.7
Indian Point 2	3	0	3	-0.3	1.2	0.6
Turkey Point 3	0	0	0	1.1	0.8	1.3
Turkey Point 4	0	0	0	1.6	1.2	2.0
Palisades	3	0	3	-1.5	0.8	-0.5
Robinson	3	0	3	-1.5	0.8	-0.5
Point Beach 1	0	0	0	1.1	0.8	1.3
Oconee 1	2	2	4	0.4	-0.5	-0.1
Oconee 2	3	0	3	-0.3	1.2	0.6
Surry 1	1	0	1	0.3	0.8	0.8
Surry 2	1	1	2	1.0	0.3	0.9
Prairie Island 1	4	2	6	-0.9	-0.5	-1.0
Fort Calhoun	3	1	4	-0.3	0.3	0.0
Oconee 3	3	2	5	-0.3	-0.5	-0.6
Three Mile Island	4	4	8	-0.9	-2.2	-2.2
Zion 1	3	3	6	-0.3	-1.4	-1.2
Point Beach 2	3	0	3	-0.3	1.2	0.6
Zion 2	1	0	1	1.0	1.2	1.6
Kewaunee	2	2	4	0.4	-0.5	-0.1
Prairie Island 2	1	2	3	1.0	-0.5	0.4
Maine Yankee	1	0	1	0.3	0.8	0.8
Rancho Seco	2	1	3	0.4	0.3	0.5
Arkansas	2	1	3	0.4	0.3	0.5
Cook	4	3	7	-0.9	-1.4	-1.6
Calvert Cliffs	1	0	1	1.0	1.2	1.6
Millstone	3	0	3	-0.3	1.2	0.6
Trojan	6	8	14	-2.1	-5.6	-5.4

Table 11

FH76 LER FREQUENCIES AND Z-SCORES: BWR'S

	Frequencies			Z-Scores		
Facility	Personnel	Procedural	Total	Personnel	Procedural	Combined
Dresden 1	0	0	0	1.5	0.8	1.6
Humboldt Bay	2	2	4	0.2	-1.6	-1.0
Big Rock Point	1	1	2	0.9	-0.4	0.4
Oyster Creek	2	0	2	0.2	0.8	0.7
Nine Mile Point 1	3	0	3	0.2	0.8	0.7
Dresden 2	6	2	8	-2.4	-1.6	-2.8
Millstone 1	2	2	4	0.2	-1.6	-1.0
Dresden 3	1	0	1	0.9	0.8	1.2
Quad Cities 1	3	0	3	1.5	-0.4	0.8
Monticello	0	0	0	1.5	0.8	1.6
Quad Cities 2	0	1	1	1.5	-0.4	0.8
Vermont Yankee	3	1	4	-0.5	-0.4	-0.6
Peach Bottom 2	7	1	8	-3.1	-0.4	-2.5
Peach Bottom 3	2	0	2	0.2	0.8	0.7
Pilgrim	3	1	4	-0.5	-0.4	-0.6
Cooper	1	0	1	0.9	0.8	1.2
Hatch	17	4	21	-1.4	-0.3	-1.2
Brunswick 2	14	5	19	-0.6	-0.8	-1.0
Duane Arnold	8	4	12	1.2	-0.3	0.6
Fitzpatrick	10	0	10	0.6	1.9	1.8
Lacrosse	0	0	0	1.5	0.8	1.6

D. Effluent Releases

Results on effluent releases are not presented because all licensees have not reported this information for FH 76 and because the data which have been reported are not yet in a suitable form for the analysis described in the preceding chapter and Appendix A.

E. Personnel Exposures

These data are reported on an annual basis and 1976 results are not yet available for this analysis.

F. Overall Performance

At this time, the only performance measures available for inclusion in an overall performance measure are the Z-scores determined for noncompliances and LERs. The results of the sensitivity analysis in Appendix C and judgment as to the relative importance of these two performance measures led to a choice of weighting these two Z-scores in a 3:1 ratio. Tables 12 and 13 give the resulting overall Z-score, denoted by Z_{FINAL}.

As a summary, the Z_{FINAL} values are categorized, as follows, and tabulated by category in Tables 14 and 15:

Range of Z_{FINAL}	Category
$Z_{FINAL} > 1.0$	"Above Average"
$-1.0 \leq Z_{FINAL} \leq 1.0$	"Average"
$Z_{FINAL} < -1.0$	"Below Average"

Summarizing the licensee performance using these three categories aids in interpreting and presenting the results and also helps to prevent an unwarrantedly precise interpretation. Licensee performance measures are subject to many sources of variation. While the dominant and appropriate sources have probably been accounted for, additional sources of variation may not have been accounted for. For this reason and because of arbitrariness in the choice of weights, small differences among Z-scores or rankings should not be taken as indicative of real differences in performance. Only the larger differences among the three categories are reliable indicators of actual performance difference.

If all licensees were homogeneous in their performance as measured by noncompliance rates and LER rates, then the expected results would be that about two thirds of the reactors would fall in the "Average" category and one-sixth in each of the other two categories. Tables 14 and 15 show somewhat more variation than would be expected - more reactors in the tails of the distributions than expected. This result supports the hypothesis, (but does not prove) that there are real differences in licensee performance.

Table 12

FH76 Overall Performance: PWR's

Reactor	Z_{NC}	Z_{LER}	Z_{FINAL}
Yankee Rowe	0.7	-0.7	0.4
San Onofre	-1.7	1.3	-1.2
Connecticut Yankee	-3.2	-0.1	-3.1
Ginna	1.8	-0.7	1.5
Indian Point 2	-0.8	0.6	-0.6
Turkey Point 3	0.7	1.3	1.1
Turkey Point 4	0.7	2.0	1.3
Palisades	-3.3	-0.5	-3.3
Robinson	-0.2	-0.5	-0.4
Point Beach 1	1.3	1.3	1.6
Oconee 1	-0.3	-0.1	-0.3
Oconee 2	-0.2	0.6	0.0
Surry 1	-1.6	0.8	-1.3
Surry 2	-0.8	0.9	-0.5
Prairie Island 1	-1.5	-1.0	-1.7
Fort Calhoun	0.5	0.0	0.5
Oconee 3	-0.3	-0.6	-0.5
Three Mile Island 1	0.3	-2.2	-0.4
Zion 1	-4.3	-1.2	-4.5
Point Beach 2	1.4	0.6	1.5
Zion 2	0.6	1.6	1.1
Kewaunee	0.9	-0.1	0.8
Prairie Island 2	0.0	0.4	0.1
Maine Yankee	-1.0	0.8	-0.7
Rancho Seco 1	0.9	0.5	1.0
Arkansas 1	-1.5	0.5	-1.3
Cook 1	-0.3	-1.6	-1.3
Calvert Cliffs 1	0.3	1.6	0.8
Millstone 2	1.3	0.6	1.4
Trojan	-0.3	-5.4	-2.0

Table 13

FH76 Overall Performance: BWR's

Reactor	Z_{NC}	Z_{LER}	Z_{FINAL}
Dresden 1	1.5	1.6	1.9
Humboldt Bay	-1.9	-1.0	-2.1
Big Rock Point	-0.2	0.4	-0.1
Oyster Creek	-3.1	0.7	-2.7
Nine Mile Point 1	-1.0	0.2	-0.9
Dresden 2	2.6	-2.8	1.6
Millstone	-0.9	-1.0	-1.2
Dresden 3	3.0	1.2	3.2
Quad Cities 1	-0.8	0.2	0.7
Monticello	-2.2	1.6	-1.6
Quad Cities 2	-1.0	0.8	-0.7
Vermont Yankee	-1.2	-0.6	-1.3
Peach Bottom 2	0.4	-2.5	-0.4
Peach Bottom 3	3.0	0.7	3.1
Pilgrim	0.4	-0.6	0.2
Cooper	-0.9	1.2	-0.5
Hatch	2.9	-1.2	2.4
Brunswick 2	-0.1	-1.0	-0.4
Duane Arnold	2.3	0.6	2.4
Fitzpatrick	-0.2	1.8	0.4
Lacrosse	1.4	1.6	1.8

Table 14

PWR Overall Performance, FH76, By Categories

Above Average	Average	Below Average
Ginna	Yankee Rowe	San Onofre
Turkey Point 3	Indian Point 2	Connecticut Yankee
Turkey Point 4	Robinson	Palisades
Point Beach 1	Oconee 1	Surry 1
Point Beach 2	Oconee 2	Prairie Island 1
Zion 2	Surry 2	Zion 1
Millstone 2	Fort Calhoun	Arkansas 1
	Oconee 3	Cook 1
	Three Mile Island 1	Trojan
	Kewaunee	
	Prairie Island 2	
	Maine Yankee	
	Rancho Seco 1	
	Calvert Cliffs	

Table 15

BWR Overall Performance, FH76, By Categories

Above Average	Average	Below Average
Dresden 1	Big Rock Point	Humboldt Bay
Dresden 2	Nine Mile Point 1	Oyster Creek
Dresden 3	Quad Cities 1	Millstone
Peach Bottom 3	Quad Cities 2	Monticello
Hatch	Peach Bottom 2	Vermont Yankee
Duane Arnold	Pilgrim	
Lacrosse	Cooper	
	Brunswick 2	
	Fitzpatrick	

G. Sensitivity Analysis

The overall licensee performance measure, tabulated in the previous section, is obtained by taking a weighted sum of violations, Z-scores pertaining to type and severity of noncompliance items, and LERs. When the data become available, this sum will also include Z-scores for effluent releases and personnel exposures. Table 16 displays the ratio of the weights used at each stage of the FH 76 analysis. First, equal weights are given the types of infractions, deficiencies, and LERs. Then violations, and the Z-scores for infractions and deficiencies are weighted in a -10:5:1 ratio. Finally, the Z-scores for noncompliances and LERs are weighted 3:1. The last column of Table 16 shows the cumulative effect on Z_{FINAL} of these choices of weights. The quantities given are the change in Z_{FINAL} which result from an increase of one unit in each of the components of ZFINAL. Thus, for example, an increase of one violation would decrease Z_{FINAL} by almost 2.0; if the Z-score for one type of infraction increased (improved) by 1.0, then Z_{FINAL} would increase by .47; the effect of an increase in a Z-score for one type of deficiency is to increase Z_{FINAL} by .11 and the effect of LERs is between that of infractions and deficiencies.

The effects in Table 16, with the exception of violations, are those of Z-scores on Z_{FINAL}, not the effects of the raw performance measures. To obtain those effects, recall that a Z-score is equal to the number of standard deviations that an observation deviates from its group average. When the standard deviations are considered (see Appendix C for details), each additional infraction per year, not per half year, would decrease Z_{FINAL} by about .24 and each additional deficiency would decrease Z_{FINAL} by about .05. Note that the effects of violations, infractions, and deficiencies are approximately in the ratio of 100:10:2 which are the IE enforcement point values. Each additional LER per year would decrease Z_{FINAL} by about .13. These results show that sources of variation not accounted for in the analysis, but which might be responsible for a few infractions, deficiencies, or LERs per year, will not have a major effect on the overall performance measure, Z_{FINAL}.

Table 16

Ratios of Weights Used In Determining Z_{FINAL} and Sensitivity of Z_{FINAL} To Changes in the Components of Z_{FINAL}

Performance Measure	Relative Wts.	Relative Wts.	Relative Wts.	Effect on Z_{FINAL}
Violations		-10		1.86
Infractions				
Ad. Control	1			.47
Ops.	1	5	3	.47
Safeguards	1			.47
Others	1			.47
Deficiencies				
Ad. Control	1			.11
Ops.	1	1		.11
Others	1			.11
LER's				
Personnel	1		1	.22
Procedural	1			.22

Chapter IV

RESOLUTION OF STAFF CONCERNS

A. Introduction

While there are a number of reasons why NRC should rate the performance of licensees, the initial efforts to do so have generated a number of legitimate concerns that should be satisfactorily addressed before the evaluations are offered in the form of public reports. The major concerns can be summarized as follows:

- Compliance versus safety. Licensee noncompliance histories may be random occurrences that do not reflect the degree of safety provided at a given facility.

- Variables not under licensee control. The present evaluation may be affected by a number of factors beyond the control of the licensee, including the influence of inspector and regional differences.

- Uniformity of requirements. Requirements on individual licensees may differ to the extent that meaningful comparisons of noncompliance are not possible.

- Impact on licensee motivation. The evaluation process may not provide the licensee incentive to improve, and could motivate actions that are counter-productive to safety.

- Subjectivity of the evaluation. The usefulness of the evaluation may be limited by its subjectivity, particularly in the use of weighting factors.

- Possibility of misinterpretation. The results of the evaluation may be misinterpreted or misused by the public.

- Absolute versus relative rankings. The value of the evaluation may be limited because it ranks licensees relative to each other rather than against absolute standards of acceptability.

B. Compliance versus Safety

The safety of a nuclear reactor is an abstract "latent variable" that cannot be measured directly. This evaluation postulates that the noncompliance histories of licensees and other performance measures are reasonable indicators of safety. If NRC rules are proper, then the case can be made that the higher the degree of compliance with the rules and regulations, the less likely is a serious event to occur as a result of a facility management decision. Paramount to the discussion of compliance versus safety is the simple and logical conclusion, which cannot be statistically verified, that a facility which is operating within the parameters established by NRC, free of personnel and procedural errors, but excluding design and fabrication errors, must therefore be less prone to incidents or accidents.

If NRC rules are not properly related to safety, this shortcoming should be remedied as soon as possible. This evaluation should not be used as a vehicle to compensate for weak rules by "ratcheting" licensees. However, a lack of complete satisfaction with NRC rules should not prevent us from measuring the performance of licensee management based on their compliance with those rules.

A related argument is that compliance is an incomplete performance measure because it does not explicitly credit extra effort by a licensee to exceed minimum requirements. This is true if only compliance measures are considered, because compliance is measured against minimum standards. It is not true for LERs, effluents, and exposures. It can also be argued that a licensee who strives to exceed requirements will, in the long run, see that attitude reflected in a low level of noncompliance. It does not seem reasonable that extra effort by a licensee in one area should compensate for a failure to meet minimum requirements in another.

C. Variables Not Under Licensee Control

If we can identify "indicators" of licensee performance that are both objective and measurable, the next concern is assuring that the evaluation includes only those facets of management performance under the direct control of the licensee. It is unfair to rate the licensee using measures that are beyond his control, and it is also pointless, because the licensee is not able or motivated to improve under these circumstances.

For these reasons, a number of adjustments have been made to eliminate those facets of the performance measures that are not licensee-controllable. The results of the analysis (see Appendix B for details) show that adjustments for the effects of these factors, such as inspection effort and reactor age, reduce the variability among reactors such that remaining variation appears "random." Additional effects beyond licensee control may still remain; however, they are not large enough to be detected. Further experience with Licensee Management Performance Indicators may suggest additional adjustments (or omissions of existing ones).

As shown in the sensitivity analysis (Appendix C), the effects of factors beyond licensee control are not likely to influence a licensee's overall evaluation. This would require changes on the order of three or four infractions, 10 to 15 deficiencies, or six to eight LERs per year.

Differences in licensee performance can also be attributed to inspection differences. The analysis to date has not distinguished between "good" and "average" inspectors or between "hard" and "easy" inspectors. For these reasons, it can be argued that a licensee inspected by one individual might be at a competitive disadvantage with a licensee inspected by another. This possibility exists because noncompliance data is a reflection of the inspector as well as the licensee.

The possibility of inspection differences is a problem that IE faced at an early date, recognizing both inspector and management differences. As a result, the NRC inspection program was designed to provide uniform application of NRC rules and regulations. A systematic program of enforcement was developed concurrently to match this uniform inspection program. The enforcement program catalogues items of noncompliance, establishes severity levels, and defines thresholds for enforcement actions. While these programs cannot preclude inspection differences, it seems reasonable that they would significantly reduce the variation among inspectors.

A cursory review of "noncompliance yield" shows some differences between specialists and generalists. The former tend to have lower "inspection yields" because they do not cover a broad spectrum of activities; this does not detract from their effectiveness. A strong argument for not formally pursuing this review is the possibility that inspectors could believe that they were being judged on the basis of "noncompliance yield." This perception would likely result in the proliferation of insignificant issues to the detriment of substantive issues. The judgment of an inspector's performance lies with regional management and the Director of IE. Thus, the impact of inspector differences cannot be determined with a high degree of accuracy, and such an evaluation may not be warranted in view of the uniform inspection program and enforcement guidance.

Comparisons made to date do not indicate significant differences between the five NRC regions in terms of inspection yield rate (noncompliance findings per hour of inspection). In summary it is not possible to completely separate the evaluation of licensees from the influence of those doing the measurement, but it is possible to gain insight into the extent of that influence.

D. Uniformity of Requirements

Another question of uniformity involves NRC requirements governing both physical plant operation and reporting of weaknesses. Measuring licensee compliance with NRC requirements is meaningful only to the extent that the licensees can be measured against common standards. While NRC rules and regulations are the same for all similar licensees at any given time, conditions of the Operating License vary from facility to facility because different rules were in effect at the time each license was issued.

In assessing this concern, there are two questions that should be addressed. The first is: How great are the differences? The second is: Does this level of difference in requirements permit meaningful comparison of compliance with those requirements? Whereas the first question is deterministic, the second is judgmental and should be resolved by decision.

While actual requirement differences for various licensees are important, those requirements that are the basis for inspections at the various facilities are more relevant to the question of uniformity. The IE inspection program was designed with the expressed intent of providing a uniform and standard inspection program to assure that each facility is subjected to the same level of inspection against the same comparative requirements.

Because the largest differences in requirements are attributed to age and type of reactor, a reasonable sample of facilities might include four reactor licensees consisting of one PWR and one BWR in each of two age groups early and recently-licensed. If this effort is judged necessary, it should build upon an earlier study by the Sandia Corporation that identified some 2,500 distinct regulatory requirements for the Three Mile Island Unit 1 facility and categorized those requirements into three levels of safety significance. It should be noted that in the present evaluation, adjustments made for reactor age may compensate for requirement differences.

In summary, the inspectors' perception and application of requirements is considered more important than their actual differences. Even if a detailed comparison is made, the question of uniformity of requirements will remain a judgmental question and differences of pinion will no doubt persist.

E. Impact on Licensee Motivation

The eventual impact of this evaluation on licensee motivation is unknown. The intent of the evaluation is to motivate licensees toward compliance with NRC rules. It is important that the evaluation process does not encourage licensees to take actions that are counter-productive to safety or do not make sense. Licensee motivation has been considered as one factor in the selection of performance measures. Nonetheless, we must continue to be alert to the possibility of motivating improper actions on the part of the licensees.

Another concern about licensee motivation involves the possibility that we might penalize a licensee for his attempts at self-improvement. LERs have been cited as an example. Although licensee reporting of certain events is required by regulation, it has been suggested that counting LERs as negative factors in the evaluation penalizes the licensee for self-reported weaknesses and we may be providing an incentive for the licensee to report as little as possible. Because we want to encourage self-improvement, we should - as a general rule - avoid actions that have the opposite effect. Recognizing these potential problems, the evaluation considers only those events (and resulting LERs) that are preventable by licensee management, that could apply to all licensees, and that would be the basis for enforcement action if not reported. Those LERs citing "personnel errors" and "procedural errors" were chosen because they seem to satisfy those three criteria.

F. Subjectivity of the Evaluations

Another concern that has been raised about the performance evaluation is that it involves too much subjectivity, particularly in the weighting factors that are applied in combining individual parameters to obtain overall rankings. Several comments can be made on this point. First, any attempt to combine a number of factors into single evaluation necessarily involves a weighting process, even if all factors are considered of equal importance. The important points to remember in assigning weights are: (1) the original data should be preserved so that people can use their own weighting factors to arrive at independent rankings, and (2) the weighting process should be clearly explained. All raw data, all intermediate analysis steps, and the nature of the weighting process will continue to be included in all reports on this evaluation.

Furthermore, a number of different weighting schemes have been applied to the data analyzed so far, and the overall rankings have been relatively insensitive to a wide range of weightings (see Appendix C). Further consideration of the choice of weights is suggested as additional data are collected and analyzed.

G. Possibility of Misinterpretation

Misinterpretation is a potential problem any time that a government agency releases information to the public, because some members of the public will misrepresent this information for self-serving purposes, particularly in nuclear matters. The pertinent question for this evaluation is: Should we put this information into the public domain, when there is a possibility that it may be misinterpreted and exploited? Since the unprocessed information is already in the public domain or could become so through a Freedom of Information Act request, there is nothing to prevent any group or individual from gaining access to, analyzing, and publishing this information. Thus, the question may not be whether to release the information, but whether to take the initiative in using it as equitably as we can.

There is probably general agreement in NRC that we should continue to release raw data to the public. There is some disagreement, however, on the advisability of releasing processed information, that is, interpreted data. A decision to do so would indicate among other things - that the benefits to be gained from publicizing licensee management performance outweigh the individual inequities, whether viewed as large or small that invariably accompany any large evaluation process, be it NRC licensees, promotion selection boards, college football rankings, or other similar processes.

H. Absolute versus Relative Rankings

A final area of concern is that the licensee management performance indicators rank licensees relative to each other rather than against some absolute standard of acceptability. The relative rankings bear no relationship to the NRC enforcement program, and the NRC cannot penalize or improve licensees on the basis of low evaluation rankings.

The evaluation is inten to be used as a management tool. However, it is possible that these indicators of licensee performance can eventually be used to support the need for enforcement action, much as the present "point system" is used.

It has been suggested that for various reasons, perhaps beyond the licensee's control, some licensees will find it difficult to improve their performance relative to their contemporaries, particularly if the total population is improving. Since it would otherwise be difficult to motivate a licensee who considered himself inescapably caught in the low group, it might be desirable to measure licensees against some fixed standards. Using this scheme, a licensee would have some incentive to improve, even if he couldn't better his position with respect to the other licensees.

I. Summary

This chapter has identified the major concerns about the Licensee Management Performance Indicators study that have been raised by the IE staff. It has also attempted to resolve those concerns or propose additional efforts that could facilitate their resolution. Those efforts judged necessary by the project team are included in the following chapter.

Chapter V

RECOMMENDATIONS

A. Introduction

This chapter presents the recommendations of the project team concerning the implementation of the results of the current effort, the content and uses of future reports of this type, and a number of future tasks that should be undertaken to improve the usefulness of the Licensee Management Performance Indicators methodology or resolve staff concerns.

B. For the Current Report

The project team recommends that:

o The concept and methodology of the Licensee Management Performance Indicators study be approved by the Director of IE, with the understanding that further improvements in both areas will continue to be made.

o This report be coordinated throughout NRC and subsequently presented to the Commission for endorsement.

o The methodology and results of this study be provided to licensees and the public for information and comment.

o IE publish the report for internal NRC, licensee, and public comment.

o The results of this evaluation be used, on a test basis and at the discretion of Regional Directors, as a management tool for allocating inspection resources. Results of this test should be complete by the end of 1977.

C. For Future Reports

The project team recommends that:

o Future evaluations be based upon:

- Noncompliance data.
- LER, Effluent Release, and Personnel Exposure data, unless the NRC coordination process, licensee comment, or public response indicate persuasive reasons to the contrary.

o Subjective evaluations of NRC regional people should be considered, but should not be a part of the evaluations that are released to the public.

o Results of future evaluations be produced semi-annually as NRC products and released to the public, unless persuasive arguments or Commission guidance to the contrary is received.

D. <u>For Additional Analysis</u>

The project team recommends that the following tasks be continued:

o Validate the existing methodology through an independent review process.

o Solicit subjective rankings of the operating reactor licensees from regional people, and correlate these rankings with those obtained in this evaluation.

o (A detailed comparison of requirements for selected licensees is not considered necessary.)

The project team recommends that the following tasks be undertaken:

o Take steps necessary to improve the timeliness of effluent release and personnel exposure data.

o Conduct additional sensitivity analyses to determine the sensitivity of results to weighting factors.

o Develop a methodology that will provide the capability of measuring licensee management performance against fixed standards or against the previous year's performance for any given licensee.

Appendix A

Methodology

A. Introduction

This appendix describes in detail the methodology used to reduce each performance variable to a Z-score. To illustrate the methodology, some of the results from the analysis of data from the first six months of 1976 are included. This is because details of the analysis cannot be laid out separate from the data; the data guide in the selection of the variables and in the way in which adjustments are made. Analysis of data from 1974, '75, and '76 showed that patterns of variation are not always repeatable so the precise formulas used to arrive at an overall measure of licensee performance this year, the next, and afterwards cannot be given because they will depend on observed data and because changes in licensee and NRC practices and policies may alter from year to year the patterns of variation in the observed data. However, the same principles and methods can be applied from year to year to arrive at measures of licensee performance.

The order of presentation in this appendix is to describe the methodology for the analysis of noncompliances - violations, infractions, and deficiencies in that order - then for LER's, effluent releases, and personnel exposures. Section 6 gives the relative weights assigned to these measures in order to obtain an overall performance indicator and Appendix C describes the resulting effect of those weights on the overall performance indicator.

2. Noncompliances

Noncompliance data are counting data -- the numbers of occurrences of a particular event over a specified period, for example the number of infractions pertaining to operations at a plant during the first six months of 1976. A statistical model often used to describe counting data is the Poisson distribution. One application of this distribution is to describe the variation in counts one would expect to observe "just by chance." That is, suppose 60 infractions were incurred among a group of 30 reactors. Even under the assumption that all reactors perform equally, one would not expect to see exactly two infractions for each of those reactors. Rather, one would expect to see some variation among those reactors, the sort of distribution one would get by "randomly" throwing 60 balls into 30 baskets. It is this distribution that the Poisson distribution can be used to approximate.

This model is useful because if one adjusts infraction frequencies, for example, for the effect of inspection effort, and as a result the residual variation among infractions (the variation remaining after the adjustment) is "random" or "Poisson-like," then this is an indication that, at least based on the observed data, licensees are homogeneous with respect to infraction rate and that there is no need to look for further adjustments. The statistical method used to assess whether the residual variation can be described as "random" is the chi-square goodness of fit test. If this test does not show adequate agreement between the residual variation and what one would expect just by chance, then the indication is one or both of the following: (1) there are differences among licensees or (2) there are other factors for which further adjustment is required. Deciding between these two is not clear cut. The approach followed here is to make further adjustment only if the data and an understanding of what sort of adjustments are reasonable support doing so.

One property of the Poisson distribution is that its mean is equal to its standard deviation squared. Let that mean value be denoted by NOM, for nominal, and let NC_i denote the number of noncompliances in some particular category incurred by licensee i. Then, making use of this property of the Poisson distribution leads to calculating a Z-score by

$$Z = \frac{NOM - NC_i}{\sqrt{NOM}}$$

As has been discussed, the influence of variables outside the licensees control may require adjustment of the performance measures for the effect of these variables. This could be accomplished by adjusting the observed count, NC_i, or by relfecting the effect of these variables in the calculation of NOM. The approach followed is the latter. For example, statistical analysis of infraction frequencies for PWR's the first six months of 1976 indicated that the rate of infractions incurred by a reactor was equal to about 1.1 per 100 hours of inspection time. Thus for a reactor with 500 hours of inspection, the nominal value would be 5.5 infractions, for 1,000 hours, it would be 11 infractions, and Z-scores were calculated accordingly.

For the purpose of evaluating licensee performance, noncompliance counts are obtained for each severity level and each type of noncompliance - a total of 18 performance variables. For time periods of interest, such as six months or a year, many of these 18 counts will have quite low frequencies, say less than an average of .5 per reactor. This means that it will be difficult to detect the effect of factors for which one might want to adjust and that Z-scores calculated from quite small nominal values may be unstable. To avoid these problems, the following approach was taken in analyzing noncompliance data from the first half of 1976 (FH76).

a. Violations were quite infrequent -- a total of 2 for the 51 reactors considered. Because of this and because it does not seem appropriate to make any compensation or adjustment to the number of violations -- the most severe noncompliance -- a licensee incurs, no adjustment is made and violations are included, and weighted heavily, in arriving at an overall noncompliance measure.

b. Infractions were analyzed by considering first the total number of infractions summed across the six types which were considered. Variation among the infraction frequencies of the 30 PWR's considered was more than one would expect by chance so the necessity of adjusting was considered. As already mentioned, one might expect that the more inspection effort afforded a reactor, the more infractions would be found. Inspection effort can be measured in several ways. The measure selected here was the total in- and out-of-office inspection hours pertaining to a plant. (If a plant included more than one reactor, then the inspection hours are totaled across the two or three reactors at that plant. This is done because quite often one occurrence at a plant site results in each of the reactors at that site receiving a noncompliance citation.) By regression analysis the relationship between the nominal number of infractions and inspection time was found to be NOM = 1.1T, where T is inspection time in hundreds of hours. Calculating Z-scores and performing a goodness of fit test indicated that, by this adjustment the residual variation of infractions was reduced to "random" variation.

To obtain Z-scores for each type of infraction, the approach taken was to distribute the nominal 1.1T across the six types according to the overall relative frequencies of those types. Thus, because 22% of all the infractions incurred by the 30 PWR's under consideration were in the category of administrative control, the nominal value taken in this case is .22 (1.1T) = .25T. For operations, the nominal is .40T and for safeguards it is .25T. The other types of infractions occurred with lesser frequencies so they were grouped, this remaining group having a nominal frequency of .20T

(An alternative approach which could have been taken to analyze the frequencies of each type of infraction separately. This could lead to a different selection of variables for which to adjust. For example, it might be found that operations infractions should be adjusted for out-of-office inspection time while administrative control infractions should be adjusted for in-office inspection time and age of reactor. For the sake of simplicity and because only six months of data do not seem to warrant this refinement, this approach was not followed. A check on the adequacy of the approach taken is possible and is described in the next paragraph.)

To obtain an overall Z-score for infractions the four Z-scores just obtained -- administrative control, operations, safeguards, and the remaining types combined -- are first summed. If the Z-scores are independent random variables with standard normal distributions, as one would expect if indeed licensee performance was homogeneous, then the sum of four Z-scores should have a mean of zero and a variance of 4.0. Thus, to convert the sum to a Z-score, the sum should be divided by $\sqrt{4} = 2$, so that is done to get an overall Z-score for infractions for each PWR. For the data under consideration, the resulting Z-scores were quite similar to the Z-scores obtained initially for total infractions. This is an indication that the short cut approach followed to obtain nominal frequencies by type of infraction did not introduce excessive additional variation.

c. Deficiencies were analyzed in much the same way as infractions, the only exception being that safeguards deficiencies were infrequent enough that they were grouped with the remaining types rather than being analyzed separately.

d. To obtain an overall Z-score for noncompliances, a weighted sum is used. To choose the weights requires a judgment of how strongly the overall Z-score should reflect performance in terms of violations, infractions, and deficiencies. The particular sum used in this analysis is

$$Z_{NC} = (-10V + 5Z_I + Z_D)/\sqrt{25}$$

where V is the number of violations and Z_I and Z_D are the overall Z-scores pertaining to infractions and deficiencies, respectively. Note that the denominator of Z_{NC} is the sum of the weights squared for only infractions and deficiencies. Thus, in essence, this formula is treating the nominal number of violations as zero -- a mean of zero and a standard deviation of zero. A later section on the sensitivity of the overall Z-score to the choice of weights will show the effect of this treatment.

e. The statistical analysis of BWR noncompliance data for the first six months of 1976 is much the same for PWR's, the only exception being that the Z-scores for infractions reflect an adjustment for inspection time and for age.

3. Licensee Event Reports

The methodology for analyzing LER's is similar to that for noncompliances because LER's are also counts of occurrences. The analysis is simpler because rather than having 18 subcategories to consider there were only two -- personnel and procedural errors. For LER's of FH76, the variation among reactors, both PWR's and BWR's, was more than would be expected by chance, so it was necessary to consider possible adjustments. Because newer plants tend to have tighter reporting requirements and because newer plants might be expected to have "shake-down" sorts of problems, age is a logical candidate for adjustment. Examination of the data supported this conjecture. Examination of the data also suggested that rather than a continuous sort of adjustment, as noncompliances were adjusted for inspection time, the reactors should be divided into two age groups and nominal LER frequencies obtained for each group. These nominal frequencies are then divided between personnel and procedural errors and Z-scores calculated accordingly. An overall Z-score for LER's is then obtained by adding the Z's for personnel and procedural LER's and dividing by $\sqrt{2}$.

4. Effluent Releases

The five categories of release considered are: noble gases, halogens and particulates, tritium, mixed fission and activation products, and solid waste. The analysis of these data is considerably different from that of noncompliance and LER data for several reasons. First, these are measurement data, not counting data, so there is no framework of "randomness" to compare these data to. Second, in principle at least, though perhaps not economically possible, any licensee can reduce effluents to an arbitrarily low level so no consideration was given to adjustments for factors outside the licensee's control. (Another reason for not doing an extensive analysis of the possible adjustments is that in obtaining an overall performance measure, effluents are weighted lightly enough relative to noncompliance and LER's, that adjusting the effluent Z-scores is not apt to markedly affect the overall performance measure.) Third, the amount of release in any category may vary considerably, over several orders of magnitude, so Z-scores calculated from the mean and standard deviation of such data may be distorted. To avoid this, the actual releases were replaced by their ranks, low to high, for subsequent analysis. A fourth difference is that effluents are reported on a plant basis rather than a reactor basis. In order to revert to a reactor basis, which is the way noncompliances and LER's are reported, the effluent releases were divided by the number of reactors per plant. In ranking the effluents and calculating Z-scores, only one reactor per plant was included in the calculations in order not to introduce artificial ties and dependencies. Remaining reactors at that plant site are then given the same Z-scores as the included reactor.

An overall measure for effluent releases is obtained by first summing the ranks for a licensee across the five categories of release. If licensees are homogeneous with respect to effluent releases, then the ranks attained should behave like ranks obtained under the model of randomness and independence - randomness within each category and independence among categories. Under this model, for ranks covering n licensees, the mean and standard deviation of the sum of ranks can be obtained, namely $5(n + 1)/2$ and $\sqrt{5(n^2 - 1)/12}$, respectively. Thus, the Z-score for effluent releases used in the analysis is

$$Z = \frac{\frac{5(n+1)}{2} - \text{Rank Sum}}{\sqrt{\frac{5(n^2 - 1)}{12}}}$$

5. Personnel Exposures

Personnel exposure data for each licensee are reported as a histogram, a table giving the number of personnel who during the year (only annual reporting is required) received exposures (in rems) in particular successive exposure ranges. From these data, many summary statistics could be calculated and used as performance measures. The analysis of 1974 data (1975 data have not been analyzed and 1976 data are not available) led to a measure which reflects the upper end of the histogram, rather than its center as, for example, an average exposure would. The measure chosen is the percentage of personnel among all personnel who received measurable exposure who received 3 rems or more. Z-scores for personnel exposure are then calculated from the mean and standard deviation of this statistic. Personnel exposures are reported on a plant basis rather than on a reactor basis. In calculating the mean and variance, only one value for each plant is included and then the multiple reactors at a plant site are all given the same Z-score. Because any licensee can control the percentage of personnel receiving 3 or more rems and because the Z-score for personnel exposures receives a small weight relative to those of noncompliance and LER's in obtaining an overall performance measure, no consideration is given to possible adjustments of the Z-score for personnel exposures. One refinement which will be considered in the event that this percentage statistic varies erratically among licensees is to replace the mean and standard deviation used in calculating Z by measures less susceptible to erratic variation, such as the median and median absolute deviation.

6. Overall Performance

The analyses described in the previous four sections yield Z-scores pertaining to noncompliance, LER's, effluents, and personnel exposures. All that is required to obtain an overall performance measure is to assign weights to these Z-scores. The weights which have been selected are in the ratio 9:3:2:1, in the same order in which the performance measures were just listed. Some results are given in Appendix C pertaining to the sensitivity of the overall Z-score to, and the implications of, this choice of weights.

Appendix B

Results of Analysis of 1976 Data, January - June

A. Introduction

This appendix presents details and results of the analysis of licensee performance data for the period of January to June 1976 (abbreviated FH76). The order of presentation is the same as Chapter 3 - noncompliances, LER's, effluent releases, and personnel exposures. Within each performance category the results for PWR's are given first, the results for BWR's second. Following these sections, the overall results are given. For ease of reading, the tables of results given in Chapter 3 are repeated in this appendix.

The reactors included in the analysis are those which went into commercial operating prior to 1 January 1976 and which were not shut down nearly all of 1976. This latter condition means Indian Point 1 and Browns Ferry 1 and 2 are excluded. Thirty PWR's and 21 BWR's satsify this criterion. The data from those 51 reactors are tabulated in the sections which pertain to their analysis.

B. Analysis of Noncompliances

a. PWR's

Previous licensee performance evaluations, based on the 1974 and 1975 data, considered only the total noncompliances in each of the three severity categories - violations, infractions, and deficiencies. Much more detailed information is available, though, and it was decided to use some of it in the analysis of 1976 data. In particular, the three-letter "766" codes by which each item of noncompliance is labeled were grouped into six categories. Table 1 depicts the resulting assignment. Descriptions of the events corresponding to each entry in Table 1 are given in the IE Manual Chapter 0535.

TABLE 1

"766" CODES, BY CATEGORIES

Administrative Control		Operations		Emergency Planning	Radiological Protection and Control	Safeguards	Quality Assurance
AEA	FJA	DAD	FDR	FEL	ABA	NCK	VTA
AEB	FJB	DAE	FJC	FEP	ABB	RKA	VTB
AED	FJE	DAF	FJD	FEM	ABC	RLC	VTC
ALA	FJG	DAG	FJF	FPD	ABD	RLD	VTD
ALB	FJJ	DAH	FJH		ABE	RLE	VTE
ALC	FJL	DAJ	FJK		ABF	RLF	VWA
ALD	FJP	DAK	FJM		ABG	RLH	VWB
ALE	FPC	DAL	FJN		ABH	RLJ	VWC
ALF	FPE	DCG	FJR		ABI	RLK	VWD
ALG	FPF	DDA	FJS		ABJ	RLL	VWE
AMA	FPG	DDB	FMY		ADA	RMA	VWF
AMB	GHA	DDC	GAA		ADB	RMB	VWG
AMC	GHB	DDH	GAB		ADC	RMC	
APA	GHD	FAA	GAC		ADD	RMD	
ARA	HAA	FAB	GAE		AEC	RME	
ARB	HAC	FBA	GAF		ARD	RMH	
ARC	HAD	FBB	FBA		ARH	RMK	
ARE	HAE	FCA	GBB		ARM	RML	
ARF	HAF	FCB	GBC		ARY	RPA	
ARG	HAG	FCC	GCA		ASA	RPB	
ARI	HAH	FCD	GCB		ASB	RPC	
ARJ	HAI	FCE	GCC		ASC	RRA	
ARK	HAK	FCF	GDA		ASD	RRB	
ARR	HAL	FCG	GDB		ASE	RRC	
ARS	HAM	FCH	GDC		ASF		
ART	HAN	FCJ	GFA		ASG		
ARV	HAP	FCK	GFB		ASH		
DAW	HAR	FCL	JAA		ASI		
DAY	HAS	FCM	JAB		ASJ		
DDD	HAY	FCN	JAC		ASK		
DDE	RAA	FCP	JAD		BAA		
DDF	RAB	FCR	JAY		BAB		
DDG	RAC	FCS			BAC		
DEC	RAD	FDA			BAY		
DFA	RKB	FDB			FAC		
DFB	RPH	FDC			FBC		
DFC		FDD			FCN		
DFD		FDE					
FEA		FDF					
FEB		FDG					
FEC		FDH					
FED		FDJ					
FEE		FDK					
FEH		FDL					
FEK		FDM					
FER		FDN					
FES		FDP					

Table 2 lists the FH76 noncompliance frequencies of the 30 PWR's being considered by cause and severity categories. (The reactors are listed in docket number order.) Also listed are the totals across cause categories and the inspection effort as measured by total inspection time in and out of office. This latter is the independent, or explanatory, variable for which adjustment is meaningful and which has been indicated by previous analyses. Only two violations, both at Zion-1, occurred during the period of interest so no analysis of violations is feasible. The dependent variable first considered in detail is infraction frequency.

a.1 Infractions

Figure 1 shows total infractions for a reactor plotted versus inspection time for a plant. For example, the Zion-1 infractions (19) and the Zion-2 infractions (13) are plotted versus the total inspection time for those two reactors, 828 + 551 = 1379. The reason for using the plant total rather than the amount of time attributed to each reactor is that in many instances a single event can result in multiple noncompliance findings. Examination of a detailed listing of noncompliances at several multi-reactor plants indicated that about 90% of a reactor's noncompliances were also cited against the other reactors at the plant. Thus, the effective inspection time, with respect to a reactor's noncompliance, is more nearly the total plant inspection time than the portion of that time allotted to a particular reactor. This is the basis for the decision to use plant inspection time as an independent variable. Note from Table 2 that there are instances where one reactor from a multi-reactor plant met the criterion for inclusion in the analysis while another reactor at the plant did not, e.g., Indian Point and Calvert Cliffs. In these cases only the inspection time for the single reactor was considered.

Figure 1 indicates (not unexpectedly) that noncompliances increase with increasing inspection time. Various parametric assumptions could lead to an optimal fit of this apparent relationship. However, for simplicity and because these data may not be consistent with the required parametric assumptions, we fit a line nonparametrically. In particular we choose a line which passes through the origin and bisects the data. The line $\hat{INF} = 1.1T$, where T = Inspection Hrs./100, satisfies this criterion. (The 13 reactors at multi-reactor sites divide 7 and 6 about this line so the treatment accorded them is supported by this fit. That is, using plant total inspection time does not appear to introduce any bias.)

TABLE 2*

FH76 NONCOMPLIANCES: PWR'S

Reactor	Admin. Control			Oper- ations			Emerg. Plan.			Rad. Prot. & Control			Safe- guards			Quality Assur.			Total			Insp. Hours
	Severity			Severity			Severity			Severity			Severity			Severity			Severity			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
Yankee Rowe		2			1													3		3	3	434.5
San Onofre #1		1												5				1		6	1	286.5
Connecticut Yankee		5	2		3	2		1			2			2						13	4	519.0
Ginna					2									1				1		3	1	619.9
Indian Point #2		1	4		1	1		4			1	1		4						11	6	874.0
Turkey Point #3		3	2		2	1														5	3	284.1
Turkey Point #4		3	2		2	1														5	3	298.0
Palisades		5	2		4									6			2	1		17	3	690.5
Robinson #2		2	1		3										1			1		5	3	395.0
Point Beach #1						2					1			2						[illegible]	2	280.1
Oconee #1		1	1		4	2					2			3	1		1			11	4	376.9
Oconee #2		1			4	2					2			3	1		1			11	3	218.8
Surry #1		2	5		6						1			3	1		3			15	6	440.5
Surry #2		2	4		2						2			3	1		3			12	5	472.1
Prairie Island #1		2	3		3	3					1			5						11	6	336.0
Ft. Calhoun		2	1		1															3	1	349.0
Oconee #3		2			3	2					2			3	1		1			11	3	297.5
Three Mile Island#1		1	1		4	1					1			2	1		1	1		9	4	858.0
Zion #1		3	2		13	3		1		2	2								2	19	5	827.5
Point Beach #2			1		1	2								2						3	4	[illegible]
Zion #2		4	3		7	7					1						1			13	10	550.5
Kewaunee			1		1	2					2									3	3	470.5
Prairie Island #2			1		1	3					1			5						7	5	320.5
Maine Yankee		3	3		1	1								1	2			1		5	7	389.5
Rancho Seco #1					1	2												1		1	3	261.0
Arkansas #1		2	2		2													1		4	1	244.0
Cook #1		3	7		3	3			1											6	11	515.7
Calvert Cliffs #1														4						4		380.5
Millstone #2			3		4										3					4	6	672.5
Trojan			1		1	1					1	1					1	2		3	5	311.5
Total	0	50	52	0	80	41	0	6	1	2	22	2	0	54	12	0	14	13	2	226	121	

*Severity 1 = Violation
Severity 2 = Infraction
Severity 3 = Deficiency

B-5

REACTOR INFRACTIONS

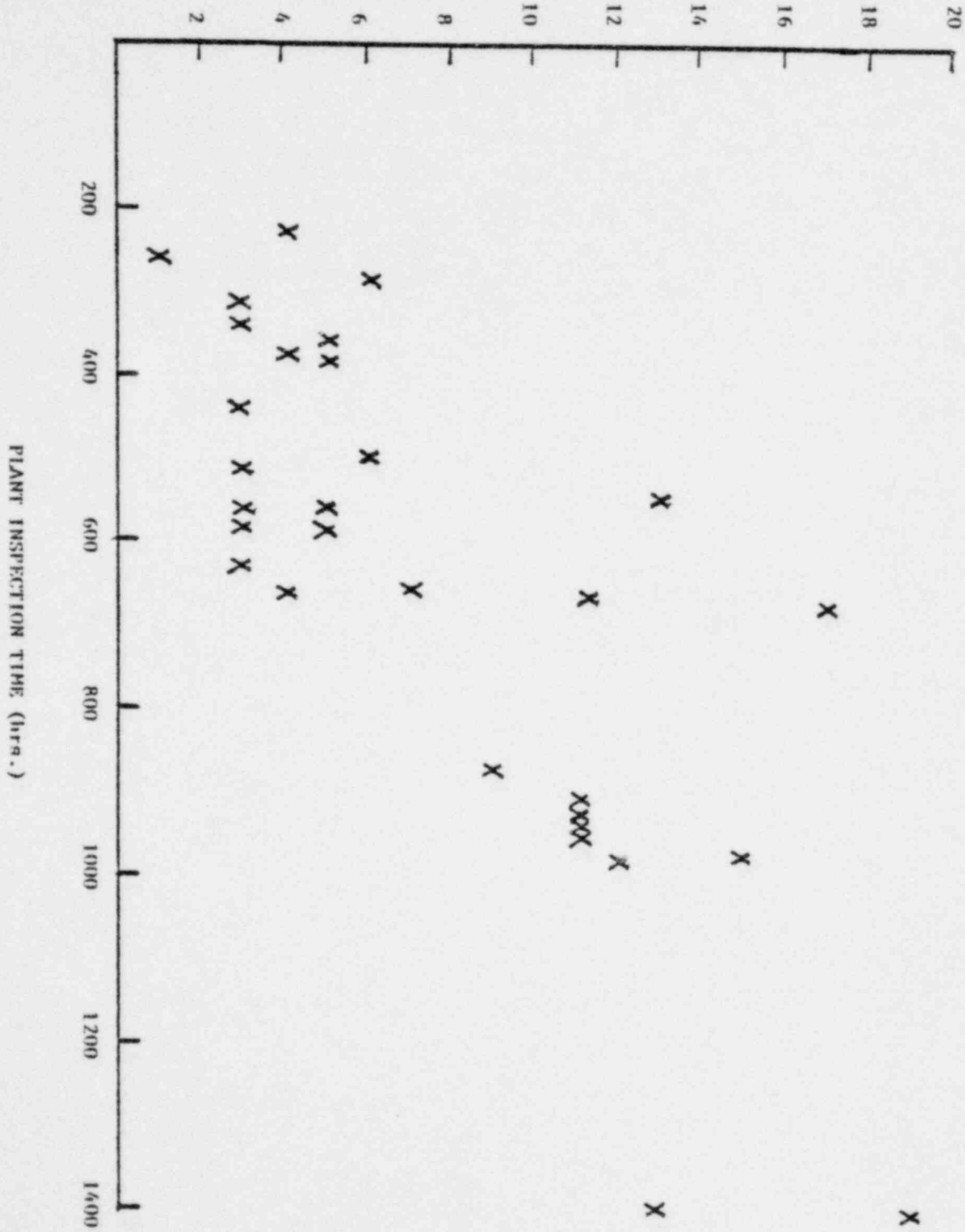


Figure 1. PWR Infractions vs. Plant Inspection Time: FY76

The fit of this line to the data can be assessed by calculating the sum of $(INF - \hat{INF})^2/\hat{INF}$ over the 30 PWR's and comparing the resulting quantity, deonted by χ^2, to a chi-square distribution with 28 degrees of freedom (df). These data yield $\chi^2 = 42.2$ which falls at about the upper 5% point on the chi-square distribution with 28 degrees of freedom. This is not unusually large for data subject to as many sources of variation as these, and in fact about half of this χ^2 value is attributable to two reactors, Connecticut Yankee and Palisades, so it appears that adjusting infractions for inspection time is all the adjustment that is required. (By way of contrast, the χ^2 value obtained assuming that the variation in infraction frequency among reactors is "random" - not attributable to inspection effort or any other independent variable - equals 89.4 on 29 degrees of freedom, which is unusually large, so a real reduction is accomplished by adjusting for inspection time.)

If the analysis were to be based on only total infractions, Z-scores would be calculated by

$$Z = \frac{1.1T - INF}{\sqrt{1.1T}}$$

However, the Z-scores of interest in this analysis, and which will now be developed, are those corresponding to the six types of infractions tabulated.

The total number of infractions, by type, are as follows:

Type (Abbreviation)	Number of Infractions	% of Total
Administrative Control (AC)	50	22%
Operations (OP)	80	35
Emergency Planning (EP)	6	3
Radiological Protection and Control (RP)	22	10
Safeguards (SG)	54	24
Quality Assurance (QA)	14	6
Total	226	100

The infraction frequencies for three of infraction types - EP, RP, QA, - are quite small, less than 1 per reactor for an average, so that calculated Z-scores may be inappropriate. Thus for subsequent analysis these three will be combined, the combined infractions being labeled Remainder (RE).

The estimated expected number of infractions at a reactor is given about by $\widehat{INF} = 1.1T$. From the above total infractions, one would expect 22% of these to fall in the AC category. Thus, the estimated expected number of AC infractions is .22 (1.1T) = .25T and the Z-score for AC will be defined as

$$Z_{INF(AC)} = \frac{.25T - INF(AC)}{\sqrt{.25T}}$$

where INF(AC) is the observed number of AC infractions. Similarly, the estimated expected infractions for the other three categories are obtained as follows: OP - .40T, SG - .25T, Remainder - .20T. Table 3 lists the Z-scores which result. Also given is an overall Z-score for infractions obtained by summing the Z's for the four categories, then dividing by 2. (This division is chosen to that the correct frame of reference for the overall scores will again be the standard normal distribution.) Weighting the Z's other than equally can also be done if a relative importance of infraction types can be established.

a.2 <u>Deficiencies</u>

A plot of total deficiencies versus site inspection time indicates, as with infractions, that inspection time appears to contribute to the observed variation of deficiencies. Fitting a line through the origin nonparametrically yields $\widehat{DEF} = .6T$. Note that the total number of deficiencies reported, 121, is 54% of the total number of infractions reported, 226. Fifty-four percent of 1.1, the slope of the line fit to infractions, is .6, so the two models are in good agreement. The χ^2 value pertaining to the fit of this model equals about 50, which, on 28 degrees of freedom, is unusually large (there is less than a probability of .01 of obtaining this large a value by chance alone). A sizeable portion of this χ^2 value is attributable to one reactor, Connecticut Yankee, and when that observation is deleted, the resulting χ^2 value is about 30, which, for 27 degrees of freedom, is not at all unusual. It should be noted that the χ^2 value obtained assuming no relationship between inspection time and deficiencies becomes acceptably small if two observations, Connecticut Yankee and Zion-1, are deleted, so it could be argued that no adjustment of deficiencies for inspection time is needed. However, because of the similarity of patterns between infractions and deficiencies and for the sake of consistency of the two analyses, that adjustment will be made.

TABLE 3

Z-SCORES FOR INFRACTIONS: PWR'S

Reactor	Admin. Control	Operations	Safeguards	Remaining Types*	Combined
Yankee Rowe	-0.9	0.6	1.0	0.9	0.8
San Onofre #1	-0.3	1.1	-5.1	0.8	-1.8
Connecticut Yankee	-3.3	-0.6	-0.6	-1.9	-3.2
Ginna	1.2	0.3	0.4	1.1	1.6
Indian Point #2	0.8	1.3	-1.2	-2.5	-0.8
Turkey Point #3	-1.3	0.2	1.2	1.1	0.6
Turkey Point #4	-1.3	0.2	1.2	1.1	0.6
Palisades	-2.5	-0.8	-3.3	-0.5	-3.5
Robinson #2	-1.0	-1.1	1.0	0.9	-0.1
Point Beach #1	1.2	1.5	-0.4	0.2	1.2
Oconee #1	0.8	-0.2	-0.5	-0.9	-0.4
Oconee #2	0.8	-0.2	-0.5	-0.9	-0.4
Surry #1	0.2	-1.2	-0.5	-1.6	-1.6
Surry #2	0.2	0.9	-0.5	-2.3	-0.9
Prairie Island #1	-0.3	-0.2	-2.6	0.3	-1.4
Ft. Calhoun	-1.2	0.3	0.9	0.3	0.4
Oconee #3	0.2	0.3	-0.5	-0.9	-0.5
Three Mile Island #1	0.8	-0.3	0.1	-0.2	0.2
Zion #1	0.2	-3.2	1.9	-0.1	-0.6
Point Beach #2	1.2	0.9	-0.4	1.1	1.4
Zion #2	-0.3	-0.6	1.9	0.5	0.7
Kewaunee	1.1	0.6	1.1	-1.1	0.9
Prairie Island #2	1.3	1.0	-2.6	0.3	-0.0
Maine Yankee	-2.1	0.4	-0.0	0.9	-0.4
Rancho Seco #1	0.8	0.0	0.8	0.7	1.2
Arkansas #1	-1.3	-1.0	0.8	0.7	-0.7
Cook #1	-1.5	-0.7	1.1	1.0	-0.0
Calvert Cliffs	1.0	1.2	-3.1	0.9	-0.0
Millstone #2	1.3	-0.8	1.3	1.2	1.5
Trojan	-0.2	0.2	0.9	-0.5	0.2

*Remaining Types are Emergency Planning, Radiation Protection and Control, and Quality Assurance infractions combined.

The distribution of deficiency types is as follows:

Type	Number of Deficiencies	% of Total
Administrative Control	52	43%
Operations	41	34
Emergency Planning	1	1
Rad. Protection and Control	2	2
Safeguards	12	10
Quality Assurance	13	11

For the purpose of calculating Z-scores by types, the last four types will be grouped under the label Remainder. The nominal values obtained from partitioning the estimated total expected deficiencies, .6T, are thus:

Cause Category	Nominal Value
AC	.25T
OP	.20T
RE	.15T

Table 4 gives the resulting Z-scores as well as an overall Z-score for deficiencies defined as

$$Z_{DEF} = (Z_{DEF(AC)} + Z_{DEF(OP)} + Z_{DEF(RE)})/\sqrt{3}$$

a.3 Combined Measure of Noncompliance

The results of the sensitivity study described in Appendix C and consideration of the relative severity of items of noncompliance led to a combined measure of noncompliance of

$$Z_{NC} = (-10V + 5Z_I + Z_D)/\sqrt{26}$$

Analysis of previous years' data included the use of principal components analysis to select a weighting. However, because in FY76 only one reactor had any violations (V), performing a principal components analysis on V, Z_I, and Z_D is not appropriate so the present data cannot be used to arrive at a different weighting. (Principal

TABLE 4

Z-SCORES FOR DEFICIENCIES: PWR'S

Reactor	Admin. Control	Operations	Remaining Types*	Combined
Yankee Rowe	1.0	0.9	-2.9	-0.5
San Onofre #1	0.8	0.8	-0.9	0.4
Connecticut Yankee	-0.6	-0.9	0.9	-0.4
Ginna	1.2	1.1	-0.1	1.3
Indian Point #2	-1.2	0.6	0.3	-0.2
Turkey Point #3	-0.5	0.2	0.9	0.4
Turkey Point #4	-0.5	0.2	0.9	0.4
Palisades	-0.2	1.2	0.0	0.6
Robinson #2	-0.0	0.9	-1.8	-0.5
Point Beach #1	1.2	-0.8	0.9	0.8
Oconee #1	0.8	-0.2	0.3	0.6
Oconee #2	1.5	-0.2	0.3	0.9
Surry #1	-1.8	1.4	0.3	-0.1
Surry #2	-1.1	1.4	0.3	0.3
Prairie Island #1	-1.1	-1.5	1.0	-0.9
Ft. Calhoun	-0.1	0.8	0.7	0.8
Oconee #3	1.5	-0.2	0.3	0.9
Three Mile Island #1	0.8	0.5	-0.6	0.4
Zion #1	0.8	-0.1	1.4	1.2
Point Beach #2	0.4	-0.8	0.9	0.3
Zion #2	0.2	-2.6	1.4	-0.5
Kewaunee	0.2	-1.1	0.8	-0.0
Prairie Island #2	0.5	-1.5	1.0	0.0
Maine Yankee	-2.1	-0.2	-3.2	-3.2
Rancho Seco #1	0.8	-2.0	-1.0	-1.3
Arkansas #1	-1.8	0.7	-1.0	-1.2
Cook #1	-5.0	-2.0	-0.3	-4.2
Calvert Cliffs #1	1.0	0.9	0.8	1.5
Millstone #2	-1.0	1.2	-2.0	-1.1
Trojan	-0.2	-0.5	-3.7	-2.6

*Remaining Types are Emergency Planning, Radiation Protection and Control, Quality Assurance, and Safeguards deficiencies combined.

TABLE 5

COMBINED MEASURE OF NONCOMPLIANCE: PWR'S

Reactor	Viol.	Z_I	Z_D	Z_{NC}
Yankee Rowe	0	0.1	-0.5	0.7
San Onofre	0	-1.8	0.4	-1.7
Connecticut Yankee	0	-3.2	-0.4	-3.2
Ginna	0	1.6	1.3	1.8
Indian Point 2	0	-.8	-0.2	-0.8
Turkey Point 3	0	0.6	0.4	0.7
Turkey Point 4	0	0.6	0.4	0.7
Palisades	0	-3.5	0.6	-3.3
Robinson	0	-0.1	-0.5	-0.2
Point Beach 1	0	1.2	0.8	1.3
Oconee 1	0	-0.4	0.6	-0.3
Oconee 2	0	-0.4	0.9	-0.2
Surry 1	0	-1.6	-0.1	-1.6
Surry 2	0	-0.9	0.3	-0.3
Prairie Island 1	0	-1.4	-0.9	-1.5
Fort Calhoun	0	0.4	0.8	0.5
Oconee 3	0	-0.5	0.9	-0.3
Three Mile Island 1	0	0.2	0.4	0.3
Zion 1	2	-0.6	1.2	-4.3
Point Beach 2	0	1.4	0.3	1.4
Zion 2	0	0.7	-0.5	0.6
Kewaunee	0	0.9	0.0	0.9
Prairie Island 2	0	0.0	0.0	0.0
Maine Yankee	0	-0.4	-3.2	-1.0
Rancho Seco 1	0	1.2	-1.3	0.9
Arkansas 1	0	-0.7	-4.2	-1.5
Cook 1	0	0.0	-4.2	-0.8
Calvert Cliffs 1	0	0.0	1.5	0.3
Millstone 2	0	1.5	-1.1	1.3
Trojan	0	0.2	-2.6	-0.3

components analysis of just Z_I and Z_D would lead only to considering the sum or difference of Z_I and Z_D, niether of which combination is consistent with the fact that infractions are a more severe item of noncompliance than deficiencies.) Table 5 gives values of V, Z_I, Z_D, and Z_{NC}.

b. BWR's

Table 6 lists the FH76 noncompliance frequencies of the 21 BWR's being considered by cause and severity categories along with total noncompliances and inspection hours. No BWR incurred a violation during FH76, so the first performance measure considered is total infractions.

b.1 Infractions

A plot of infractions versus inspection hours does not show nearly as strong a relationship for BWR's as it did for PWR's. This could happen because no relationship exists or because it is masked by other variables. We pursue the latter possibility. The line $\widehat{INF} = 1.1T$ nearly bisects the BWR data as it did the PWR data so there is some support for again adjusting for inspection hours by this relationship. The analysis will consider this adjustment first, then seek other independent variables which might explain the residual variation remaining after adjusting infractions for the effect inspection hours.

Previous analyses of 1974 and 1975 BWR data showed some patterns of variation in the data seemingly associated with age or design. In particular, reactors which came after Dresden 3 showed a different frequency of infractions than did reactors up to and including Dresden 3. The FH76 data reflect a similar separation. Examination of the Z-scores, $Z = (1.1T - INF)/\sqrt{T.1T}$, showed that among post-Dresden-3 reactors, the newer reactors tended to have lower Z-scores (i.e., poorer relative performance) than the older reactors. This could reflect problems associated with beginning operations or it could reflect more extensive technical specifications for newer reactors and hence more opportunities for noncompliance. A plot of Z-scores for the post Dresden-3 reactors versus commercial operation (C.O.) date shows a marked trend and regression analysis (least squares and nonparametric) led to the fitted line.

$$\hat{Z} = 1.8 + .85 \text{ Age},$$

where Age = 1976 - C.O. date, recorded to the nearest half year. Adjusted Z-scores for these 12 reactors are then obtained by subtracting $\hat{Z}$ for each reactor from its original Z-score which was obtained from adjusting only for inspection hours. The result of this is that considerably more homogeneity of Z-scores is obtained. After adjusting only for inspection hours, the sum of squares of Z-scores for these 12 reactors was 27.2. After further adjusting for age, the sum of squares of Z-scores is

Table 6
FH76 NONCOMPLIANCES: BWR'S

Reactor	Admin. Control			Oper-ations			Emerg. Plan.			Rad. Prot. & Control			Safe-guards			Quality Assur			Total			Insp. Hours
	Severity			Severity			Severity			Severity			Severity			Severity			Severity			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
Dresden #1		2	2		1						1			4			1			9	2	325.5
Humboldt Bay														4			4	1		8	1	338.5
Big Rock Point			2		5									2	1		1			8	3	679.0
Oyster Creek		3	3		1	1					1	1		2	2		9	9		16	16	831.0
Nine Mile Point #1		2	2		3	2					2						4	5		11	9	810.5
Dresden #2		1	2			1								3				1		4	4	600.2
Millstone #1		3	4		3						3			2	1					11	5	821.0
Dresden #3			1		1									3						4	1	316.0
Quad Cities #1		1	1		3	1					1			2						7	2	500.5
Monticello		1	1											1				1		2	[illegible]	594.0
Quad Cities #2			2		4						2			2				1		8	3	303.0
Vermont Yankee		1			1									1	1					3	1	489.5
Peach Bottom #2		4	2		8										1					12	3	731.5
Peach Bottom #3		2	2		4										1					6	3	403.0
Pilgrim #1			1			1											1			1	2	779.0
Cooper		2	1		1												6	2		9	3	434.0
Hatch #1			2		6	1								1			1			8	3	526.2
Brunswick #2		1	2		5	6					2			2	2		2	1		12	11	606.5
Duane Arnold			1		4	3		2									2			8	4	768.0
Fitzpatrick			4			3			1		1			9	1		1	4		11	13	580.5
LaCrosse		1	1								2							1		3	2	577.5
Total	0	24	36	0	50	19	0	2	1	0	15	1	0	39	10	0	37	26	0	167	93	

*Severity 1 = Violation
Severity 2 = Infraction
Severity 3 = Deficiency

only 12.8. The appropriate frames of reference for these results are the chi-square distributions with 11 and 9 degrees of freedom, respectively, and comparison of these results to these distributions shows that the value of 27.2 is considerably larger than would be expected, while 12.8 is not. Thus, adjustment for age seems warranted and is sufficient to reduce the residual variation to an acceptable level.

It would be helpful if an adjustment could be applied to the other 9 BWR's which would account for much of the variation among them. However, it does not seem likely that such an adjustment can be found because of factors such as the following: Four of these reactors are much older and smaller than the rest and one of them is at a site which includes two newer and larger reactors. Thus infractions for these 9 reactors will be adjusted only for inspection time. In interpreting these Z-scores, one should bear in mind these characteristics. (In fact, interpretation of all the performance measures developed in this report should include consideration of pecularities or special circumstances which might affect performance. However, this does not mean that each reactor should be regarded as unique and thus not comparable to any other reactor. The fact that simple relationships have been found linking the performance of different reactors and the fact that the residual variation after accounting for these relationships has been reduced essentially to "random noise" mean that comparisons are possible.)

The following table gives the distribution of BWR infractions by type.

Type (Abbreviation)	Number of Infractions	% of Total
Administrative Control (AC)	24	14%
Operations (OP)	50	30
Emergency Planning (EP)	2	1
Rad. Protection and Control (RP)	15	9
Safeguards (SG)	39	23
Quality Assurance (QA)	37	22
TOTAL	167	

This distribution is similar to that of PWR's, except for QA infractions which are more numerous among BWR's. Because of this similarity, and for consistancy with the PWR analysis the same grouping of types will be followed, namely AC, OP, SG, and RE. The percentages associated with these are 14, 30, 23, and 32%

respectively. Apportioning the "nominal" number of infractions, 1.1T, by these percentages yields the following nominal values for infractions by type:

Type	Nominal No. of Infractions
AC	.15T
OP	.35T
SG	.25T
RE	.35T

Thus, Z-scores for AC infractions will be given by

$$Z_{INF(AC)} = \frac{1.15T - INF(AC)}{\sqrt{15T}} - (-1.8 + .85\ Age),$$

for the 12 reactors requiring an age adjustment, and by

$$Z_{INF(AC)} = \frac{.15T - INF(AC)}{\sqrt{.15T}}$$

for the other 11 BWR's. Similar expressions yield Z-scores for the other infraction types. An overall Z-score for infractions will be obtained by adding the Z's for the four types, then dividing by two. The results of all these calculations are given in Table 7.

b.2 Deficiencies

A plot of total deficiencies versus site total inspection time for the 21 BWR's under consideration does not show as much evidence of a trend as did the PWR's. However, as with infractions, BWR deficiency frequencies are subject to several sources of variation, so a two-variable plot might not be expected to show any pattern. The approach which will be followed will be to calculate Z-scores for deficiencies, based on nominal values of .6T, the same as used for PWR's, and then seek ways to adjust these Z-scores for other variables. This is the same approach as was followed for BWR infractions (the previous section) where it was found that among post Dresden-3 design reactors, the Z-scores were associated with reactor age.

TABLE 7

Z-SCORES FOR INFRACTIONS: BWR'S

Reactor	Admin. Control	Operations	Safeguards	Remaining Types*	Combined
Dresden 1	-0.1	1.6	-0.5	1.1	1.1
Humboldt Bay	0.7	1.1	-3.4	-2.6	-2.1
Big Rock Point	1.0	-1.7	-0.2	0.9	-0.2
Oyster Creek	-1.6	1.1	0.1	-4.2	-2.3
Nine Mile Point 1	-0.7	-0.1	1.4	-1.9	-0.6
Dresden 2	0.6	2.1	0.1	2.1	2.4
Millstone 1	-1.6	-0.1	0.0	-0.1	-0.9
Dresden 3	1.4	1.6	0.1	2.1	2.6
Quad Cities 1	-0.6	-0.9	-0.8	0.3	-1.0
Monticello	-2.1	-0.6	-1.6	-0.6	-2.4
Quad Cities 2	0.3	-1.5	-0.8	-0.3	-1.2
Vermont Yankee	-1.5	-0.7	-1.0	0.1	-1.5
Peach Bottom 2	-1.7	-1.9	1.6	1.9	0.1
Peach Bottom 3	0.3	0.5	2.2	2.5	2.7
Pilgrim	-0.1	0.5	0.2	-0.2	0.2
Cooper	-1.2	0.9	1.5	-3.1	-0.9
Hatch	2.7	-1.3	2.1	2.4	3.0
Brunswick 2	1.3	-0.6	1.0	0.1	0.9
Duane Arnold	2.1	0.2	2.4	0.2	2.4
Fitzpatrick	2.3	2.8	-4.9	1.4	0.9
Lacrosse	-0.1	1.4	1.2	0.0	1.2

*Remaining Types are Emergency Planning, Radiation Protection and Control, and Quality Assurance infractions combined.

A plot of Z-scores, where

$$Z = \frac{.6T - DEF}{\sqrt{.6T}}$$

versus age shows a positive trend, but there are 4 points (out of 14) which depart considerably from the main pattern and the trend is not as pronounced as with infractions. Because deficiencies are not weighted heavily compared to other performance measures, it is deemed not necessary to make this further adjustment and no other independent variables will be considered at this time.

The distribution of BWR deficiencies by types is the following:

Type	Number of Deficiencies	% of Total
AC	36	39%
OP	19	20
EP	1	1
RP	1	1
SG	10	11
QA	26	28
Total	93	

This distribution differs from that of the PWR's in that more QA and fewer OP deficiencies were incurred by BWR's. For the sake of consistency with the PWR analysis, Z-scores will be obtained for three categories, AC, OP, and RE - the other four deficiency types combined. It should be noted that the low frequency of OP deficiencies may lead to distorted Z-scores but because those scores will be added to others, including those obtained from infractions which will be more heavily weighted, the final result should not be unduly distorted.

The percentage of deficiencies corresponding to AC, OP, and RE are 39, 20, and 41%, respectively. Partitioning the nominal total number of deficiencies, .6T, accordingly, yields the following:

Type	Nominal No. Deficiencies
AC	.25T
OP	.10T
RE	.25T

Thus, Z-scores for AC deficiencies will be given by

$$Z_{DEF(AC)} = \frac{.25T - DEF(AC)}{\sqrt{.25T}}$$

Similar expressions yield Z-scores for the other two deficiency types. An overall Z-score for deficiencies will be obtained by by adding the Z's for the three types, then dividing by $\sqrt{3}$. The results of these calculations are given in Table 8.

b.3 Combined Measure of Noncompliance

Table 9 gives values of Z_I, Z_D, and Z_{NC}, where

$$Z_{NC} = (-10V + 5Z_I + Z_D)/\sqrt{26}$$

is the selected combined score for noncompliance. (Recall that no BWR incurred a violation in FH76 so V = 0 throughout.)

C. Analysis of LER's

a. PWR's

Table 10 lists the FH76 LER frequencies, of the 30 PWR's being considered, by whether they were classified as personnel or procedural error, and also lists the total LER's over these categories. LER's in other categories, such as design error or component failure, will not be considered as licensee performance measures. The reason for concentrating on personnel and procedural errors is to diminish the plant-specific nature of reporting requirements and to emphasize events under the licensee's control. I.e., this is another way of "adjusting" for the effect of factors not under the licensee's control.

The analysis of LER's proceeds just as that of noncompliances did. Thus first considered is the variation of total LER's. This variation is larger than one would expect by chance. Omitting the obvious outlier - Trojan which reported 14 LER's while no other PWR reported more than 8 - reduces the variation to what is not greatly unexpected by chance alone, (the χ^2 value is 40.9, which on 28 degrees of freedom corresponds to about the 5% level of significance.) However, examination of the data in Table 10 shows some evidence of an age trend - later reactors, in terms of docket numbers, appear to have more LER's than earlier reactors. There are various ways to measure the age of a reactor,

TABLE 8

Z-SCORES FOR DEFICIENCIES: BWR'S

Reactor	Admin. Control	Operations	Remaining Types*	Combined
Dresden 1	0.6	1.1	1.8	2.0
Humboldt Bay	0.9	0.6	-0.2	0.8
Big Rock Point	-0.2	0.8	0.5	0.6
Oyster Creek	-0.6	-0.2	-6.9	-4.5
Nine Mile Point 1	0.0	-1.3	-2.1	-2.0
Dresden 2	0.6	0.2	1.2	1.2
Millstone 1	-1.4	0.9	0.7	0.1
Dresden 3	0.9	1.1	1.8	2.2
Quad Cities 1	0.7	-0.2	1.4	1.1
Monticello	0.4	0.8	0.4	0.9
Quad Cities 2	0.0	0.9	0.7	0.9
Vermont Yankee	1.1	0.7	0.2	1.2
Peach Bottom 2	0.5	1.1	1.1	1.6
Peach Bottom 3	0.5	1.1	1.1	1.6
Pilgrim	0.7	-0.3	1.4	1.0
Cooper	0.1	0.7	-0.9	-0.1
Hatch	-0.6	-0.7	1.2	-0.1
Brunswick 2	-0.4	-6.9	-1.2	-4.9
Duane Arnold	0.7	-2.6	1.4	-0.3
Fitzpatrick	-2.1	-3.2	-3.8	-5.3
Lacrosse	0.4	0.8	0.4	0.9

*Remaining Types are Emergency Planning, Radiation Protection and Control, Quality Assurance, and Safeguards deficiencies combined.

TABLE 9

COMBINED MEASURE OF NONCOMPLIANCE: BWR'S

Reactor	Z_I	Z_D	Z_{NC}
Dresden 1	1.1	2.0	1.5
Humboldt Bay	-2.1	0.8	-1.9
Big Rock Point	-0.2	0.6	-0.2
Oyster Creek	-2.3	-4.5	-3.1
Nine Mile Point 1	-0.6	-2.0	-1.0
Dresden 2	2.4	1.2	2.6
Millstone	-0.9	0.1	-0.9
Dresden 3	2.6	2.2	3.0
Quad Cities 1	-1.0	1.1	-0.8
Monticello	-2.4	0.9	-2.2
Quad Cities 2	-1.2	0.9	-1.0
Vermont Yankee	-1.5	1.2	-1.2
Peach Bottom 2	0.1	1.6	0.4
Peach Bottom 3	2.7	1.6	3.0
Pilgrim	0.2	1.0	0.4
Cooper	-0.9	-0.1	-0.9
Hatch	3.0	-0.1	2.9
Brunswick 2	0.9	-4.9	-0.1
Duane Arnold	2.4	-0.3	2.3
Fitzpatrick	0.9	-5.3	-0.2
Lacrosse	1.2	0.9	1.4

such as the dates construction permits or operating licenses were granted, or the date when a reactor began commercial operation. The latter is considered here because it was the criterion by which reactors were selected for this analysis.

Examination of a plot of LER frequencies versus date of commercial operation (C.O.) shows not so much of a smooth trend as a dichotomy. Reactors with C.O. dates prior to 1973 are one group and those with C.O. dates of 1973 and after are the other. In the former group, the average number of LER's is about 2.0, while in the latter, the average is about 4.0. Thus these two values will be used as nominal LER frequencies.

The 99 LER's divide 64:35 for personnel versus procedural errors. Applying this ratio to the nominal frequencies yields the following nominal values for determining Z-scores for LER's.

	Nominal Values	
	Personnel	Procedural
Pre 1/73 O.L.	1.3	.7
Post 1/73 O.L.	2.6	1.4

Z-scores, based on the appropriate nominal value and calculated as before by

$$Z = \frac{NOM - LER}{\sqrt{NOM}} ,$$

are given in Table 11. Also given is a combined LER Z-score calculated as

$$Z_{LER} = (Z_{PERS} + Z_{PROC})/\sqrt{2}$$

b. BWR's

Table 11 lists the FH76 LER frequencies of the 21 BWR's being considered, the frequencies being given for personnel errors and procedural errors as well as the total.

Examination of the LER frequencies as a function of age shows, as with the PWR's, a dichotomy. In this case, the division is more pronounced and occurs at a later date. In particular, the 17

Table 10

FH76 LER FREQUENCIES AND Z-SCORES: PWR'S

Reactor	Personnel	Procedural	Total	Personnel	Procedural	Combined
Yankee Rowe	2	1	3	-0.6	-0.4	-0.7
San Onofre	0	0	0	1.1	0.8	1.3
Connecticut Yankee	1	1	2	0.3	-0.4	-0.1
Ginna	2	1	3	-0.6	-0.4	-0.7
Indian Point 2	3	0	3	-0.3	1.2	0.6
Turkey Point 3	0	0	0	1.1	0.8	1.3
Turkey Point 4	0	0	0	1.6	1.2	2.0
Palisades	3	0	3	-1.5	0.8	-0.5
Robinson	3	0	3	-1.5	0.8	-0.5
Point Beach 1	0	0	0	1.1	0.8	1.3
Oconee 1	2	2	4	0.4	-0.5	-0.1
Oconee 2	3	0	3	-0.3	1.2	0.6
Surry 1	1	0	1	0.3	0.8	0.8
Surry 2	1	1	2	1.0	0.3	0.9
Prairie Island 1	4	2	6	-0.9	-0.5	-1.0
Fort Calhoun	3	1	4	-0.3	0.3	0.0
Oconee 3	3	2	5	-0.3	-0.5	-0.6
Three Mile Island	4	4	8	-0.9	-2.2	-2.2
Zion 1	3	3	6	-0.3	-1.4	-1.2
Point Beach 2	3	0	3	-0.3	1.2	0.6
Zion 2	1	0	1	1.0	1.2	1.6
Kewaunee	2	2	4	0.4	-0.5	-0.1
Prairie Island 2	1	2	3	1.0	-0.5	0.4
Maine Yankee	1	0	1	0.3	0.8	0.3
Rancho Seco	2	1	3	0.4	0.3	0.5
Arkansas	2	1	3	0.4	0.3	0.5
Cook	4	3	7	-0.9	-1.4	-1.6
Calvert Cliffs	1	0	1	1.0	1.2	1.6
Millstone	3	0	3	-0.3	1.2	0.6
Trojan	6	8	14	-2.1	-5.6	-5.4

Table 11

FH76 FREQUENCIES AND Z-SCORES: BWR'S

	LER Frequencies			Z-Scores		
Facility	Personnel	Procedural	Total	Personnel	Procedural	Combined
Dresden 1	0	0	0	1.5	0.8	1.6
Humboldt Bay	2	2	4	0.2	-1.6	-1.0
Big Rock Point	1	1	2	0.9	-0.4	0.4
Oyster Creek	2	0	2	0.2	0.8	0.7
Nine Mile Point 1	3	0	3	0.2	0.8	0.7
Dresden 2	6	2	8	-2.4	-1.6	-2.8
Millstone 1	2	2	4	0.2	-1.6	-1.0
Dresden 3	1	0	1	0.9	0.8	1.2
Quad Cities 1	3	0	3	1.5	-0.4	0.8
Monticello	0	0	0	1.5	0.8	1.6
Quad Cities 2	0	1	1	1.5	-0.4	0.8
Vermont Yankee	3	1	4	-0.5	-0.4	-0.6
Peach Bottom 2	7	1	8	-3.1	-0.4	-2.5
Peach Bottom 3	2	0	2	0.2	0.8	0.7
Pilgrim	3	1	4	-0.5	-0.4	-0.6
Cooper	1	0	1	0.9	0.8	1.2
Hatch	17	4	21	-1.4	-0.3	-1.2
Brunswick 2	14	5	19	-0.6	-0.8	-1.0
Duane Arnold	8	4	12	1.2	-0.3	0.6
Fitzpatrick	10	0	10	0.6	1.9	1.3
Lacrosse	0	0	0	1.5	0.8	1.6

reactors beginning commercial operation prior to 1975 form one group with a nominal LER frequency of 3.0. The four reactors which began commercial operation in 1975 averaged 15.5 LER's in FH76, so that value will be used as nominal for that small subgroup.

The 109 LER's reported by BWR's in FH76 divide 85:24 between personnel and procedural errors. Applying this ratio to the nominal LER totals yields 2.3 and 0.7 as nominal values for the 17 older reactors, 12.0 and 3.5 for the four newer reactors. Z-scores based on these nominal values are also given in Table 11 along with the combined Z-score for LER's.

D. Effluent Releases

Results on effluent releases are not presented because all licensees have not reported this information and because the data which have been reported are not yet in a suitable form for analysis.

E. Personnel Exposure

These data are reported on an annual basis so are not available for FH76.

F. Combined Measure of Licensee Performance

For FH76 only noncompliance and LER Z-scores are available for combining into an overall performance measure. The conclusion from Appendix C, on the choice of weighting factors, was that a ratio of 3:1 was to be used in combining these two performance categories. Tables 12 and 13 give values of

$$Z_{FINAL} = (3Z_{NC} + Z_{LER})/\sqrt{10}$$

for PWR's and BWR's, respectively. Tables 14 and 15 summarize these tables by categorizing Z_{FINAL} as follows:

Range of Z_{FINAL}	Category
$Z_{FINAL} > 1.0$	"Above Average"
$-1.0 \le Z_{FINAL} \le 1.0$	"Average"
$Z_{FINAL} < -1.0$	"Below Average"

Table 12

FH76 Overall Performance: PWR's

Reactor	Z_{NC}	Z_{LER}	Z_{FINAL}
Yankee Rowe	0.7	-0.7	0.4
San Onofre	-1.7	1.3	-1.2
Connecticut Yankee	-3.2	-0.1	-3.1
Ginna	1.8	-0.7	1.5
Indian Point 2	-0.8	0.6	-0.6
Turkey Point 3	0.7	1.3	1.1
Turkey Point 4	0.7	2.0	1.3
Palisades	-3.3	-0.5	-3.3
Robinson	-0.2	-0.5	-0.4
Point Beach 1	1.3	1.3	1.6
Oconee 1	-0.3	-0.1	-0.3
Oconee 2	-0.2	0.6	0.0
Surry 1	-1.6	0.8	-1.3
Surry 2	-0.8	0.9	-0.5
Prairie Island 1	-1.5	-1.0	-1.7
Fort Calhoun	0.5	0.0	0.5
Oconee 3	-0.3	-0.6	-0.5
Three Mile Island 1	0.3	-2.2	-0.4
Zion 1	-4.3	-1.2	-4.5
Point Beach 2	1.4	0.6	1.5
Zion 2	0.6	1.6	1.1
Kewaunee	0.9	-0.1	0.8
Prairie Island 2	0.0	0.4	0.1
Maine Yankee	-1.0	0.8	-0.7
Rancho Seco 1	0.9	0.5	1.0
Arkansas 1	-1.5	0.5	-1.3
Cook 1	-0.8	-1.6	-1.3
Calvert Cliffs 1	0.3	1.6	0.8
Millstone 2	1.3	0.6	1.4
Trojan	-0.3	-5.4	-2.0

Table 13

FH76 Overall Performance: BWR's

Reactor	Z_{NC}	Z_{LER}	Z_{FINAL}
Dresden 1	1.5	1.6	1.9
Humboldt Bay	-1.9	-1.0	-2.1
Big Rock Point	-0.2	0.4	-0.1
Oyster Creek	-3.1	0.7	-2.7
Nine Mile Point 1	-1.0	0.2	-0.9
Dresden 2	2.6	-2.8	1.6
Millstone	-0.9	-1.0	-1.2
Dresden 3	3.0	1.2	3.2
Quad Cities 1	-0.8	0.2	0.7
Monticello	-2.2	1.6	-1.6
Quad Cities 2	-1.0	0.8	-0.7
Vermont Yankee	-1.2	-0.6	-1.3
Peach Bottom 2	0.4	-2.5	-0.4
Peach Bottom 3	3.0	0.7	3.1
Pilgrim	0.4	-0.6	0.2
Cooper	-0.9	1.2	-0.5
Hatch	2.9	-1.2	2.4
Brunswick 2	-0.1	-1.0	-0.4
Duane Arnold	2.3	0.6	2.4
Fitzpatrick	-0.2	1.8	0.4
Lacrosse	1.4	1.6	1.8

Table 14

PWR Overall Performance, FY76, By Categories

Above Average	Average	Below Average
Ginna	Yankee Rowe	San Onofre
Turkey Point 3	Indian Point 2	Connecticut Yankee
Turkey Point 4	Robinson	Palisades
Point Beach 1	Oconee 1	Surry 1
Point Beach 2	Oconee 2	Prairie Island 1
Zion 2	Surry 2	Zion 1
Millstone 2	Fort Calhoun	Arkansas 1
	Oconee 3	Cook 1
	Three Mile Island 1	Trojan
	Kewaunee	
	Prairie Island 2	
	Maine Yankee	
	Rancho Seco 1	
	Calvert Cliffs	

Table 15

BWR Overall Performance, FH76, By Categories

Above Average	Average	Below Average
Dresden 1	Big Rock Point	Humboldt Bay
Dresden 2	Nine Mile Point 1	Oyster Creek
Dresden 3	Quad Cities 1	Millstone
Peach Bottom 3	Quad Cities 2	Monticello
Hatch	Peach Bottom 2	Vermont Yankee
Duane Arnold	Pilgrim	
Lacrosse	Cooper	
	Brunswick 2	
	Fitzpatrick	

Appendix C

Sensitivity Analysis of Weighting Factors

The overall licensee performance me..re, expressed as a Z-score, is obtained by taking a linear combination of the number of violations and the Z-scores for the other performance measures described in Appendix B. In particular, the overall Z-score is a linear combination of Z-scores pertaining to noncompliances, LER's, effluent releases, and personnel exposures. In turn, these Z-scores are themselves weighted sums of other Z-scores. Table 1 displays the fifteen primary Z-values which, along with the violation frequency, go into the overall Z-score. Also shown are one set of weights - the ones developed in part from the analysis of the 1974 data. Note first, in the column headed Wt., that the Z-values pertaining to types of infractions are weighted equally, as are those pertaining to types of deficiencies, LER's, and effluent releases. Violations, and the combined Z's for infractions and deficiencies, are combined, as the column headed Wt2 indicates, in a ratio of -5:2:1. The overall Z is then obtained by weighting the Z's for noncompliances, LER's, effluent releases, and personnel exposures in a ratio of 18:6:2:1 (see the column headed Wt3). The basis of this latter choice is simply that the weight for each Z equals twice the sum of the weights for Z-scores lower in the hierarchy. In all cases except one, the weights are normalized so that the sum of the weights squared equals 1.0. The one exception is that in obtaining the Z-score for noncompliances, the weight given violations is not included in the squaring and summing because of the infrequency of violations and because of a desire that violations, when they occur, have a large effect on the overall Z-score.

Multiplying the weights yields the last column of Table 1. These values are the derivatives of the overall Z-score with respect to the individual Z's. The results mean, for example, that each additional violation decreases Z_{FINAL} by 2.1. Thus, a reactor in the "Average" category otherwise, would be dropped into the "Below Average" category by the incurrence of a violation. A further result is that an increase in the Z-value pertaining to administrative control (AC) infractions would increase Z_{FINAL} by .42; etc. (Recall that a decrease in infractions, deficiencies, LER's, effluents, and percent of personnel exposures over 3 rems each result in increases in their corresponding Z-scores.) Note that the overall Z-score is about equally sensitive to a change in Z-value in any of the three deficiency categories as it is to a change in either of the LER Z-scores. The Z-scores for effluents and personnel exposures have a very small effect on Z_{FINAL}; a change in any of them would result in a change in Z_{FINAL} from one category to the other only in borderline cases. Tables 2 through 4 give results for other choices of weights. The differences these tables reflect come from weighting the four performance variables more equally than 18:6:2:1 and from weighting infractions higher relative to deficiencies than in Table 1. These alternative weights are not radically different from Table 1, but are thought to be "reasonable." The changes in the derivatives resulting from these changes in weighting coefficients are not marked.

One can also work backward from a set of desired derivatives to find the set of weights which would yield those derivatives. For example, consider the following ratios:

Z_1, Z_2, Z_3, Z_4		Z_5, Z_6, Z_7		Z_8, Z_9		$Z_{10}, Z_{11}, Z_{12}, Z_{13}, Z_{14}$		Z_{15}
	:		:		:		:	
10		2.5		5.0		2.0		2.0

This choice weights infractions, LER's, and deficiencies in a 4:2:1 ratio and weights effluents and personnel exposures slightly less than deficiencies. Normalizing these ratios by dividing by the square root of the sum of the squared weights (= 492.75 = 22.2) yields the following derivatives, for the indicated grouping of Z-scores:

.45 : .11 : .23 : .09 : .09

From these values, the weights under the column headed Wt_1, and the constraints on the sum of squares of the weights, the entries under Wt_2 and Wt_3 can be obtained. Setting the derivative of Z_{FINAL} with respect to violations, V, equal to -2.0 leads to the indicated weight for V. The results of this derivation are given in Table 5. Rounding the weights so that the ratios in the Wt_2 and Wt_3 columns are integers yields Table 6, which differs negligibly from Table 5. Table 6 shows that the Z-scores for sub-categories of deficiencies, effluents, and personnel exposure have nearly the same influence on Z_{FINAL}; the Z-scores for the LER sub-categories have about twice the effect of these; and the Z-scores for infraction sub-categories have about twice again this effect. A violation will decrease Z_{FINAL} by about 2.0. This pattern seems reasonable and consistent with the judged relative importance of these variables as indicators of overall performance and so the results presented in this report are based on these weights.

All of the preceding results are in terms of Z-scores - primary and final. To re-express the results in terms of the original data, recall that the primary Z-scores are actually the number of standard deviations that an observation falls below the mean. That is, a change in a Z-value of 1.0 corresponds to a change in the original measurement of one standard deviation. The analysis of noncompliance and LER data from the first six months of 1976 yields estimated standard deviations. By multiplying by 2, the standard deviation for yearly totals can be obtained. (This presumes that performance in successive halves of 1976 are independent. This may not be true, but for the approximate results given here, is not a concern.)

For noncompliances, the estimated standard deviation depends on the number of hours of inspection and the sub-category of noncompliance AC, OP, SG, and RE. However, rough averages can be obtained. These are a standard deviation of 2 infractions for each of the infraction sub-categories and also for each of the deficiency categories. Thus, for the weights of Table 6, each additional infraction per year in any of the four sub-categories decreases Z_{FINAL} by .45/2 = .23; each additional deficiency would decrease Z_{FINAL} by .05. Note that the relative influence of violations, infractions, and deficiencies on Z_{FINAL} fall in a ratio of about 50:5:1 which is the weighting already used by I&E, Manual Chapter 0800. This result just happened to fall out; Tables 5 and 6 were not picked with that objective in mind.

For LER's, the estimated standard deviation depends on reactor age and the sub-category personnel or procedural error of the LER. A rough average is a standard deviation of 1.7. Thus, each additional LER per year decreases ZFINAL by about .22/1.7 = .13.

Effluent releases were converted to ranks and Z-scores determined from the ranks. The standard deviation for a rank is $(n^2 - 1)/12$, where n is the number of reactors in the ranking. For n = 30, this standard deviation equals 8.7. Thus, if a reactor increases its rank by 1, (recall that releases are ranked from low to high) then Z_{FINAL} would decrease by .09/8.7 = .01. For n = 21, the corresponding decrease in Z_{FINAL} is .09/6.1 = .016.

From 1974 data, the percentage of personnel who were exposed to 3 rems or more had a standard deviation of 2.4 percentage points. Thus, an increase of one percentage point in the percentage of employees with this exposure results in a decrease in Z_{FINAL} of .10/2.4 = .04.

Table 1

Weights on Primary Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Type	Variable	Wt_1	Wt_2	Wt_3	$Wt_1 \times Wt_2 \times Wt_3$
Violations		V	1	$-5/\sqrt{5}$		-2.1
Infractions						
	Admin. Control	Z_1	1/2			.42
	Operations	Z_2	1/2	$2/\sqrt{5}$	$18/\sqrt{365}$	.42
	Safeguards	Z_3	1/2			.42
	Remaining	Z_4	1/2			.42
Deficiencies						
	Admin. Control	Z_5	$1/\sqrt{3}$			.24
	Operations	Z_6	$1/\sqrt{3}$	$1/\sqrt{5}$		.24
	Remaining	Z_7	$1/\sqrt{3}$			.24
LER's						
	Personnel	Z_8	$1/\sqrt{2}$			.22
	Procedural	Z_9	$1/\sqrt{2}$		$6/\sqrt{365}$	.22
Effluent Release						
	Noble Gas	Z_{10}	$1/\sqrt{5}$			.05
	Hal. & Part.	Z_{11}	$1/\sqrt{5}$			.05
	Tritium	Z_{12}	$1/\sqrt{5}$		$2/\sqrt{365}$	.05
	Mixed Fission	Z_{13}	$1/\sqrt{5}$			.05
	Solid Waste	Z_{14}	$1/\sqrt{5}$			.05
Personnel Exposures						
	P3	Z_{15}	1		$1/\sqrt{365}$	.05

Table 2

Weights on Primary Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Type	Variable	Wt_1	Wt_2	Wt_3	$Wt_1 \times Wt_2 \times Wt_3$
Violations		V	1	$-10/\sqrt{26}$		-1.67
Infractions						
	Admin. Control	Z_1	1/2			.42
	Operations	Z_2	1/2			.42
	Safeguards	Z_3	1/2	$5/\sqrt{25}$		.42
	Remaining	Z_4	1/2		$4/\sqrt{22}$	.42
Deficiencies						
	Admin. Control	Z_5	$1/\sqrt{3}$			.10
	Operations	Z_6	$1/\sqrt{3}$	$1/\sqrt{26}$		.10
	Remaining	Z_7	$1/\sqrt{3}$			.10
LER's						
	Personnel	Z_8	$1/\sqrt{2}$		$2/\sqrt{22}$	.30
	Procedural	Z_9	$1/\sqrt{2}$			.30
Effluent Release						
	Noble Gas	Z_{10}	$1/\sqrt{5}$			.10
	Hal. & Part.	Z_{11}	$1/\sqrt{5}$			.10
	Tritium	Z_{12}	$1/\sqrt{5}$		$1/\sqrt{22}$	.10
	Mixed Fission	Z_{13}	$1/\sqrt{5}$			.10
	Solid Waste	Z_{14}	$1/\sqrt{5}$			.10
Personnel Exposures						
	P3	Z_{15}	1		$1/\sqrt{22}$	.21

Table 3

Weights on Primary Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Type	Variable	Wt_1	Wt_2	Wt_3	$Wt_1 \times Wt_2 \times Wt_3$
Violations		V	1	$-5/\sqrt{5}$	$4/\sqrt{22}$	-1.91
Infractions						
	Admin. Control	Z_1	1/2	$2/\sqrt{5}$		.38
	Operations	Z_2	1/2			.38
	Safeguards	Z_3	1/2			.38
	Remaining	Z_4	1/2			.38
Deficiencies						
	Admin. Control	Z_5	$1/\sqrt{3}$	$1/\sqrt{5}$		.22
	Operations	Z_6	$1/\sqrt{3}$			.22
	Remaining	Z_7	$1/\sqrt{3}$			.22
LER's						
	Personnel	Z_8	$1/\sqrt{2}$		$2/\sqrt{22}$	.30
	Procedural	Z_9	$1/\sqrt{2}$			.30
Effluent Release						
	Noble Gas	Z_{10}	$1/\sqrt{5}$		$1/\sqrt{22}$	.10
	Hal. & Part.	Z_{11}	$1/\sqrt{5}$			.10
	Tritium	Z_{12}	$1/\sqrt{5}$			.10
	Mixed Fission	Z_{13}	$1/\sqrt{5}$			.10
	Solid Waste	Z_{14}	$1/\sqrt{5}$			.10
Personnel Exposures						
	P3	Z_{15}	1		$1/\sqrt{22}$	.21

Table 4

Weights on Primary Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Type	Variable	Wt_1	Wt_2	Wt_3	$Wt_1 \times Wt_2 \times Wt_3$
Violations		V	1	$-10/\sqrt{26}$		-1.85
Infractions						
	Admin. Control	Z_1	1/2			.46
	Operations	Z_2	1/2	$5/\sqrt{26}$		.46
	Safeguards	Z_3	1/2			.46
	Remaining	Z_4	1/2		$18/\sqrt{365}$	.46
Deficiencies						
	Admin. Control	Z_5	$1/\sqrt{3}$			.11
	Operations	Z_6	$1/\sqrt{3}$	$1/\sqrt{26}$		.11
	Remaining	Z_7	$1/\sqrt{3}$			.11
LER's						
	Personnel	Z_8	$1/\sqrt{2}$		$6/\sqrt{365}$	.22
	Procedural	Z_9	$1/\sqrt{2}$			.22
Effluent Release						
	Noble Gas	Z_{10}	$1/\sqrt{5}$			.05
	Hal. & Part.	Z_{11}	$1/\sqrt{5}$			.05
	Tritium	Z_{12}	$1/\sqrt{5}$		$2/\sqrt{365}$	.05
	Mixed Fission	Z_{13}	$1/\sqrt{5}$			.05
	Solid Waste	Z_{14}	$1/\sqrt{5}$			.05
Personnel Exposures						
	P3	Z_{15}	1		$1/\sqrt{365}$	.05

Table 5

Weights on Primary Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Type	Variable	Wt_1	Wt_2	Wt_3	$Wt_1 \times Wt_2 \times Wt_3$
Violations		V	1	-2.17		-2.0
Infractions						
	Admin. Control	Z_1	1/2			.45
	Operations	Z_2	1/2	.98		.45
	Safeguards	Z_3	1/2			.45
	Remaining	Z_4	1/2		.92	.45
Deficiencies						
	Admin. Control	Z_5	$1/\sqrt{3}$			.11
	Operations	Z_6	$1/\sqrt{3}$	.21		.11
	Remaining	Z_7	$1/\sqrt{3}$			.11
LER's						
	Personnel	Z_8	$1/\sqrt{2}$		.32	.23
	Procedural	Z_9	$1/\sqrt{2}$			.23
Effluent Release						
	Noble Gas	Z_{10}	$1/\sqrt{5}$			.09
	Hal. & Part.	Z_{11}	$1/\sqrt{5}$			.09
	Tritium	Z_{12}	$1/\sqrt{5}$		.20	.09
	Mixed Fission	Z_{13}	$1/\sqrt{5}$			.09
	Solid Waste	Z_{14}	$1/\sqrt{5}$			.09
Personnel Exposures						
	P3	Z_{15}	1		.09	.09

Table 5

Weights on Primary Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Type	Variable	Wt_1	Wt_2	Wt_3	$Wt_1 \times Wt_2 \times Wt_3$
Violations		V	1	$-10/\sqrt{26}$		-1.96
Infractions						
	Admin. Control	Z_1	1/2			.45
	Operations	Z_2	1/2	$5/\sqrt{26}$		.45
	Safeguards	Z_3	1/2			.45
	Remaining	Z_4	1/2		$9/\sqrt{95}$	.45
Deficiencies						
	Admin. Control	Z_5	$1/\sqrt{3}$			.10
	Operations	Z_6	$1/\sqrt{3}$	$1/\sqrt{25}$		.10
	Remaining	Z_7	$1/\sqrt{3}$			.10
LER's						
	Personnel	Z_8	$1/\sqrt{2}$		$3/\sqrt{95}$	.22
	Procedural	Z_9	$1/\sqrt{2}$			.22
Effluent Release						
	Noble Gas	Z_{10}	$1/\sqrt{5}$			.09
	Hal. & Part.	Z_{11}	$1/\sqrt{5}$			.09
	Tritium	Z_{12}	$1/\sqrt{5}$		$2/\sqrt{95}$	.09
	Mixed Fission	Z_{13}	$1/\sqrt{5}$			.09
	Solid Waste	Z_{14}	$1/\sqrt{5}$			.09
Personnel Exposures						
	P3	Z_{15}	1		$1/\sqrt{95}$	.10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

September 26, 1977

MEMORANDUM FOR: Ernst Volgenau, Director
Office of Inspection and Enforcement, HQ

FROM: E. Morris Howard, Director, Region IV
Office of Inspection and Enforcement

SUBJECT: DRAFT REPORT - LICENSEE INSPECTION AND ENFORCEMENT INDICATORS

The final Draft Report of Licensee Inspection and Enforcement Indicators which is intended to fulfill the assignment to establish and validate techniques for Licensee Inspection and Enforcement Indicators is submitted for your consideration. The Draft Report is a detailed statistical analysis which has been examined by an independent contractor (ORNL) and found to be mathematically and statistically valid. Suggestions made by ORNL are encompassed in the revision of this detailed statistical analysis.

I consider the detailed statistical as both desirable and necessary supportive information to any analysis of performance indicators; however, it is felt that a simplified technique, using the identical data base, but requiring considerably less analysis was in order. In the development of the simplified technique, items of noncompliance were assigned a value, summed, and the Z score calculated. Figure No. 1 is the flow diagram for these calculations. The Z scores, which are the number of standard deviations that an observation differs from the mean of its group, are shown on Figures No. 2 and No. 3. The comparisons between the simplified and detailed analysis are shown on Tables No. 1 and No. 2.

An attempt was made to separate functional areas in the Draft Report with what I consider less than roaring success due to the lack of data. It appears that a clearer relationship between total noncompliance and the functional areas is more clearly discernable by recalculating a new total Z score after subtracting the contribution of a given functional area, and then comparing the two total Z scores. Figure 4 shows the contribution of Safeguards to the total score of the several pressurized water reactor sites.

This simplified concept uses the same basic techniques described in the Draft Report except for pre-weighting and it would be redundant to redescribe them here.

It is recommended that this simplified technique be used and that an annual detailed statistical analysis be performed to evaluate possible emerging and presently elusive relationships.

E. Morris Howard

E. Morris Howard
Director

Enclosures:
As stated

cc: J. G. Davis
H. D. Thornburg

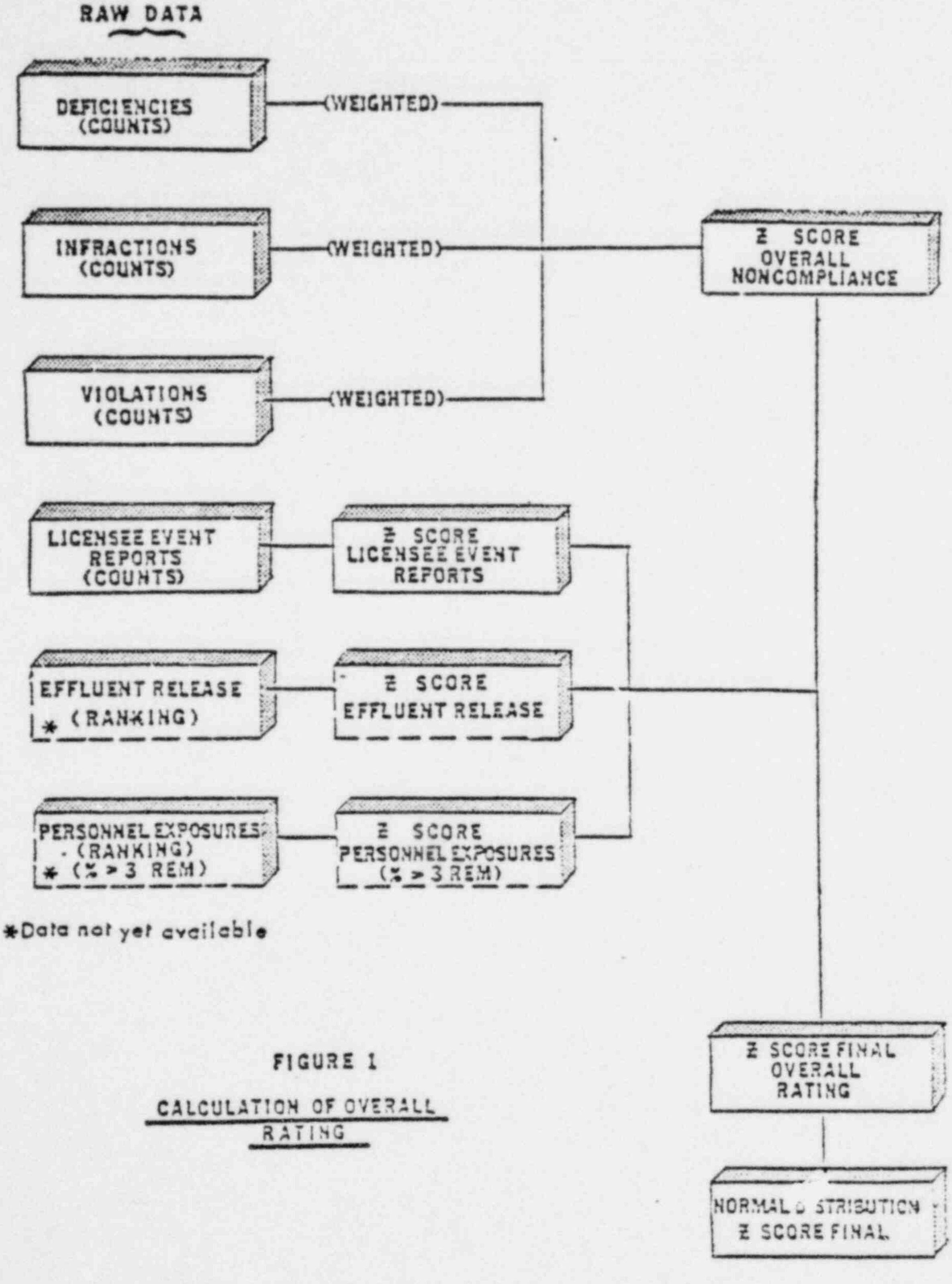
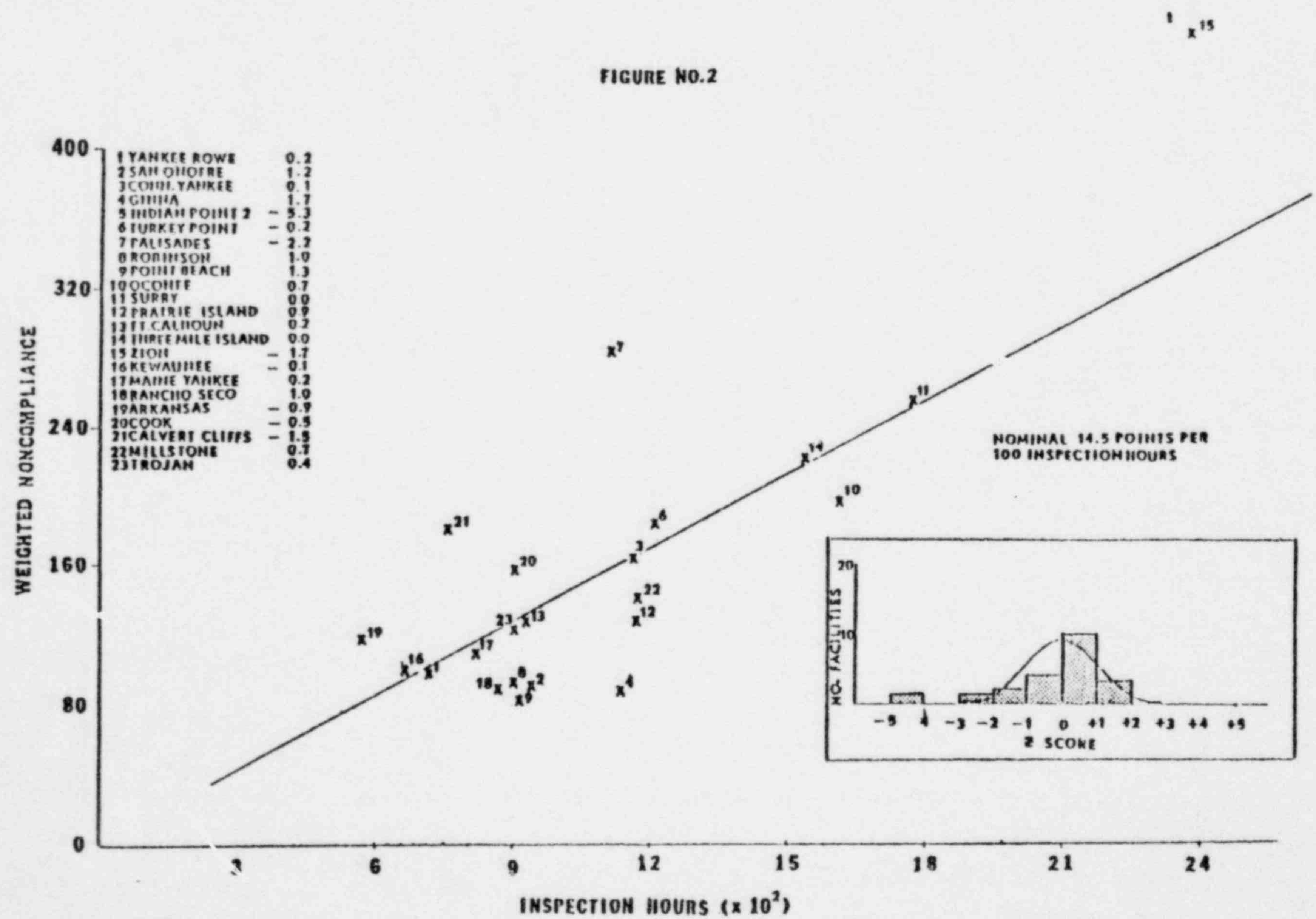


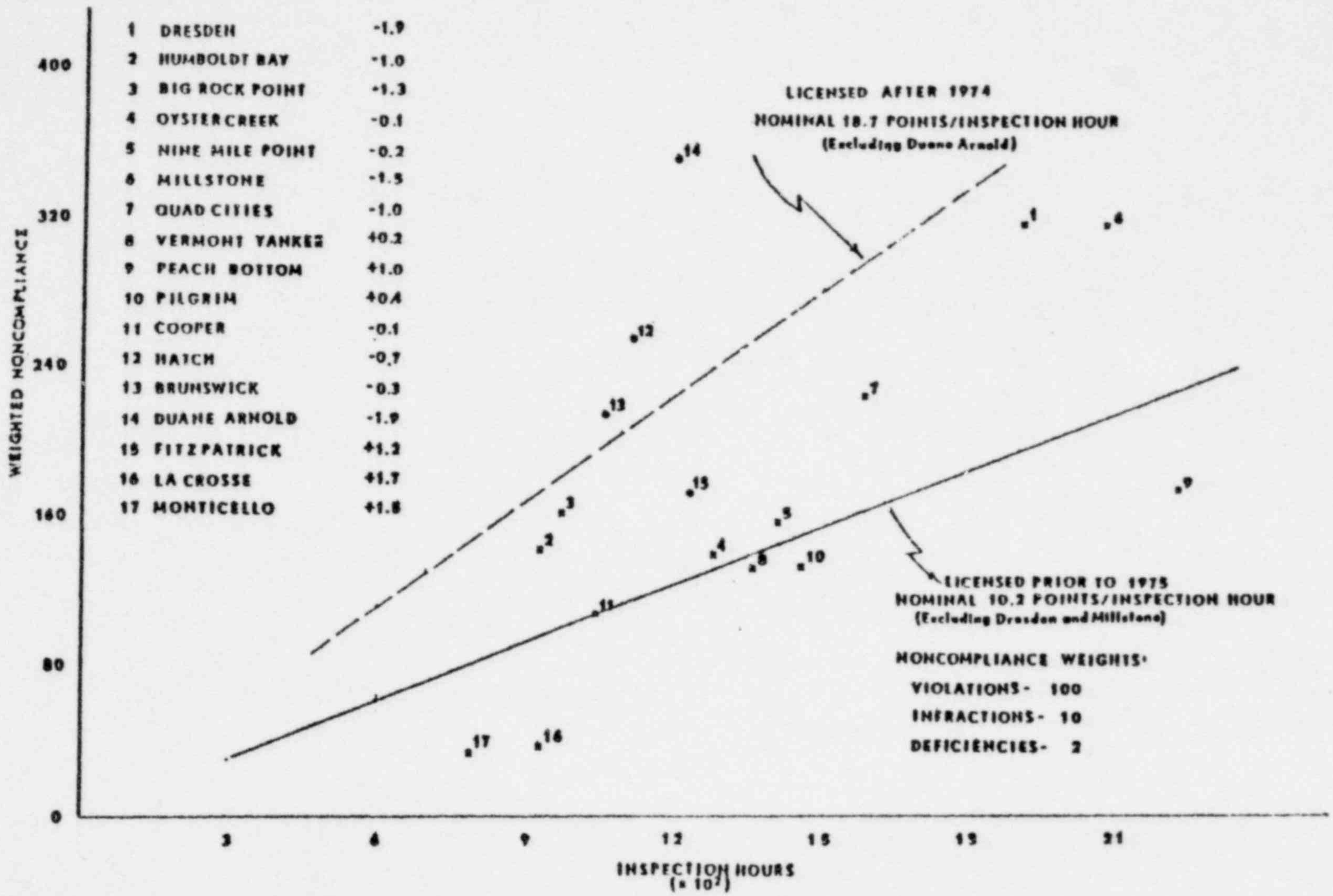
FIGURE 1

CALCULATION OF OVERALL RATING

FIGURE NO.2



PRESSURIZED WATER REACTOR-CALENDER YEAR-1976



BOILING WATER REACTORS CALENDER YEAR 1976

FIGURE NO.3

Table 1

COMPARATIVE

Z SCORES

PRESSURIZED WATER REACTORS

CALENDAR YEAR - 1976

FACILITY	Z SCORE NONCOMPLIANCE SIMPLIFIED	Z SCORE NONCOMPLIANCE DETAILED	Z SCORE TOTAL SIMPLIFIED	Z SCORE TOTAL DETAILED	CATEGORY SIMPLIFIED	CATEGORY DETAILED
Yankee Rowe	0.2	0.3	0.5	0.6	B	B
San Onofre	1.2	1.3	2.4	2.4	A	A
Conn. Yankee	0.1	0.1	0.2	0.2	B	B
Ginna	1.7	2.1	1.8	2.1	A	A
Indian Point #2	-5.3	-7.4	-5.0	-6.9	C	C
Turkey Point	-0.2	-0.5	0.3	0.1	B	B
Palisades	-2.2	-2.7	-2.4	-2.8	C	C
H. B. Robinson	1.0	1.0	0.6	0.6	B	B
Point Beach	1.3	1.4	1.4	1.4	A	A
Oconee	0.7	0.5	-0.1	-0.3	B	B
Surry	0.0	-0.5	-0.6	-1.1	B	C
Prairie Island	0.9	0.9	0.3	0.3	B	B
Ft. Calhoun	0.2	0.0	0.3	0.1	B	B
Three Mile Island	0.0	-0.2	-0.2	-0.4	B	B
Zion	-1.7	-3.3	-1.8	-3.4	C	C
Kewaunee	-0.1	-0.2	-0.4	-0.5	B	B
Maine Yankee	0.2	0.5	0.5	0.8	B	B

Table 1 (cont'd)

COMPARATIV

Z SCORES

PRESSURIZED WATER REACTORS

CALENDAR YEAR - 1976

FACILITY	Z SCORE NONCOMPLIANCE SIMPLIFIED	Z SCORE NONCOMPLIANCE DETAILED	Z SCORE TOTAL SIMPLIFIED	Z SCORE TOTAL DETAILED	CATEGORY SIMPLIFIED	CATEGORY DETAILED
Rancho Seco	1.0	1.1	1.6	1.7	A	A
Arkansas	-0.9	-1.1	-0.7	-0.8	B	B
Cook	-0.5	-0.9	-0.9	-1.2	B	C
Calvert Cliffs	-1.5	-1.6	-0.8	-0.8	B	B
Millstone 2	0.7	0.8	1.0	1.1	B	A
Trojan	0.4					

Table 2

COMPARATIVE

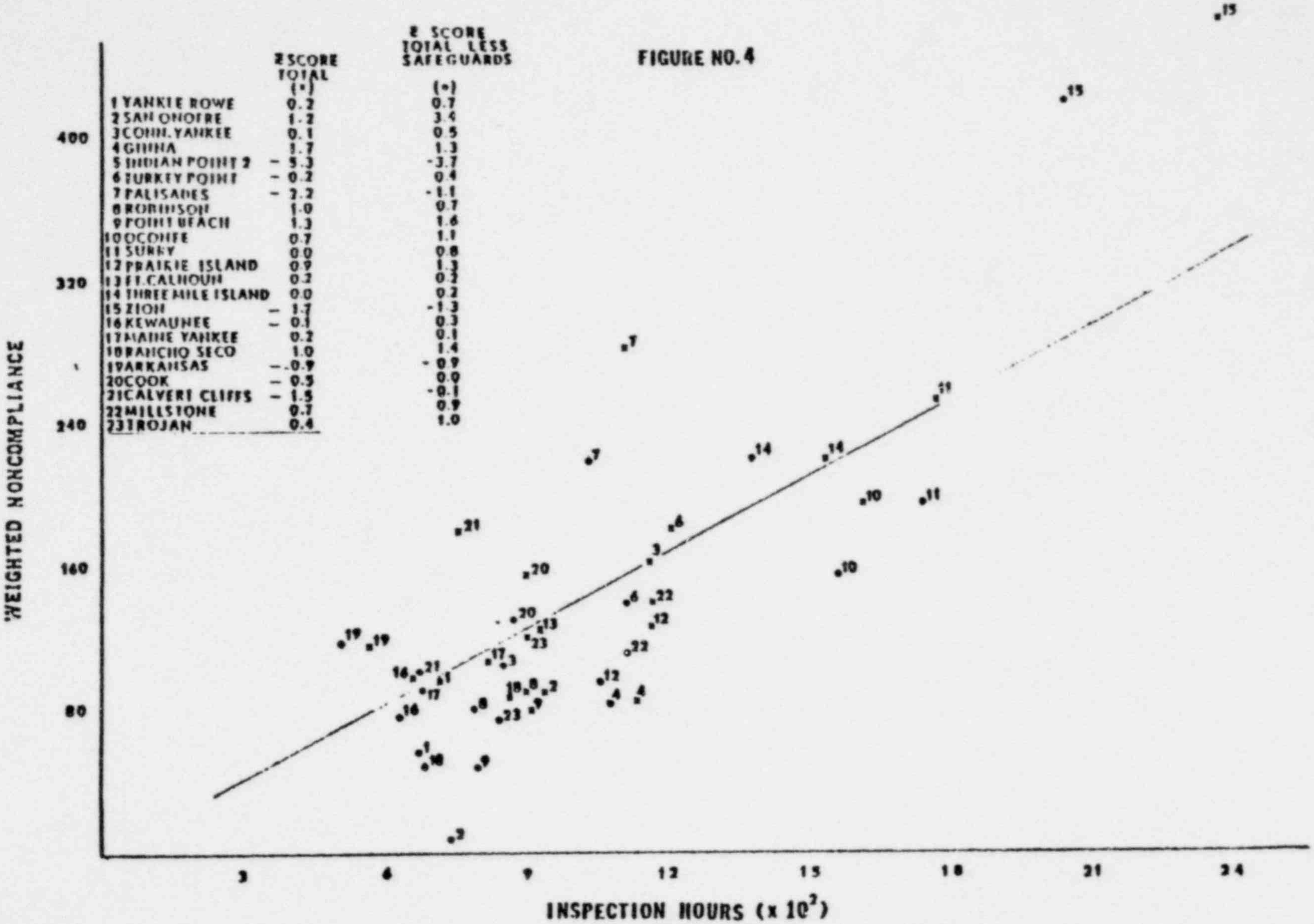
Z SCORES

BOILING WATER REACTORS

CALENDAR YEAR - 1976

FACILITY	Z SCORE NONCOMPLIANCE SIMPLIFIED	Z SCORE NONCOMPLIANCE DETAILED	Z SCORE TOTAL SIMPLIFIED	Z SCORE TOTAL DETAILED	CATEGORY SIMPLIFIED	CATEGORY DETAILED
Dresden	-1.9	-2.6	-3.0	-3.6	C	C
Humboldt Bay	-1.0	-1.4	-0.7	-1.1	B	C
Big Rock Point	-1.3	-1.7	-1.6	-2.0	C	C
Oyster Creek	-0.1	-0.5	0.2	0.8	B	B
Nine Mile Point	-0.2	0.0	-0.2	0.0	B	B
Millstone	-1.5	-2.1	-1.6	-2.2	C	C
Quad Cities	-1.0	-1.2	-1.2	-1.4	C	C
Monticello	1.5	2.3	1.9	2.6	A	A
Vermont Yankee	0.2	0.6	0.2	0.6	B	B
Peach Bottom	1.0	1.4	1.1	1.5	A	A
Pilgrim	0.4	1.2	0.5	1.3	B	A
Cooper	-0.1	-0.4	1.1	0.8	A	B
Hatch	-0.7	-1.1	-1.1	-1.5	C	C
Brunswick	-0.3	0.6	-0.6	-0.9	B	B
Duane Arnold	-1.9	-2.8	-1.8	-2.6	C	C
Fitzpatrick	1.2	1.0	1.7	1.5	A	A
La Crosse	1.7	2.4	2.0	2.7	A	A

FIGURE NO. 4



PRESSURIZED WATER REACTOR-CALENDER YEAR-1976.

Draft Report

A STATISTICAL EVALUATION OF THE NUCLEAR SAFETY-RELATED PERFORMANCE OF NRC OPERATING REACTOR LICENSEES DURING 1976

(Licensee Inspection and Enforcement Indicators)

August 1977

E. Morris Howard, Project Director
Stephen K. Conver
Robert G. Easterling
Walter S. Schwink

Chapter I

INTRODUCTION

A. Background

"Licensee Inspection and Enforcement Indicators" is the term used to describe the efforts of the NRC Office of Inspection and Enforcement to evaluate the nuclear safety-related management performance of its licensees. This draft report addresses the management performance of NRC's operating reactor licensees during the calendar year 1976. This report is the culmination of an effort initiated in April 1976 to develop and test a methodology for evaluating licensee management performance, as a technique for improving manpower utilization in the Office of Inspection and Enforcement, and establishing thresholds for initiation of corrective enforcement action.

This evaluation is made on the basis of two factors that are considered to reflect the degree of success licensee management has achieved in carrying out its responsibilities for safe operation and protection of the public. These factors indicate each licensee's compliance with NRC rules and the numbers of occurrences at a licensee's facility with potential safety implications. These factors are analyzed using standard statistical techniques that permit comparisons of licensee performance both in specific areas and from an overall perspective.

There are two additional factors which are considered to be pertinent to the overall licensee management performance which were not considered in this evaluation. While pertinent when considered solely on their own merit and exclusive of their contribution to noncompliance and LER's, these two factors, radiation exposures at the licensee's facility and the extent to which the licensee limits effluents, are not an integral part of this evaluation and in no way affect the results. As these data become available they will be analyzed and incorporated in the overall evaluation, adding an additional increment of knowledge of the licensee's management performance.

The choice of measures of licensee management performance reflects the concerns that licensees be measured objectively, using measurable and collectable statistics that apply uniformly to all operating reactor licensees. It is also important that the measures of performance include only items that are controllable by the licensee.

The use of statistical analysis to develop performance measures or indices has many precedents. Economic indicators, such as the Dow Jones averages, are commonly used to give the public an appreciation of the overall state of the economy. The "Quality of Life" index published by the Midwest Research Institute is a similar effort that ranks American cities on the basis of weighted sums of a number of indicators. Overall product rankings of Consumer's Reports and NFL quarterback rankings are other examples of the process of ranking individuals, groups, or objects according to some function of selected attributes or statistics.

In all these cases, there is some "latent variable" that is of interest, but which cannot be measured directly - economic health, quality of life, product quality, or athletic ability. By carefully choosing and analyzing data, one hopes to develop useful indicators of the latent variables. The various indicators are not equivalent to the latent variables; however, as the measured indicators are improved (numbers of libraries, interceptions, etc.), often the latent variables (quality of life, athletic performance, etc.) are also improved.

This effort to determine NRC licensee inspection and enforcement indicators has similar objectives. While we recognize that "safety" cannot be measured directly, we hope to improve it by evaluating the success of licensee management in controlling safety-related indicators of performance.

B. NRC and Licensee Responsibilities

Direct responsibility for conducting nuclear operations in a manner that protects public health and safety lies with the licensee. One of the ways that the licensee satisfies this obligation is by complying with NRC rules and regulations.

NRC shares this responsibility for protecting the public with the licensees. NRC responsibilities, as described in law, are to generate rules to insure safe operations and to verify that those rules are being followed. The NRC Office of Inspection and Enforcement (IE) is the arm of NRC charged with conducting this verification. IE uses its inspection force to insure compliance with the rules. Another important function of IE is to identify existing rules that need improvement or new rules that are needed.

Compliance with NRC rules is a function of licensee management. In general, a low level of noncompliance indicates that licensee management is doing a good job of carrying out its responsibilities to NRC and to the public. On the other hand, a high level of noncompliance may indicate a low level of management interest or participation in this regard. The performance of licensee management is similarly evident in trends for LERs. The present effort proposes a method to evaluate licensee management performance on a systematic and objective basis so that negative trends can quickly be identified and so that "management breakdown" can be prevented.

C. Why Licensee Inspection and Enforcement Indicators?

The NRC practice of focusing inspection attention on "poor performers" is well established and generally accepted. This evaluation of licensee management performance is designed to permit this allocation of IE inspection resources to be conducted more systematically than in the past. This effort should also allow a more objective allocation, because all licensees will be measured against a single set of performance standards. And because each licensee will be compared against the total population of similar facilities, the identification of poor performers and subsequent allocation of IE resourdes to these facilities (and to specific areas at a given facility) should be more uniform across the NRC Regional Offices.

The methodology, which is discussed in a subsequent section, must necessarily result in a ranking of licensee management performance. A numerical ranking, however, is not the intent of this study nor is there any particular merit in the exact numerical rating of a licensee. Historical data, while lacking the precision to establish an exact numerical difference between facilities, can adequately portray those facilities and areas of concern within a facility which require additional Inspection and Enforcement attention.

The primary concern of the NRC is the health and safety of the public with each undertaking oriented toward this concern. Within this context, this evaluation can be related to safety of reactor facilities. The relationship to safety, however, is clearly one of improving the efficiency of IE manpower resource allocation to effect improvement in licensee's performance where weaknesses are detectable.

D. Structure of the Report

The main body of this report is presented at a level of detail appropriate for IE and NRC management. Brief chapters summarize the methodology (Chapter II) and results (Chapter III).

Additional technical detail is provided in two appendices to the report. These consist of results (Appendix A), as well as the documentation of an analysis of the sensitivity of overall rating results to the choice of weighting factors (Appendix B).

Chapter II

METHODOLOGY

A. Introduction

This chapter describes the data elements that are considered in the evaluation, the analysis tools that are used, and the specific approach taken to analyze each distinct type of data. A detailed description of the methodology is provided in Appendix A.

B. Data Elements

This analysis is based upon two measures of performance - noncompliance history of licensees and selected Licensee Event Reports (LERs). Two other measures - effluent releases and personnel exposures, which were mentioned briefly in the introduction, will be included in future evaluations as data become available. Each of these four measures are discussed below.

Noncompliance items result from NRC regulation and inspection of licensee facilities. Noncompliance data consist of "counts" of NRC findings in a given time period. These noncompliance items are classified into three categories: in decreasing order of severity, these are violations, infractions, and deficiencies. The noncompliance data thus consist of number of violations, infractions, and deficiencies for each licensee considered. The data are further broken down to describe the licensee function or operation that is the source of each noncompliance. Six areas are used: (1) Administrative Control, (2) Operations, (3) Emergency Planning, (4) Radiological Protection and Control, (5) Safeguards, and (6) Quality Assurance. Individual counts of noncompliance data are presented for each of the three severity levels. Also, an overall measure on noncompliance is developed for each licensee that is a weighted sum of the numbers of infractions and deficiencies.

Licensee Event Report data are also stated in terms of "counts" for each licensee. LERs are reports submitted by reactor licensees when certain safety-related events occur at a facility. It is not appropriate to measure licensees by a count of all LERs submitted, because not all reportable events are controllable by the licensee, applicable to the total population of reactor licensees, or serious enough to warrant NRC enforcement action if not reported by the licensee. For this reason, only those LERs characterized as "personnel errors" and "procedural errors" are considered.

Effluent Release data will be expressed in terms of licensee rankings. This is done because the actual effluent measurements may vary over several orders of magnitude for selected licensees. Effluent data are also categorized into five types: (1) noble gases, (2) halogens and particulates, (3) tritium, (4) mixed fission and activation products, and (5) solid waste. Overall rankings for effluent release are obtained by summing the licensee rankings in each of the five categories.

Personnel exposure data for each operating reactor are reported annually in the form of a table listing the numbers of persons exposed in various ranges of exposure (in rems). This evaluation of licensee management's success in limiting exposures is measured in terms of the percent of all people receiving measurable doses of radiation that received three rems or more in one year (rather than on a "ranking" or "counting" basis).

C. Analysis Tools

Three types of statistical techniques are used in this analysis - adjustment, normalization, and weighting procedures.

Adjustment of data is necessary because direct comparisions of licensee management performance are not always meaningful. For example, if one plant has twice as much inspection effort as another, a direct comparison of the noncompliance findings resulting from those inspections may not be meaningful, and it is necessary to make appropriate adjustments. The purpose of adjustment is to compensate for those measurement factors that are not under the licensee's control. This technique is used sparingly in the analysis to preclude elimination of actual licensee differences.

The specific techniques used to make these adjustments include linear regression, goodness of fit tests, and graphical techniques. These methods are used to identify and compensate for factors beyond the control of the licensees that can account for differences in their performance. For example, using the earlier example, Figure 1 depicts the number of infractions for each operating PWR reactor as a function of the hours of inspection devoted to each during 1976. This chart shows that infractions increase as inspection effort increases. Since this relationship also has an intuitive explanation (the more you look, the more you find), an appropriate adjustment is made.

The diagonal straight line that bisects the data in Figure 1 accounts for a significant portion of the differences in the "performance" of the various licensees. It shows that every 100 hours of inspection, on the average result in about 1.27 infractions. Thus, those licensees below the line are considered in this analysis to have better performance than those above the line, and for this measure, performance is essentially measured on a "rate" basis (infractions per hour of inspection).

The line representing the average rate of infractions can be obtained by one of several analytical methods - linear regression or graphical techniques. Goodness of fit tests can be used to assess whether the residual variations in licensee performance (the deviations of the individual points from the diagonal line in Figure 1) are random (what could be expected by chance). If these tests, such as the "chi square test," show that the residual variation (after adjustment) is random, this means that licensees are "homogeneous" with respect to the variable being measured (infractions) and that there is no need to look for further adjustments. A lack of randomness indicates either actual differences between licensees or the need to look for further adjustments. The approach in this analysis has been to make adjustments only when there is a logical cause and effect explanation for the relationships identified.

The purpose and effect of adjustment can be seen by comparing Figures 2 and 3. Figure 2 is a histogram of the PWR infractions plotted in Figure 1. If infraction frequencies were homogeneous, that is, if the expected number of infractions were the same for all facilities and only "chance" variations affected the infraction frequencies actually observed, then the histogram should agree reasonably well with the Poisson distribution, which is also shown in Figure 1. (Note that the Poisson distribution is only defined for integer numbers of infractions. The distribution is shown as a continuous curve only for visual purposes.) The lack of agreement between the observed histogram and the expected Poisson distribution is apparent even without performing a statistical goodness of fit test. A major source of the discrepancy is inspection hours, because after the infraction frequencies are adjusted for inspection hours, then normalized (as described below), the histogram of Figure 3 results. If after adjustment and normalization, the infraction frequencies are homogeneous, then their histogram would be expected to agree with the standard normal distribution and Figure 3 shows that agreement is much closer than before (Figure 2).

As discussed, it is the aim of the analysis to reduce the residual variation in a performance measure, after adjustment, to the sort of variation variation one would expect by chance alone. This means that for that measure, apparent differences cannot be hailed as statistically significant, although these differences can still guide IE activities. The feature of the licensee performance evaluation which will permit differences to be identified statistically is replication. If some licensees score consistantly high over several performance measures and others consistently low, then this is evidence of true differences among licensees. In order to examine this possibility, the data were analyzed for the first and second halves of 1976 separately. Comparing and combining performance measures for these two periods is then done as an aid to identifying licensee differences. It should be noted that a lack of statistical significance of licensee differences does not mean no differences exist, only that they are too small to be detected by available data.

The techniques for adjustment of data explained in the preceding paragraphs enable comparisons of licensee performance for single measures of that performance. Another objective of this analysis is to combine the various performance measures so that an overall measure of performance in terms of inspection of licensee activities can be obtained.

Normalization is an analytical technique that makes these overall comparisons possible. Its purpose is to transform each performance measure into a dimensionless quantity so that a sum of difference measure is possible. The transformation used is a "Z-score," which can be regarded as the number of standard deviations that an observation differs from the mean of its group. The particular Z-score used, in the case of noncompliances and LERs, however, is not mathematically defined in terms of standard deviations. This is because the model used for such counting data is the Poisson distribution which is not symmetric about its nominal, or expected, value. In order to reduce the asymmetry, Z-scores will be calculated by

$$Z = 2(\sqrt{NCM} - \sqrt{X})$$

where NCM is the estimated expected number of counts and X is the observed count. The factor of 2 is so that the variance of Z will be approximately that of the standard normal distribution. Note also the Z is defined so that positive scores indicate better performance, and vice-versa.

The appropriate frame of reference for Z-scores is the standard normal distribution shown in Figure 3. This distribution has a mean of zero and standard deviation of one. When converted to Z-scores, licensee performance measures can be compared to this distribution. Under the standard normal distribution about two-thirds of the Z-scores would be expected to fall between plus and minus one, with one-sixth on either side of this interval. While these fractions are rarely achieved exactly, the Z-scores are still comparable. And, because they are dimensionless and comparable, Z-scores can be summed for various performance measures.

Weighting is the process by which Z-scores for various performance measures are summed in a manner that reflects the relative importance (weight) associated with each of the factors contributing to the overall score. The process of transforming raw data for noncompliances, LERs, effluents, and exposures to overall licensee management ratings is depicted in Figure 4. While the overall rating is of interest, the raw data and all intermediate results leading to that overall rating are significant results in their own right.

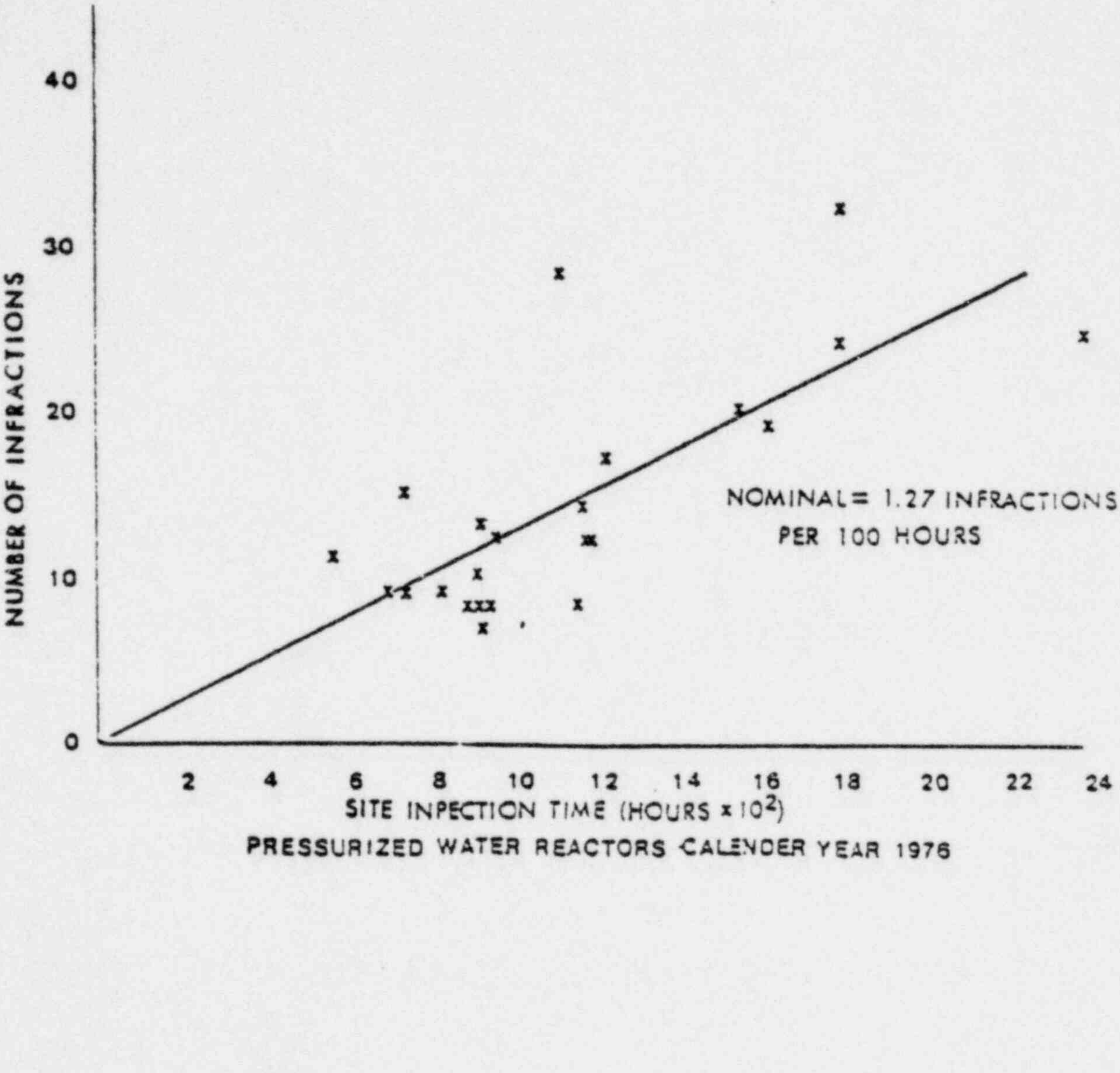
Because the process of weighting is inherently judgmental, a "sensitivity analysis" was conducted to assess the influence of a variety of alternative weightings on the overall ratings. The results of this analysis are presented in Appendix B.

In summary, the analysis process leading to overall evaluations of the performance of licensee management involves the following steps:

- Adjust data to remove factors beyond the control of licensee management.
- Normalize licensee performance in each measure to a dimensionless Z-score.
- Obtain overall ratings using weighted sums of the Z-scores.

FIGURE NO. 1

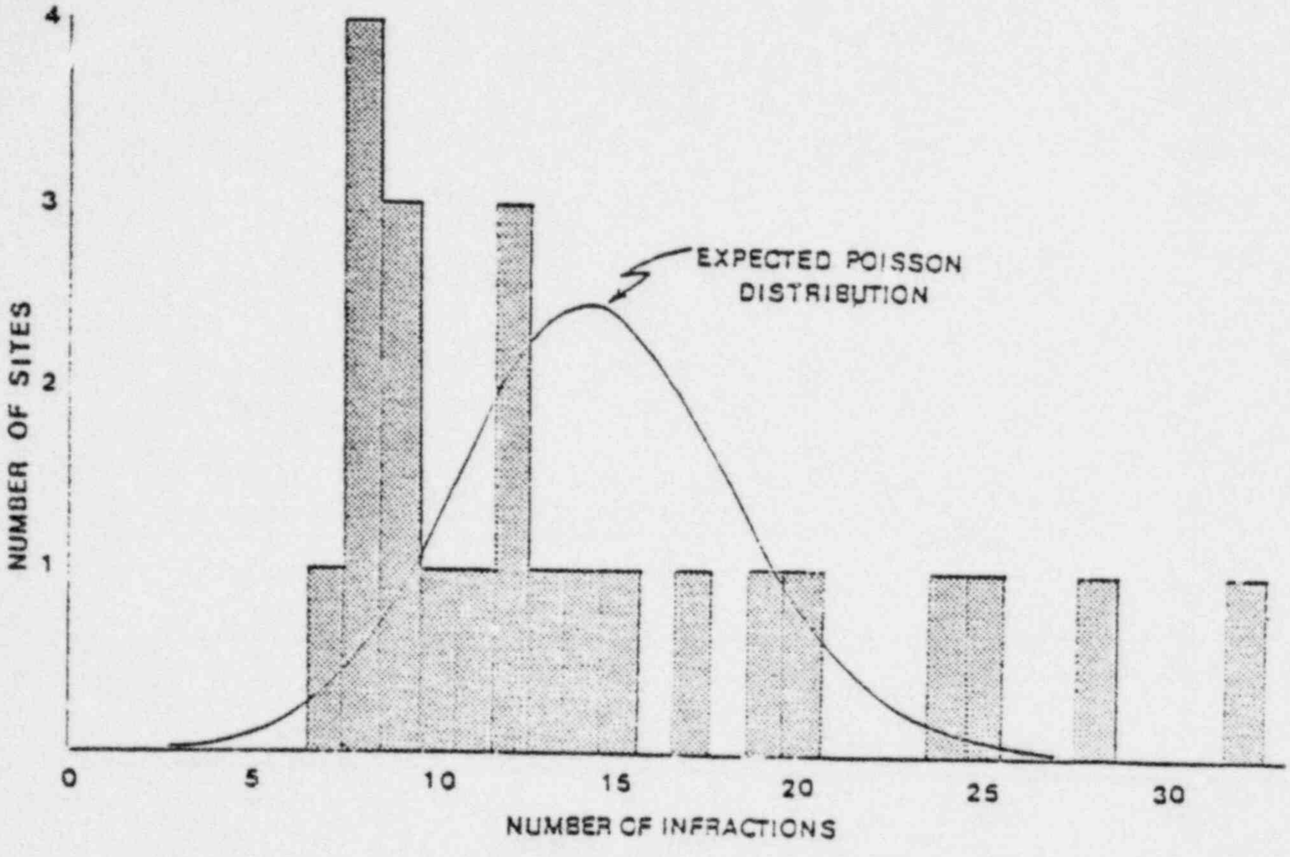
ADJUSTMENT OF PWR INFRACTION DATA FOR INSPECTION TIME



PRESSURIZED WATER REACTORS CALENDER YEAR 1976

FIGURE NO. 2

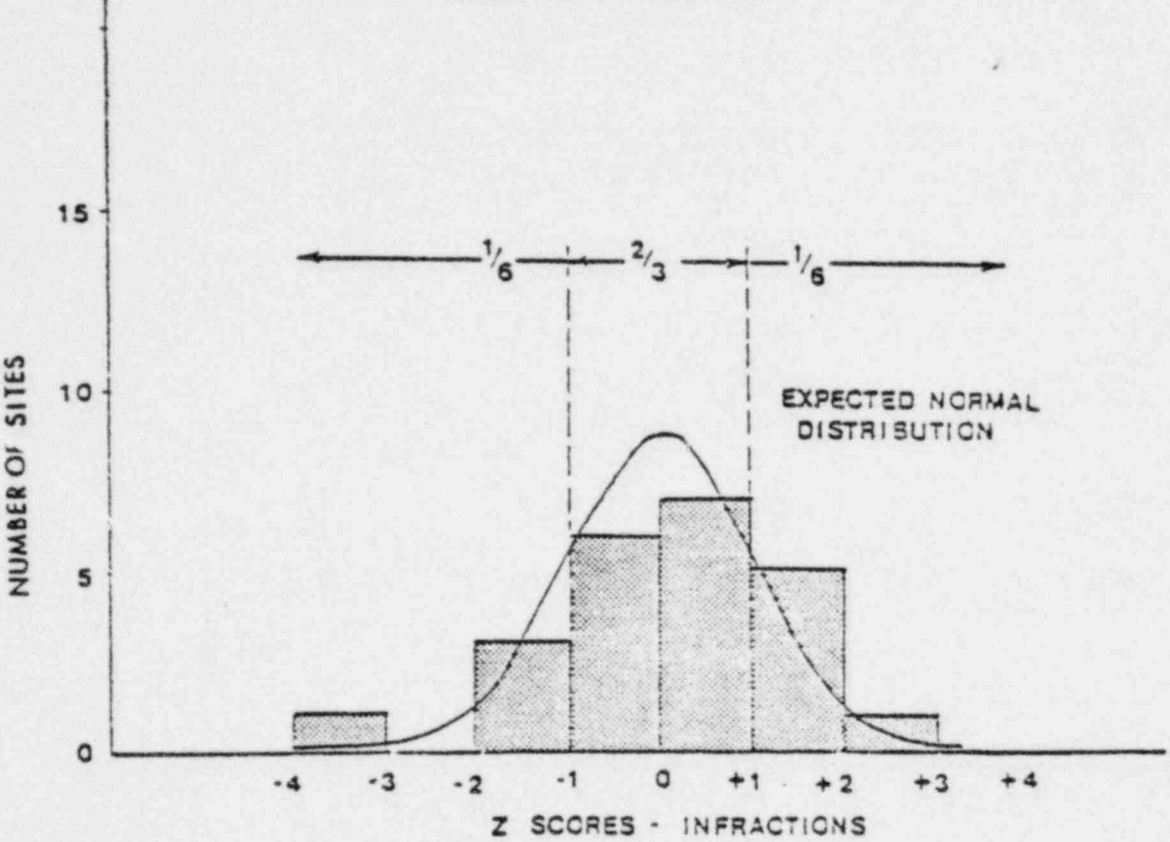
HISTOGRAM - UNADJUSTED FREQUENCY OF INFRACTION OCCURRENCE



PRESSURIZED WATER REACTORS CALENDER YEAR 1976

FIGURE NO. 3

HISTOGRAM - Z SCORE FOR ADJUSTED INFRACTIONS



PRESSURIZED WATER REACTORS CALENDER YEAR 1975

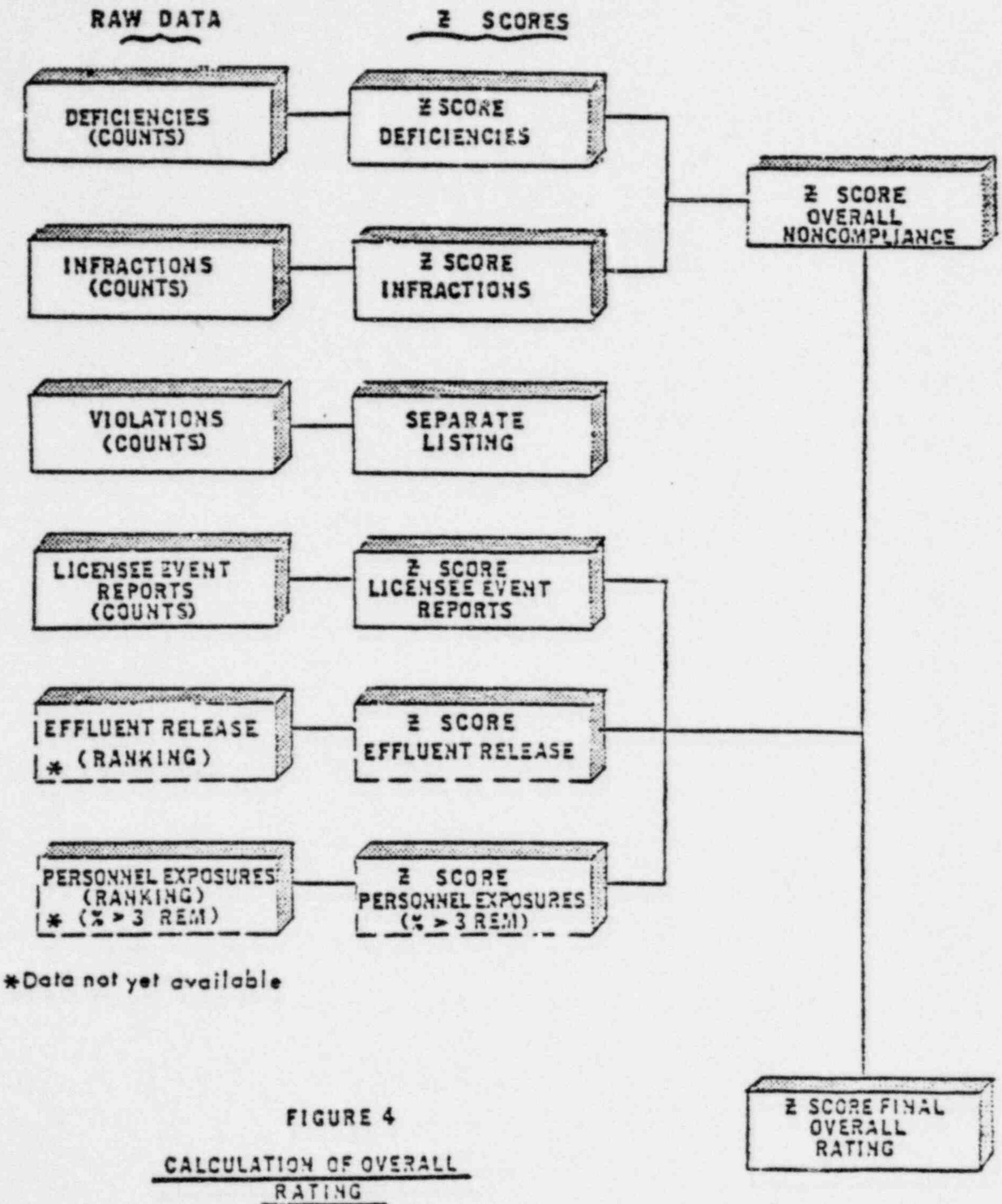


FIGURE 4

CALCULATION OF OVERALL RATING

Chapter III

ANALYSIS RESULTS

A. Introduction

The methodology described in the previous chapter has been developed through analyses of 1974 and 1975 data. This chapter summarizes the results of applying these analysis methods to data obtained in 1976. Details of the analysis are given in Appendix A.

The data analyzed are from reactors which went into commercial operation prior to 1 January 1976, with the exceptions of Indian Point 1 and Browns Ferry 1 and 2, which were shut down all or most of FH76. This leaves thirty PWRs and twenty one BWRs to be considered. At this writing, only data on noncompliances and LERs are available for FH76. Effluent release and personnel exposure data will be included in a final report on 1976 licensee performance.

The preceding chapter describes how the raw data are first to be adjusted for the effect of any identifiable variable not under management control, and then normalized. Table 1 summarizes the adjustments made for the 1976 data. Selection of the independent variables shown in Table 1 was based on patterns observed in the current and previous data and on the grounds that they were sensible. For example, one might expect that the more a reactor is inspected, the more noncompliances will be found, and the data support this hypothesis. The observed variation of performance with a variable such as date of commercial operation could reflect an aging effect or a systematic difference among reactor vintages (including license differences). In either case, it is considered appropriate to adjust the raw performance measures for this effect. There are many other candidate variables for use in adjusting performance measures. Although not all have been considered in the analysis of these 1976 data, several adjustments were included in this and previous years' analyses. The adjustments shown in Table 1 are considered meaningful, and as shown in detail in Appendix A, they do effect a considerable reduction in the variation among licensees.

Table 1

Summary of Adjustments in Analysis of 1976 Data
(Adjustments are denoted by X)

		Independent Variable		
Reactor Type	Performance Measure	Inspection HRS.	Age/Vintage	Reg. Office
PWR	Infractions	X		
	Deficiencies			X
	LERs		X	
BWR	Infractions	X	X	
	Deficiencies			X
	LERs		X	

As described in the preceding chapter, the analysis of licensee performance data leads to the calculation of Z-scores by which licensees can be compared. It is recognized, through, that with limited data, inexact adjustments, and basically homogeneous licensees, numerical Z-scores may convey an unwarranted sense of precision. Thus, for presentation of the results, the following categorization will be used.

Z-Score	Category	Designation
Greater than 1.0	"Above Average"	A
Between -1.0 and 1.0	"Average"	B
Less than -1.0	"Below Average"	C

If licensee performance is essentially homogeneous, after adjustment for factors not under licensee control, then about two-thirds of the licensees should fall in Category B, and one-sixth in each of the other two. Note that by the method of analysis, licensees are being compared against each other, rather than to some absolute norms, so this practically assures a mixture of A's, B's, and C's.

Licensee performance results for 1976 are summarized in Tables 2 and 3. Given there are the results for infractions and deficiencies, but excluding violations, then a combined noncompliance score obtained by weighting the scores for infractions and deficiencies in a 5:1 ratio. Also given is the score for LERs and an overall score obtained by weighting the scores for noncompliances and LERs in a 3:1 ratio. Appendix A gives the methods by which the scores were developed along with further results. These include results for the first and second halves of 1976 separately as well as for various subcategories of noncompliances and LERs. Appendix B describes a sensitivity study pertaining to the choice of weighting factors. Violations, while not discussed in Appendix B, result in a decrease of the Z-score by 1.86 for each violation, which would change the rating of Zion and Millstone #1 to C and while not changing Indian Point #2 which is already in the C category, would significantly reinforce the C categorization.

Table 2

Summary of Licensee Performance Analysis: PWRs

Station	Infractions	Deficiencies	Noncompliance	LERs	Overall
Yankee Rowe	B	A	B	B	B
San Onofre	A	B	A	A	A
Conn. Yankee	B	B	B	B	B
Ginna	A	B	A	B	A
Ind. Pt. 2	C	B	C	B	C
Turkey Pt. 3, 4	B	B	B	A	B
Palisades	C	A	C	B	C
Robinson	A	B	B	C	B
Point Beach 1, 2	A	B	A	B	A
Oconee 1, 2, 3	B	B	B	C	B
Surry 1, 2	B	B	B	C	C
Prairie Is. 1, 2	B	B	B	C	B
Ft. Calhoun	B	B	B	B	B
Three Mile Is.	B	B	B	B	B
Zion 1, 2	B	C	B	B	B
Kewaunee	B	B	B	B	B
Main Yankee	B	B	B	B	B
Rancho Seco	A	B	A	A	A
Arkansas 1	C	B	C	B	B
Cook	B	C	B	C	C
Calvert Cliffs	C	B	C	A	B
Millstone 2	B	B	B	A	A
Trojan	B	C	B	C	B

Table 3

Summary of Licensee Performance Analysis: BWRs

Station	Infractions	Deficiencies	Noncompliance	LERs	Overall
Dresden 1, 2, 3	C	B	C	C	C
Humboldt Bay	C	C	C	B	C
Big Rock Pt.	C	B	C	C	C
Oyster Creek	B	C	B	B	B
Nine Mile Pt.	B	B	B	B	B
Millstone 1	B	B	B	B	B
Quad Cities 1, 2	C	B	C	B	C
Monticello	A	A	A	A	A
Vt. Yankee	B	A	B	B	B
Peach Bottom 2, 3	A	B	A	B	A
Pilgrim	A	B	A	B	A
Cooper	B	B	B	A	B
Hatch	C	B	C	C	C
Brunswick	B	C	B	B	B
Duane Arnold	C	B	C	B	C
Fitzpatrick	A	B	B	A	A
Lacrosse	A	B	A	B	A

Appendix A

Results of Analysis of 1976 Data

A. Introduction

This appendix presents details and results of the analysis of licensee performance data for calendar year 1976 (CY76). Results are given first for noncompliances, then LERs, then for overall performance. Within each performance category the results for PWR's are given first, the results for BWR's second. The data are analyzed by six month periods, first half and second half (FH76 and SH76) in order to look for patterns in the data.

The reactors included in the analysis are those which went into commercial operation prior to 1 January 1976 and which were not shut down nearly all of 1976. This latter condition means Indian Point 1 and Browns Ferry 1 and 2 are excluded. The data from the 51 reactors thus considered are tabulated in the sections which pertain to their analysis.

B. Analysis of Noncompliances

Previous licensee performance evaluations, based on the 1974 and 1975 data, considered only the total noncompliances in each of the three severity categories - violations, infractions, and deficiencies. Much more detailed information is available, though, and it was decided to use some of it in the analysis of 1976 data. In particular, the three-letter "766"* codes by which each item of noncompliance is labeled were used to establish 10 types of noncompliance. These types are shown in Table 1.

Table 1

Type of Noncompliance	Notation
1. Administrative Control	(1)
2. Admin. Control/Operations	(1, 2)
3. Admin. Control/Emergency Planning	(1, 3)
4. Admin. Control/Radiological Protection	(1, 4)
5. Admin. Control/Safeguards	(1, 5)
6. Operations	(2)
7. Emergency Planning	(3)
8. Radiological Protection	(4)
9. Safeguards	(5)
10. Quality Assurance	(6)

*

1. PWR's

Table 2 lists the noncompliance frequencies of the 30 PWR's being considered by type, severity catagories, and by time period. (The reactors are listed in docket number order.) Also listed is the total inspection time in and out of office. This latter is the independent, or explanatory, variable for which adjustment is meaningful and which has been indicated by previous analyses.

Because multiple reactors, or units, at one generating station have the same management and because noncompliance findings among multiple units show a high degree of association (in many instances one occurrence at a station results in all units at that site being assessed a noncompliance), noncompliances are analyzed on a station basis, rather than a reactor basis. Station noncompliances were obtained by tabulating noncompliance frequencies by unit and by category and type of noncompliance and then obtaining the maximum number of noncompliances of each type. For example, for the first six months of 1976 (FH76), the Turkey Point infractions were as follows:

Type of Infractions

	(1)	(1,2)	(1,3)	(1,4)	(1,5)	(2)	(3)	(4)	(5)	(6)
Turkey Pt. 3	0	2	0	2	0	2	0	0	3	0
Turkey Pt. 4	0	2	0	3	0	2	0	0	3	0
Maximum	0	2	0	3	0	2	0	0	3	0

The maximum number of infractions in each infraction type is used as an estimate of the number of distinct infractions of that type at the station. This estimate tends to underestimate the actual number of distinct infractions, but not seriously, because examination of the infraction records at multi-unit stations showed that 80-90% of the citations are multiple citations. The only instance in which station maxima were not used is for Millstone where one unit is a PWR, the other a BWR.

a. Infractions

a.1. FH76

i. Total Infractions

Previous analyses have suggested a relationship between

PRESSURIZED WATER REACTORS

TABLE NO. 2

	Administrative Controls Total			Operations Administrative Controls			Emergency Planning Administrative Controls			Radiological Protection Administrative Controls			Safeguards Administrative Controls		
	Category			Category			Category			Category			Category		
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3
YANKEE ROWE															
F.H.-76		2	1		1	1					1				
S.H.-76															
C.Y.-76		2	1		1	1					1				
SAN ONOFRE															
F.H.-76		2												2	
S.H.-76															
C.Y.-76		2												2	
CONNECTICUT YANKEE															
F.H.-76		5	3		2	2			1		1			1	
S.H.-76			1			1									
C.Y.-75		5	4		2	3			1		1			1	
GINNA															
F.H.-76		2	2								1	2			
S.H.-76		1	2			2					1				
C.Y.-76		3	4			2					2	2			
INDIAN POINT #2															
F.H.-76	1	8	5		1	3				1	7	1			
S.H.-76		5	1		2	1					3				
C.Y.-75	1	13	6		3	4				1	10	1			
TURKEY POINT #3															
F.H.-76		4	3		2	2					2	1			
S.H.-75															
C.Y.-76		4	3		2	2					2	1			
TURKEY POINT #4															
F.H.-76		5	3		2	2					3	1			
S.H.-75															
C.Y.-75		5	3		2	2					3	1			

	Operations Total			Emergency Planning Total			Radiological Protection Total			Safeguards Total			Quality Assurance Total			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	Category			Category			Category			Category			Category			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
YANKEE ROWE																
F.H.-76		4	1					1			2				3	434.5
S.H.-76					1						1					279.0
C.Y.-76		4	1		1			1			3				3	713.5
SAN ONOFRE																
F.H.-76											8	3			1	286.5
S.H.-76															1	655.0
C.Y.-76											8	3			2	941.5
CONNECTICUT YANKEE																
F.H.-76		2	4		1	4		1			6					519.0
S.H.-76		1	3								1			2	1	645.0
C.Y.-75		3	7		1	4		1			6			2	1	1164.0
GINNA																
F.H.-76		2						2	2						1	619.9
S.H.-76		1	2			1		1			1	1			1	526.0
C.Y.-76		3	2			1		3	2		1	1			2	1145.9
INDIAN POINT #2																
F.H.-76		1	5		4		3	7	1		4					874.0
S.H.-76		6	2		1	1		4						5	1	904.5
C.Y.-75		7	7		6	1	3	11	1		4			5	1	1778.5
TURKEY POINT #3																
F.H.-76		4	2					2	1		3	2				284.1
S.H.-75		2									1	1		4		364.8
C.Y.-76		6	2					2	1		4	3		4		648.9
TURKEY POINT #4																
F.H.-76		4	2					3	1		3	2				250.0
S.H.-75		2									1	1		2	1	266.8
C.Y.-75		6	2					3	1		4	3		2	1	564.8

PRESSURIZED WATER REACTORS

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS			OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	
PALISADES																															
F.H.-76		4	1		2						2	1					4						2	1		6	1		1		690.5
S.H.-75		6			6												15														421.0
C.Y.-76		10	1		8						2	1					19						2	1		6	1		1		1111.5
H. B. ROBINSON #2																															
F.H.-76		2			2												4										1				395.0
S.H.-75			3			1						2					1	2								1			2	1	510.2
C.Y.-76		2	3		2	1						2					5	2								1	1		1	1	905.2
POINT BEACH #1																															
F.H.-75																		2					1			2					280.1
S.H.-75		2	1		2	1											2									1	2				216.5
C.Y.-75		2	1		2	1											2	2					1			3	2				496.6
OCONEE #1																															
F.H.-76		3	1		3	1											6	1					2			3	1		1		376.9
S.H.-76		2									2						1						3			1				1	251.3
C.Y.-75		5	1		3	1					2						7	1					5			4	1		1	1	628.2
OCONEE #2																															
F.H.-76		3	1		3	1											6	2					2			3	1		1		218.8
S.H.-76		2									2						1						1			1				1	186.0
C.Y.-75		5	1		3	1					2						7	2					3			4	1		1	1	404.8
SURRY #1																															
F.H.-75		4	5		1	4					2	1					4	4					3	1		4			1	1	440.5
S.H.-75		2	1			1					1						2	2					2			2					512.2
C.Y.-75		6	6		1	5					3	1					6	6					5	1		6			1	1	952.7
SURRY #2																															
F.H.-75		3	5			4					2	1					2	4					4	1		4			2		472.1
S.H.-75		2	1			1					1						4	2					2			2					346.4
C.Y.-76		5	6			5					3	1					6	6					6	1		6			2		818.5

PRESSURIZED WATER REACTORS

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS			OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	
PRAIRIE ISLAND #1																															
F.H.-76		1				3					1						1	3					2			3					336.0
S.H.-76		1			1												3	1								1					237.5
C.Y.-76		2			1						1						4	4					2			4					573.5
FORT CALHOUN																															
F.H.-76		2	1		2	1											2	1								1					349.0
S.H.-76		4			1						3						3						3						3	3	579.5
C.Y.-76		6	1		3	1					3						5	1					3			1			3	3	928.5
OCONEE #3																															
F.H.-76		3			3												6						2			3	1		1		297.5
S.H.-76		2									2						2						3			1			1	1	281.0
C.Y.-76		5			3						2						8						5			4	1		2	1	578.5
THREE MILE ISLAND #1																															
F.H.-76		2	1			1					1			1			3	1					1			5	2		1	1	859.0
S.H.-76		6	3		2	2					3						4	3		1			3	1			2		1	1	632.0
C.Y.-76		8	4		2	3					4			1			7	4					4	1		5	4		2	2	1540.0
ZION #1																															
F.H.-76		3	2		1	2					2						10	8				2	3						1		827.5
S.H.-76		1	6		1	6											5	6					1			4					533.2
C.Y.-76		4	8		2	8					2						15	14				2	4			4			1		1360.7
POINT BEACH #2																															
F.H.-76			1			1											1	3								2					306.4
S.H.-76			1			1												1								1	2				110.5
C.Y.-75			2			2											1	4								3	2				416.9
ZION #2																															
F.H.-75		2	2			2					2						6	5											1		550.5
S.H.-76			6			6											4	6					1			5					460.7
C.Y.-75		2	8			8					2						10	11					1			5			1		1010.2

PRESSURIZED WATER REACTORS

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS			OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
KEWAUNEE																															
F.H.-76			1			1												4					2			2	1				470.5
S.H.-76		2			1						1						3						1						1		199.0
C.Y.-76		2	1		1	1					1						3	4					3			2	1		1		669.5
PRAIRIE ISLAND #2																															
F.H.-76			2			2												2					1			3					320.5
S.H.-76		3			3												5	1								1					274.0
C.Y.-76		3	2		3	2											5	3					1			4					594:5
MAINE YANKEE																															
F.H.-76		2	2								2	2					1						2	2		1	2				389.5
S.H.-76		4	4		1	4					3												3				1		1	1	434.0
C.Y.-76		6	6		1	4					6	2					1						5	2		1	3		1	1	823.5
RANCHO SECO #1																															
F.H.-76																	1	2													261.0
S.H.-76		2			2												3									4	2				612.2
C.Y.-76		2			2												4	2								4	2				873.2
ARKANSAS #1																															
F.H.-76		3	2		3	2											5	2												1	244.0
S.H.-76		3			1						2						1	1					2						3		323.0
C.Y.-76		6	2		4	2					2						6	3					2						3	1	667.0
COOK #1																															
F.H.-76		4	5		2	4		1			1	1					7	5		1	1		1	1		2					515.7
S.H.-76			2			2											1	5								1					387.0
C.Y.-76		4	7		2	6		1			1	1					8	10		1	1		1	1		3					902.7
CALVERT CLIFFS #1																															
F.H.-76																										5					360.5
S.H.-76		2	4		2	3									1		5	7								2	1		3	4	379.5
C.Y.-76		2	4		2	3									1		5	7								7	1		3	4	760.0

PRESSURIZED WATER REACTORS

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS			OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
MILLSTONE #2																															
F.H.-76			3			2									1		3	3		2							4		4	1	672.5
S.H.-76		1	1			1					1							2					1			2					503.0
C.Y.-76		1	4			3					1						3	5		2			1			2	4		4	1	1175.5
TROJAN																															
F.H.-76			2			1						1					1	2					1	1						2	311.5
S.H.-76			3			1						2					2	5						2		5			1		593.9
C.Y.-76			5			2						3					3	7					1	3		5			1	2	905.4

infraction frequency and inspection effort and further support for such a relationship is found in the present data. The estimated relationship is

$$\widehat{INF} = 1.3 \text{ (Inspection Hrs./100)},$$

where Inspection Hours are the total in and out of office inspection hours at a station for the first six months of 1976. A chi-square goodness of fit statistic for this relationship shows a quite adequate fit ($\chi^2 = 25.0$ on 22 df) so no further adjustment seems necessary. Thus, Z-scores will be calculated by

$$Z_{INF} = 2 \ \sqrt{NCM} - \sqrt{INF}$$

where NCM = 1.3 (Insp. Hrs./100)

ii. Administrative Control Infractions

Total administrative control (AC) infractions are obtained by adding the infraction frequencies of (1,2), (1,3), (1,4), and (1,5) type infractions. Fitting AC infractions as a function of inspection hours yields

$$\widehat{INF}_{AC} = 0.4 \text{ (Insp. Hrs./100)}$$

and provides a fit which is not particularly good (the χ^2 value of 34.6 on 22 df falls just above the 5% significance level) but not bad enough to warrant further fitting. Thus, Z-scores will be calculated, as above, using this expression for the nominal number of AC infractions.

iii. Operations

Total operations (OP) infractions are obtained by adding types (2) and (1,2). Note that this sum overlaps that of AC infractions. Fitting OP infraction frequency versus inspection hours yields

$$\widehat{INF}_{OP} = 0.5 \text{ (Insp. Hrs./100)}$$

and a goodness of fit chi-square value, after omitting Arkansas 1, of $\chi^2 = 28.1$ of 21 df. Z-scores will be based on this fitted relationship.

iv. Emergency Planning

Too few emergency planning infractions have occurred to warrant analysis.

v. Radiological Protection and Control

Too few radiological protection and control infractions have occurred to warrant analysis.

vi. Safeguards

Total safeguards (SG) infractions are obtained by adding (1,5) and (5). The special nature of safeguards and safeguards inspection suggest that it would be inappropriate to adjust SG information frequencies for total inspection time. Examining the data indicates that the variation of SG infractions among stations is large compared to the sort of variation one would expect by chance alone ($\chi^2 = 47.5$ on 22 df). Further consideration of safeguards inspection hours, which ranged from 12 to 115 for the 23 PWRs in this period, does not reduce this variation. Because of this heterogeneity, Z-scores for SG infractions will not be calculated for first-half of 1976. This, however, does not suggest that future data cannot be used in the manner described above to obtain meaningful results. FH 76 was a period of changing emphasis in the area of safeguards, causing extreme variations in frequency and duration of inspections.

vii. Quality Assurance

Too few QA infractions occurred to warrant analysis.

a.2. SH76

i. Total Infractions

Infractions were incurred in SH76 at about the same rate as in FH76. The fitted relationship between infraction frequency and inspection hours turns out to be

$$\widehat{INF} = 1.2 \text{ (Insp. Hrs./100)}.$$

The chi-square goodness of fit statistic shows that with the exception of two stations, an adequate fit is obtained, but not as good a fit as in FH76 (χ^2 = 23.0 on 20 df, after omitting Palisades and San Onofre). Thus, Z-scores will be calculated based on the above relationship.

ii. Administrative Control

AC infractions in SH76 were less frequent and more heterogeneous than in FH76 and so were not converted to Z-scores.

iii. Operations

After omitting the data from Palisades and San Onofre, the fitted model is:

$$\widehat{INF}_{OP} = 0.5 \text{ (Insp. Hrs/100)}$$

and the chi-square value obtained is 22.1 on 20 df.

iv-vii. Too few infractions of these types were incurred to warrant analysis.

a.3. CY76

i. Total Infractions

The total number of infractions for the 23 PWR stations considered in CY76 was 330 and total inspection time, in hundreds of hours, was 259.6. Thus, the average number of infractions per hundred inspection hrs. was 1.27, which is the basis of the line plotted in Figure 1, Chapter II. Thus, Z-scores for FY76 will be based on the relationship

$$\widehat{INF} = 1.3 \text{ (Insp. Hrs./100)}.$$

The chi-square goodness of fit statistic indicates that, with the exception of Palisades, an adequate fit is provided by this relationship (χ^2 = 21.1 on 21 df). Given the consistency shown between FH and SH76 results, this is not unexpected.

ii. Administrative Control

Results for AC infractions were not consistent for the two six-month periods, but broadening the time span to the full year provides more stability. In particular, AC infractions do not appear related to inspection time but are essentially homogeneous (after omitting Indian Pt. 2 and Palisades, $\chi^2 = 22.2$ on 20 df). The average number of AC infractions is about 4.0 so this value will be used as the nominal value in obtaining Z-scores.

iii. Operations

The adjustment of OP infractions for both FH and SH 76 data was based on the relationship,

$$\widehat{INF}_{OP} = 0.5 \text{ (Insp. Hrs./100)}$$

For the full year this relationship also provides an adequate fit (excluding Palisades, $\chi^2 = 20.7$ on 21 df) and Z-scores will be based on it.

iv.,v. Emergency Planning and Radiological Protection

There were too few infractions of these two types in CY76 to warrant analysis.

vi. Safeguards

Over the full year, SG infractions were reasonably homogeneous, averaging about 4 infractions per station (and yielding a χ^2 value of 25.8 on 22 df) so Z-scores will be based on a nominal value of 4 SG infractions.

vii. Quality Assurance

There were too few QA infractions to warrant analysis.

b. Deficiencies

b.1. FH76

i. Total Deficiencies

Deficiency frequencies, in contrast to infractions, do not appear to be associated with inspection hours. A chi-square test for homogeneity shows that the variation among deficiency frequencies is not larger than what would be expected by chance alone ($\chi^2 = 27.5$ on 22 df). The average number of deficiencies per station is about 4.0 so this will be used as the nominal frequency in calculating Z-scores.

ii.-vii. Deficiency frequencies, by type, are too small to warrant analysis.

b.2. <u>SH76</u>

i. <u>Total Deficiencies</u>

As with FH76, total deficiencies in SH76 are fairly homogeneous, so no adjustment is made. ($\chi^2 = 32.5$ on 21 df after omitting Calvert Cliffs.) The average number of deficiencies, per station is about 3.0 for SH76 so Z-scores will be based on this nominal value.

ii.-vii. Deficiency frequencies by type are too small for further analysis.

b.3. <u>CY76</u>

i. <u>Total Deficiencies</u>

In both FH and SH76, PWR deficiencies were independent of inspection hours, but when the CY 76 totals are considered there is more evidence of heterogeneity than over the half year periods. Analysis of previous years data showed a possible association of deficiency frequency with the IE Regional Office, so this possibility was considered for the CY76 data. Comparison of deficiency frequency by regions showed that licensees in Region I incurred about twice as many deficiencies, on the average, than those in the other Regions. The nominal values on which Z-scores will be based are:

$\widehat{DEF}$ = 10.0, for Region I
= 5.5, for the other Regions.

The goodness of fit chi-square for this relationship, omitting Cook and Zion, is 16.8 on 19 df.

ii. Administrative Control

AC deficiencies are fairly homogeneous and do not show an association with Regional Office. The average frequency, per station, is about 3 AC deficiencies and a chi-square test based on this nominal value yields $\chi^2 = 23.7$ on 20 df, after excluding two stations, Cook and Zion. Z-scores will be based on a nominal value of 3.0.

iii. Operations

Two stations - Cook and Zion again (recall the overlap between AC and OP total frequencies) - have outlying OP frequencies, but the remaining are fairly homogeneous ($\chi^2 = 30.4$ on 20 df) about an average of 3.5 deficiencies. Z-scores will be based on a nominal of 4 OP deficiencies.

iv.-vii. Deficiency frequencies by the remaining types were too small to warrant analysis.

c. Summary of Results

Table 3 provides a summary of the results from the preceding analysis of noncompliances. Recall that a Z-score greater than 1.0 is indicated by an A, between -1.0 and 1.0 by B, and below -1.0 by C. Generally, but not always, the total Z-score is an "average" of the Z-scores by types. Exceptions can occur due to unusually large or small frequencies among the type of noncompliance for which separate Z-scores were not obtained or because several scores near a borderline can lead to a total score across the borderline.

Table 3

Summary of Licensee Performance Analysis: PWR Noncompliance

		TYPE OF NONCOMPLIANCE							
		Admin. Control		Oper-ations		Safe-guards		TOTAL	
Station	Period	INF	DEF	INF	DEF	INF	DEF	INF	DEF
Yankee Rowe	FH76	B		C		B		B	B
	SH76	-		A		-		B	A
	CY76	A	A	B	A	B		B	A
San Onofre	FH76	B		A		C		C	B
	SH76	-		A		-		A	A
	CY76	A	A	A	A	C		A	B
Conn. Yankee	FH76	C		B		B		B	C
	SH76	-		A		-		A	B
	CY76	B	B	A	C	B		B	B
Ginna	FH76	B		B		B		A	B
	SH76	-		A		-		A	C
	CY76	B	B	A	B	A		A	B
Indian Pt. 2	FH76	C		A		A		C	B
	SH76	-		B		-		C	B
	CY76	C	C	B	C	B		C	B
Turkey Pt. 3, 4	FH76	C		B		B		B	B
	SH76	-		B		-		B	B
	CY76	B	B	B	B	B		B	B
Palisades	FH76	B		B		B		C	A
	SH76	-		C		-		C	A
	CY76	C	A	C	A	B		C	A
Robinson	FH76	B		C		B		B	A
	SH76	-		A		-		B	C
	CY76	A	A	B	B	A		A	B
Point Beach 1, 2	FH76	A		A		A		A	B
	SH76	-		B		-		B	B
	CY76	A	[illegible]	B	B	B		A	B
Oconee 1, 2, 3	FH76	B		B		C		B	B
	SH76	-		B		-		B	A
	CY76	B	A	B	B	B		B	B
Surry 1, 2	FH76	B		B		B		B	B
	SH75	-		B		-		B	B
	CY76	B	C	B	C	B		B	B

Table 3 (cont'd)
Summary of Licensee Performance Analysis: PWR Noncompliance

		TYPE OF NONCOMPLIANCE							
		Admin. Control		Oper- ations		Safe- guards		TOTAL	
Station	Period	INF	DEF	INF	DEF	INF	DEF	INF	DEF
Prairie Is. 1, 2	FH76	A		A		A		A	[illegible]
	SH76	-		C		-		B	A
	CY76	B	B	B	B	B		B	B
Fort Calhoun	FY76	B		B		B		B	A
	SH76	-		B		-		B	B
	CY76	B	A	B	A	A		B	B
Three M. Is.	FY76	B		B		B		B	B
	SH76	-		B		-		B	C
	CY76	C	C	B	B	B		B	B
Zion 1, 2	FH76	A		C		C		B	C
	SH76	-		B		-		B	C
	CY76	B	C	B	C	B		B	C
Kewaunee	FH76	A		A		A		B	B
	SH76	-		C		-		C	A
	CY76	A	A	B	B	A		B	B
Main Yankee	FH76	B		B		A		B	B
	SH76	-		B		-		B	C
	CY76	B	A	A	B	A		B	B
Rancho Seco	FH76	A		B		A		A	A
	SH76	-		B		-		B	B
	CY76	A	A	B	B	B		A	B
Arkansas	FH76	C		C		B		B	B
	SH75	-		B		-		B	A
	CY76	B	B	C	B	A		C	B
Cook	FH76	C		C		C		C	C
	SH76	-		B		-		A	C
	CY76	B	C	C	C	B		B	B
Calvert Cliffs	FH76	A		A		A		B	A
	SH76	-		C		-		C	C
	CY76	A	B	B	C	C		C	B
Millstone 2	FH76	A		B		B		B	C
	SH76	-		A		-		A	B
	CY76	A	B	A	B	A		B	B
Trojan	FH76	A		B		A		A	B
	SH76	-		B		-		B	C
	CY76	A	C	B	C	B		B	C

2. BWR's

Table 4 lists the noncompliance data and inspection hours for the 21 BWR's considered. As with PWR's, the data for multi-unit stations will be converted to frequencies pertaining to the station.

a. Infractions

a.1. FH76

i. Total Infractions

BWR infraction frequencies, by stations, again show evidence of an association with inspection hours, but the pattern is not consistent across stations as it was among PWRs. In particular, the four BWRs which began commercial operation (CO) in 1975 show about twice as many infractions per hour of inspection as do the remainder. The relationship fitted is

$$\widehat{INF} = 1.0 \text{ (Insp. Hrs./100, if CO prior to 1975}$$
$$= 2.0 \text{ (Insp. Hrs./100), if CO in 1975.}$$

The chi-square for goodness of fit of this model equals 23.3 on 15 df. Thus, Z-scores will be calculated with nominal values given by the above expressions.

ii.-vii. Infraction frequencies, by type, are either too small or too heterogeneous to warrant analysis.

a.2. SH76

i. Total Infractions

The association noted in FH76 of infraction frequency with date of commercial operation is not repeated with any consistency - two of the four stations which began commercial operation in 1975 had an unusually high number of infractions, compared to other BWRs, two did not. The bulk of the data, 15 out of 17 stations, are adequately fit by the model

$$\widehat{INF} = 1.0 \text{ (Insp. Hrs./100).}$$

The goodness of fit chi-square value being 11.7 of 14 df.

BOILING WATER REACTORS

TABLE NO.

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS		
	CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY		
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3
PILGRIM #1															
F.H.-76		5	1								4	1			
S.H.-76		2	1		1	1					1				
C.Y.-76		7	2		1	1					5	1			
COOPER															
F.H.-76		1			1										
S.H.-76		1	1		1	1									
C.Y.-76		2	1		2	1									
HATCH #1															
F.H.-76		2	1		1										
S.H.-76		8	2		7	2					1				
C.Y.-76		10	3		8	2					1				
BRUNSWICK #2															
F.H.-76		2 -	3		1	3					1				
S.H.-76		3	1		2	1					1				
C.Y.-76		5	4		3	4					2				
DUANE ARNOLD															
F.H.-76		1	1		1	1.									
S.H.-76		6			6										
C.Y.-76		7	1		7	1									
FITZPATRICK															
F.H.-76			3			3									
S.H.-76		5	2		3	2								2	
C.Y.-76		5	5		3	5								2	
LA CROSSE															
F.H.-76		1			1										
S.H.-76			3			3									
C.Y.-76		1	3		1	3									

	OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
PILGRIM #1																
F.H.-76								5	1		5	2				779.0
S.H.-76		1	4					2	2			1		2	3	695.0
C.Y.-76		1	4					7	3		5	3		2	3	1474.0
COOPER																
F.H.-76		1	1											5	2	434.0
S.H.-76		1	1											3	1	597.0
C.Y.-76		2	2											8	3	1031.0
HATCH #1																
F.H.-76		8	1								1			2	2	526.2
S.H.-76		10	3					2								595.5
C.Y.-76		18	4					2			1			2	2	1121.7
BRUNSWICK #2																
F.H.-76		4	8					3			2	2		3		606.5
S.H.-76		4	2					1			1	1				450.5
C.Y.-76		8	10					4			3	3		3		1057.0
DUANE ARNOLD																
F.H.-76		7	4											5	1	768.0
S.H.-76		10									4			8		440.5
C.Y.-76		17	4								4			13		1208.5
FITZPATRICK																
F.H.-76			3								9	1		1	4	550.5
S.H.-76		3	7								2					652.0
C.Y.-76		3	10								11	1		1	4	1252.5
LA CROSSE																
F.H.-76		1						2							1	577.5
S.H.-76			3													349.0
C.Y.-76		1	3												1	925.5

BOILING WATER REACTORS

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS			OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	CATEGORY 1	2	3	
DRESDEN #3																															
F.H.-76			1			1						1					2	1						1		9					316.0
S.H.-76		1	2		1	2											3	2								1			2		250.0
C.Y.-76		1	3		1	3											5	3						1		10			2		574.0
QUAD CITIES #1																															
F.H.-76		1	1		1												3						1	1		2					500.5
S.H.-76		2	1		1	1		1									5	2		2			1			1			1	2	361.0
C.Y.-76		3	2		2	1		1									8	2		2			2	1		3			1	2	861.5
MONTICELLO																															
F.H.-76		1	1			1					1							1					1							1	594.0
S.H.-76		1			1												1									1					203.0
C.Y.-76		2	1		1	1					1						1	1					1			1				1	797.0
QUAD CITIES #2																															
F.H.-76			1									1					3						2	1		2					303.0
S.H.-76		2	1		1	1		1									2	2		2			1			1			3	3	426.0
C.Y.-76		2	2		1	1		1				1					5	2		2			3	1		3			3	3	729.0
VERMONT YANKEE																															
F.H.-76		1			1												2	1						2		1	1				469.5
S.H.-76		2	3		2	3											4	4			1					1			3	2	664.5
C.Y.-76		3	3		3	3											6	5			1			2		2	1		3	2	1354.0
PEACH BOTTOM #2																															
F.H.-76		2	3			3					2						3	7					2				1				731.5
S.H.-76		1	5		1	3			1			1					6	6						1					3		581.5
C.Y.-76		3	8		1	6			1		2	1					9	13					2	1			1		3		1316.0
PEACH BOTTOM #3																															
F.H.-76		2	3			3					2						2	5					2				1				403.0
S.H.-76			4			3						1					5	6						1					3		518.0
C.Y.-76		2	7			6					2	1					7	11					2	1			1		3		921.0

BOILING WATER REACTORS

	ADMINISTRATIVE CONTROLS TOTAL			OPERATIONS ADMINISTRATIVE CONTROLS			EMERGENCY PLANNING ADMINISTRATIVE CONTROLS			RADIOLOGICAL PROTECTION ADMINISTRATIVE CONTROLS			SAFEGUARDS ADMINISTRATIVE CONTROLS			OPERATIONS TOTAL			EMERGENCY PLANNING TOTAL			RADIOLOGICAL PROTECTION TOTAL			SAFEGUARDS TOTAL			QUALITY ASSURANCE TOTAL			TOTAL INSPECTION TIME IN & OUT OF OFFICE
	CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			CATEGORY			
	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	1	2	3	
DRESDEN #1																															
F.H.-76		1	3		1	3											3	3					1			9			1		325.5
S.H.-76																	1	1								1					219.5
C.Y.-76		1	3		1	3											4	4					1			10					545.0
HUMBOLDT BAY																															
F.H.-76																										4				9	338.5
S.H.-76		3	1		3	1											3	1								5				1	587.5
C.Y.-76		3	1		3	1											3	1								9				10	926.0
BIG ROCK POINT																															
F.H.-76			2			2											6	2								3			1		679.0
S.H.-76		2	2		1	1					1				1		2	1					2				2		1		297.5
C.Y.-76		2	4		1	3					1				1		8	3					2			3	2		2		976.5
OYSTER CREEK																															
F.H.-75		3	3		1	2			1		2						1	4		1	4		3				4		1	4	831.0
S.H.-76			2						1			1					1			1	2			1			3		2	1	467.5
C.Y.-76		3	5		1	2			2		2	1					2	4		2	6		3	1			7		3	5	1298.5
NINE MILE POINT #1																															
F.H.-76		2	1		1						1	1					1	1					4	1					2	3	810.5
S.H.-76		3	1		2	1					1						4	4					2				2			1	609.5
C.Y.-76		5	2		3	1					2	1					5	5					6	1			2		2	4	1420.0
DRESDEN #2																															
F.H.-76		1	3		1	3											5	4						1		3			6	1	600.2
S.H.-76		2	1			1					2						3	2					2			1			2		192.0
C.Y.-75		3	4		1	4					2						8	6					2	1		4			8	1	792.2
MILLSTONE #1																															
F.H.-76		3	3		3	1									2		5	1		2			3				7		1	2	821.0
S.H.-76		4			3						1					1	6	3					1			1			1		1271.0
C.Y.-76		7	3		6	1					1				2	1	11	4		2			4			1	7		2	2	2092.0

ii.-vii. Too few infractions, by type, were incurred to warrant analysis.

a.3. CY76

i. Total Infractions

The results of the analyses of FH and SH76 data suggests that, for the most part, total BWR infractions are related to inspection effort by

$$\widehat{INF} = 1.0 \text{ (Insp. Hrs./100)}.$$

Combining the data over the full year leads to some refinement. The fitted relationship becomes

$$\widehat{INF} = 0.9 \text{ (Insp. Hrs./100) if CO prior to 1975}$$
$$= 1.6 \text{ (Insp. Hrs./100) if CO in 1975}$$

The chi-square for goodness-of-fit of this model, after omitting Dresden and Duane Arnold, is 19.5 on 13 df, which is adequately small so that Z-scores will be based on nominal values given by this relationship.

ii. Administrative Control

As with PWRs, AC infractions appear to be fairly homogeneous, independent of inspection effort, and to average about 4 per station for CY76. (The chi-square value for testing for homogeneity equals 22.1 on 16 df.) Thus, Z-scores will be obtained based on a nominal AC infraction frequency of 4.0.

iii. Operations

As with PWRs, BWR OP infractions appear related to inspection hours, the fitted relationship being

$$\widehat{INF}_{OP} = 0.4 \text{ (Insp. Hrs./100)}.$$

The goodness-of-fit chi-square statistic for this model, excluding Hatch and Duane Arnold, is $\chi^2 = 20.5$ on 14 df.

iv., v. Emergency Planning and Radiological Protection

Too few EP and RP infractions were incurred to warrant analysis.

vi. Safeguards

Safeguards infractions were quite heterogeneous - 3 of the 17 BWR stations incurred 30 of the 48 SG infractions in CY76 - so Z-scores will not be calculated for this type of infraction.

vii. Quality Assurance

The variation among QA infraction frequencies was similar to that of SG infractions, so no Z-scores will be calculated.

b. Deficiencies

b.1. FH76

i. Total Deficiencies

Total deficiencies do not show evidence of an association with inspection hours, but neither do they appear to be completely homogeneous. The analysis results for PWRs suggest a consideration of Regional differences. For BWRs, about twice as many deficiencies are incurred by licensees in Region I as elsewhere. (There are three outlying frequencies - those for Pilgrim, in Region I, and for Humboldt Bay, and Brunswick in other Regions. However, as will be seen, when the yearly totals are considered, less erratic variation is encountered.) Z-scores will be based on the following nominal values:

$\widehat{DEF}$ = 8.0, for Region I
= 4.0, for other Regions

ii.-vii. Deficiency frequencies, by type, were too infrequent to warrant analysis.

b.2. SH76

i. Total Deficiencies

Fewer deficiencies were incurred in SH76 than in FH76, but the association with Regional Office persisted. The nominal values which will be used to obtain Z-scores are:

$\widehat{DEF}$ = 7.0, for Region I
= 2.0, for other Regions.

The chi-square value for goodness of fit of this relationship is 12.6 on 15 df, with no stations excluded, so the fit is quite adequate.

ii.-vii. Deficiency frequencies, by type, were too small to warrant analysis.

b.3. CY76

i. Total Deficiencies

For the full year, deficiency frequencies, by Region, lead to the following nominal values:

$\widehat{DEF}$ = 14, for Region I
= 6, for other Regions.

The chi-square value for goodness of fit of this relationship is χ^2 = 21.9 on 15 df (and 13.7 on 14 df if Brunswick is omitted), which indicates an adequate fit, so Z-scores will be calculated from these nominals.

ii. Administrative Control

AC deficiencies are fairly homogeneous (χ^2 = 17.7 on 16 df for testing homogeneity), and appear to be independant of inspection hours and Regional Office. The average number of AC deficiencies per station was about 3.0 and Z-scores will be based on this nominal value.

iii. Operations

Three stations - Peach Bottom, Brunswick, and Fitzpatrick - had about three times as many OP deficiencies as the remaining stations. Among the remaining 14 stations the OP frequencies were quite homogeneous (χ^2 = 8.6 on 13 df). As two of the three outliers are in Region I, the Regional difference observed among total deficiencies suggests itself, but the evidence does not warrant a separate nominal value for Region I. The nominal OP value used in obtaining Z-scores will be 4.0.

iv.-vii. Deficiency frequencies for the remaining types were too small to warrant analysis.

c. Summary of Results

The results of calculating the Z-scores based on the preceding analysis of BWR noncompliances are given in Table 5.

Appendix A

Table 5

Summary of Licensee Performance Analysis: BWR Noncompliances

Station	Period	Type of Noncompliance: Admin. Control INF	Admin. Control DEF	Operations INF	Operations DEF	Total INF	Total DEF
Dresden 1, 2, 3	FH76					C	B
	SH76					B	B
	CY76	B	B	B	B	C	B
Humboldt Bay	FH76					B	C
	SH76					B	B
	CY76	B	A	B	A	C	C
Big Rock Pt.	FH76					C	A
	SH76					C	B
	CY76	A	B	C	B	C	B
Oyster Cr.	FH76					B	C
	SH76					B	B
	CY76	B	C	A	B	B	C
Nine Mile Pt.	FH76					B	A
	SH76					B	B
	CY76	B	B	B	B	B	B
Millstone 1	FH76					B	B
	SH76					A	A
	CY76	C	B	B	B	B	B
Quad Cities 1, 2	FH76					B	A
	SH76					C	C
	CY76	B	B	B	A	C	B
Monticello	FH76					A	A
	SH76					B	A
	CY76	A	A	A	A	A	A
Vt. Yankee	FH76					B	A
	SH76					B	B
	CY76	B	B	B	B	B	A

Table 5 (cont'd)

Summary of Licensee Performance Analysis: BWR Noncompliances

		Type of Noncompliance					
		Admin. Control		Operations		Total	
Station	Period	INF	DEF	INF	DEF	INF	DEF
Peach Bottom	FH76					A	B
	SH76					B	B
	CY76	B	C	B	C	A	B
Pilgrim	FH76					A	A
	SH76					B	C
	CY76	C	B	A	B	B	B
Cooper	FH76					C	B
	SH76					B	B
	CY76	A	A	A	A	B	B
Hatch	FH76					B	B
	SH76					C	B
	CY76	C	B	C	B	C	B
Brunswick	FH76					B	C
	SH76					B	B
	CY76	B	B	C	C	B	C
Duane Arnold	FH76					B	B
	SH76					C	A
	CY76	C	A	C	B	C	B
Fitzpatrick	FH76					B	B
	SH76					B	B
	CY76	B	B	A	C	A	B
Lacrosse	FH76					A	A
	SH76					A	B
	CY76	A	B	A	B	A	B

C. Analysis of Licensee Event Reports (LERs)

The data analyzed, in the case of multi-unit stations, are the total LERs, of the types listed below, over the units at a station. This performance measure is used because examination of LER details indicates that these events are reported singly; one event does not result in multiple LERs, in contrast to the situation with noncompliances. In addition to considering total LERs, it was determined that various types, or subcategories, of LERs could be identified. From reading the LER files, each LER was classified according to the following matrix.

	Personnel Error	Procedural Error
Operations		
Maintenance		
Env. & Health Physics		

Other types of LERs, such as component failures or design error, are not included in the analysis because they are not as reflective as licensee performance. Thus, an analysis can be done of different types of LERs, similar to the analysis of different types of noncompliances. However, the frequencies are small enough that LER types will only be considered for the yearly totals. PWR and BWR LER data are given in Tables 6 and 8, respectively.

1. PWRs

a. FH76 and SH76

i. Total LERs

In both six month periods, total LERs varied more among stations than what would be expected if LER frequencies were homogeneous. Further examination of the data indicates the LER frequencies are associated with age. Newer stations tend to have more LERs than older. The

Appendix A

Table 6

PWR LER Frequencies

Station	Period	Personnel Error			Procedural Error		
		Operations	Maint.	Env..& HP	Operations	Maint.	Env. & HP
	FH76	1				1	
Yankee Rowe	SH76						
	CY76	1				1	
	FH76						
San Onofre	SH76						
	CY76						
	FH76		1		1		
Conn. Yankee	SH76		1				
	CY76		2		1		
	FH76		2		1		
Ginna	SH76						
	CY76		2		1		
	FH76	2		1			
Ind. Pt. 2	SH76	2	2				
	CY76	4	2	1			
	FH76						
Turkey Pt.	SH76		1				
	CY76		1				
	FH76	2	1				
Palisades	SH76	1	1				
	CY76	3	2				
	FH76	1	2				
Robinson	SH76		3				
	CY76	1	5				
	FH76	2	1				
Pt. Beach	SH76						
	CY76	2	1				
	FH76	4	4		1	1	2
Oconee	SH76	3	3		1	1	
	CY76	7	7		2	2	2
	FH76	1	1		1		
Surry	SH76	3		2			
	CY76	4	1	2	1		

Appendix A

Table 6 (cont'd)

PWR LER Frequencies

Station	Period	Personnel Error: Operations	Personnel Error: Maint.	Personnel Error: Env. & HP	Procedural Error: Operations	Procedural Error: Maint.	Procedural Error: Env. & HP
Prairie Is.	FH76	2	3		1	3	
	SH76	2	4	1	1		
	CY76	4	7	1	2	3	
	FH76		2	1		1	
Ft. Calhoun	SH76	1	2	1		1	
	CY76	1	4	2		2	
	FH76	2	2		2	2	
Three M. Is.	SH76	1	2	1			
	CY76	3	4	1	2	2	
	FH76	4			2		1
Zion	SH76	3	1				
	CY76	7	1		2		1
	FH76	2				1	1
Kewaunee	SH76	2	4	3			
	CY76	4	4	3		1	1
	FH76	1					
Maine Yankee	SH76	1					
	CY76	2					
	FH76	1	1			1	
Rancho Seco	SH76	1	1				
	CY76	2	2			1	
	FH76	2			1		
Arkansas	SH76	3	2				
	CY76	5	2		1		
	FH76	3	1		2	1	
Cook	SH76	4	1	2			
	CY76	7	2	2	2	1	
	FH76	1					
Calvert Cliffs	SH76	1	1	1			
	CY76	2	1	1			
	FH76	3					
Millstone 2	SH75	2				1	1
	CY76	5				1	1
	FH76	6			8		
Trojan	SH76	2	1				2
	CY76	8	1		8		2

trend is not a smooth one, but rather a dichotomy occurs. This leads to the following choice of nominal values:

$$\widehat{LER} = 2.0, \text{ if CO date is prior to 1973,}$$
$$= 5.0, \text{ if CO date is 1973 or later.}$$

For multi-unit stations, the CO date used is that of the first unit in operation. After excluding two FH76 observations - Trojan and Oconee - the chi-square values obtained (20.9 on 19 df and 24.0 on 21 df, for FH76 and SH76, respectively) indicate an adequate fit so Z-scores will be calculated accordingly.

b. CY76

i. Total LERs

The consistant patterns noted in the preceding paragraph over FH and SH76 of necessity carry over to the CY76 totals. The average number of LERs among the older stations (CO prior to 1973) is about 3.5 and for the remaining stations (excluding Oconee and Trojan) is about 10.0, so these values will be taken as the nominal LER CY76 frequencies in calculating Z-scores. The chi-square goodness of fit value for this relationship is 31.3 on 19 df, slightly above the upper fifth percentile, but not enough to warrant further adjustment.

ii. Personnel Errors

The variation among the subtotals of LERs involving personnel errors follows a similar pattern as that of total LERs (about three-fourths of all LERs are personnel errors). The nominal values obtained are:

$$\widehat{PERS} = 3, \text{ if CO prior to 1973}$$
$$= 8, \text{ if CO in 1973 or after,}$$

and the chi-square value for goodness of fit is 30.7 on 21 df.

iii. Other Types of LERs

Other LER subtotals, such as procedural errors, operations, maintenance, and environmental and health physics are too infrequent or too heterogeneous to warrant analysis.

c. Summary of Results

Table 7 summarizes the Z-scores calculated for PWR LERs.

Table 7

Summary of Licensee Performance Analysis: PWR LERs

Station	Personnel	Total LERs FH76	SH76	CY76
Yankee Rowe	A	B	B	B
San Onofre	A	A	A	A
Conn. Yankee	B	B	B	B
Ginna	B	B	A	B
Indian Point 2	B	A	B	A
Turkey Pt.	A	A	B	A
Palisades	C	B	B	B
Robinson	C	B	B	C
Pt. Beach	B	B	A	B
Oconee	C	C	C	C
Surry	C	B	C	C
Prairie Is.	C	C	C	C
Ft.Calhoun	B	B	B	B
Three M. Is.	B	C	B	B
Zion	B	B	B	B
Kewaunee	C	B	C	B
Maine Yankee	B	B	B	B
Rancho Seco	A	A	A	A
Arkansas	B	A	B	B
Cook	C	B	B	C
Calvert Cliffs	A	A	A	A
Millstone 2	A	A	B	A
Trojan	B	C	B	C

2. BWRs

a. FH76 and SH76

i. Total LERs

As with PWRs, LER frequency shows a dichotomy related to age. However, the split occurs at a different time and is more pronounced than with PWRs and is also not quite so consistent over the two six month periods. The nominal values, on which Z-scores for BWR LERs will be based are

FH76: $\widehat{LER}$ = 3, if CO prior to 1974,
= 14, if CO in 1974 or after.

SH76: $\widehat{LER}$ = 3, if CO prior to 1974,
= 10, if CO in 1974 or after.

After excluding Dresden in both periods and Cooper in FH76, the chi-square goodness of fit values for this relationship show an adequate fit (χ = 16.2 on 13 df for FH76 and χ^2 = 12.8 on 14 df for SH76).

b. CY76

i. Total LERs

The results of the preceding section lead to CY76 nominal LER frequencies as follows:

$\widehat{LER}$ = 6, of CO prior to 1974
= 24, if CO in 1974 or after

With the exception of one station in each group, Dresden and Cooper, the fit is quite adequate (χ^2 = 11.3 on 13 df), so Z-scores will be based on this relationship.

ii. Personnel Errors

LERs involving personnel errors (about 60% of all LERs) follow the same pattern in total LERs. The nominal values obtained from the data are:

$\widehat{PERS}$ = 4, if CO prior to 1974,
= 20, if CO in 1974 or after.

The chi-square value for this relationship is small enough (χ^2 = 9.7 on 13 df after omitting Dresden and Cooper) to indicate an adequate fit.

Table 8
BWR LER Frequencies

Station	Period	Personnel Error			Procedural Error		
		Operations	Maint.	Env. & HP	Operations	Maint.	Env. & HP
Dresden	FH76		6	1		2	
	SH76	1	5	1		2	
	CY76	1	11	2		4	
Humboldt Bay	FH76	2				2	
	SH76						
	CY76	2				2	
Big Rock Pt.	FH76	1				1	
	SH76	1	2			3	1
	CY76	2	2			4	1
Oyster Cr.	FH76	1	1				
	SH76	1				1	
	CY76	2	1			1	
Nine M. Pt.	FH76	1	2				
	SH76			1		2	
	CY76	1	2	1		2	
Millstone 1	FH76		2		1		1
	SH76	1			2		
	CY76	1	2		3		1
Quad Cities	FH76		1	2		1	
	SH76	2	1			1	
	CY76	2	2	2		2	
Monticello	FH76						
	SH76	1	2				
	CY76	1	2				
Vt. Yankee	FH76	2	1			1	
	SH76	1	1				
	CY76	3	2			1	
Peach Bottom	FH76	5	2	2		1	
	SH76	2	4	4	2		
	CY	7	6	6	2	1	
Pilgrim	FH76	3			1		
	SH76					1	
	CY76	3			1	1	

Appendix A

Table 8 (cont'd)

BWR LER Frequencies

Station	Period	Personnel Error Operations	Personnel Error Maint.	Personnel Error Env. & HP	Procedural Error Operations	Procedural Error Maint.	Procedural Error Env. & HP
Cooper	FH76	1					
	SH76	1	6			1	
	CY76	2	6			1	
Hatch	FH76	8	9		3	1	
	SH76	6	3	1			
	CY76	14	12	1	3	1	
Brunswick	FH76	9	5		4	1	
	SH76	4	4	1		1	
	CY76	13	9	1	4	2	
Duane Arnold	FH76	3	4	1	4		
	SH76	2	6	3			
	CY76	5	10	4	4		
Fitzpatrick	FH76	6	4				
	SH76	1		1	1	2	2
	CY76	7	4	1	1	2	2
Lacrosse	FH76						
	SH76	1	2		1		
	CY76	1	2		1		

c. Summary of Results

Table 9 summarizes, by A, B, and C, and Z-scores for BWR LERs determined from the analyses of the previous sections.

Table 9

Summary of Licensee Performance Analysis: BWR LERs

Station	Personnel	Total LERs FH76	Total LERs SH76	Total LERs CY76
Dresden	C	C	C	C
Humboldt Bay	A	B	A	B
Big Rock Pt.	B	B	C	C
Oyster Cr.	B	B	B	B
Nine M. Pt. 1	B	B	B	B
Millstone 1	B	B	B	B
Quad Cities	B	B	B	B
Monticello	B	A	B	A
Vt. Yankee	B	B	B	B
Peach Bottom	B	A	B	B
Pilgrim	B	B	A	B
Cooper	A	A	B	A
Hatch	C	C	B	C
Brunswick 2	B	C	B	B
Duane Arnold	B	B	B	B
Fitzpatrick	A	A	A	A
Lacrosse	B	A	B	B

D. Overall Performance Measures

In order to obtain an overall performance measure, the individual performance measure Z-scores were combined in the following way. Infraction and deficiency Z-scores were combined by weights in a ratio of 5:1 in order to obtain an overall score for noncompliances. This score was then combined in a ratio of 3:1 with the Z-score for LERs. At each stage when a weighted sum is calculated, the sum is normalized by dividing by the square root of the sum of squares of weights. Appendix B describes a sensitivity study of the choice of weights. The overall performance measures are tabulated in Chapter III.

Appendix B

Sensitivity Analysis of Weighting Factors

The overall licensee performance measure, expressed as a Z-socre, is obtained by taking a linear combination of the Z-scores for infractions, deficiencies, and LERs. Table 1 shows the weights used in this linear combination and the products of the weights, which is the resulting effect on the overall Z-score. Note that the weights are scaled so that their sum of squares equals 1.0. This is so that the overall Z-score will be comparable to the standard normal distribution. Thus, from Table 1, an increase in Z_{INF} by 1.0 results in the overall Z increasing by 0.93; Z_{DEF} increasing by 1.0 results in a 0.19 increase in overall Z; and the effect of Z_{LER} on overall Z is 0.32.

Table 1

Weights on Z-Scores and Resulting Effect on Overall Z-Score

Performance Measure	Wt 1	Wt 2	Wt 1 x Wt 2
Infractions (Z_{INF})	$5/\sqrt{26}$	$3/\sqrt{10}$	0.93
Deficiencies (Z_{DEF})	$1/\sqrt{26}$		0.19
LERs (Z_{LER})		$1/\sqrt{10}$	0.32

The choice of weights is based on judgments as to the relative importance which should be attached to deviations from average performance in the three performance measures, and as such is to some extent arbitrary. For the sake of comparison, some alternative weights are illustrated in Table 2. The effect of any other choice of weights can be readily evaluated by following the format of Table 2.

The effects shown in Tables 1 and 2 are in terms of Z-scores. The effects in terms of the frequencies of noncompliances and LERs are obtained by recalling that Z-scores are essentially numbers of standard deviations above or below the nominal frequencies. For counting data, and the Poisson distribution, the standard deviation is the square root of the nominal. As described in Appendix A, nominal noncompliance and LER frequencies are not constants, but rather functions of such variables as inspection hours, age, and Regional Office. However, some rough averages can be used to get an idea of the sensitivity of overall Z to noncompliances and LERs.

Infraction frequencies, for both PWRs and BWRs averaged about 1.3 infractions per 100 inspection hours. Average inspection hours for CY76 was about 1200 hours, so the average nominal frequency of infractions is about 16. Thus, the standard deviation is about 4. From Table 1, then, it is seen that a decrease of one infraction will result in an increase in overall Z of 0.93/4 = 0.23. Thus, it would take a swing of at least 5 infractions to move an overall Z-score from an A to a C, for example.

Table 2

Alternative Weights on Z-Scores and Resulting Effects on Overall Z-Score

	Performance Measure	Wt. 1	Wt 2	Wt 1 x Wt 2
A	Z_{INF}	$2/\sqrt{5}$	$3/\sqrt{10}$	0.85
	Z_{DEF}	$1/\sqrt{5}$		0.42
	Z_{LER}		$1/\sqrt{10}$	0.32
B	Z_{INF}	$8/\sqrt{65}$	$3/\sqrt{10}$	0.94
	Z_{DEF}	$1/\sqrt{65}$		0.12
	Z_{LER}		$1/\sqrt{10}$	0.32
C	Z_{INF}	$2/\sqrt{5}$	$1/\sqrt{2}$	0.63
	Z_{DEF}	$1/\sqrt{5}$		0.32
	Z_{LER}		$1/\sqrt{2}$	0.71
D	Z_{INF}	$5/\sqrt{26}$	$1/\sqrt{2}$	0.69
	Z_{DEF}	$1/\sqrt{26}$		0.14
	Z_{LER}		$1/\sqrt{2}$	0.71
E	Z_{INF}	$8/\sqrt{65}$	$1/\sqrt{2}$	0.70
	Z_{DEF}	$1/\sqrt{65}$		0.69
	Z_{LER}		$1/\sqrt{2}$	0.71

Nominal deficiency frequencies varied over Regions and reactor types, but averaged about 9 per station over the year. Thus, the effect of a decrease of one deficiency is to increase overall Z by $0.19/3 = 0.06$. Only in marginal cases could a swing of a few deficiencies result in the overall Z-score moving from one category to another.

The average number of LERs varies considerably, but a rough overall average of 6 LERs is adequate for use in evaluating sensitivity. For this average, a decrease of one LER results in an increase in overall Z of $0.32/\sqrt{6} = 0.13$. Thus, an LER has about twice the effect of a deficiency and half that of an infraction. These relative weights are felt to be reasonable and appropriate and for these reasons were used.

Copy__________

INDIVIDUAL SITE RATINGS

From The

IE EMPLOYEE SURVEY ON EVALUATION OF LICENSEES

April 1978

Stephen K. Conver

IE Study Group

Office of Inspection and Enforcement

U. S. Nuclear Regulatory Commission

INDIVIDUAL SITE RATINGS

From The

IE EMPLOYEE SURVEY ON EVALUATION OF LICENSEES

Background

This report documents the "Individual Site Rating" portion of the "IE Employee Survey on Evaluation of Licensees" that was conducted in the fall of 1977. The purpose of this survey was to solicit the views of employees of the Office of Inspection and Enforcement (IE) on a variety of subjects related to Licensee Performance Evaluation (LPE). For several years, IE has been attempting to develop a method of identifying those licensees whose level of performance (as measured principally, but not solely, by compliance) requires improvement.

A persistent IE staff criticism of early in-house efforts to develop an LPE methodology was that proposed quantitative rating schemes did not capture the subjective judgments of those Regional employees familiar with the specific licensed activities. This questionnaire was developed as one way of responding to that valid criticism. In addition to asking a number of questions on the advisability and mechanics of conducting evaluations of licensees, the questionnaire also asked each Regional respondent to evaluate each of the sites he was familiar with in terms of its overall safety and a number of other factors. This report summarizes the results of those ratings.

A survey instrument was prepared and statistical calculations were performed by Hay Associates under NRC Purchase Orders DR-77-1322 and DR-77-2631. After the questionnaire was developed with significant input from the IE staff, it was distributed by IE to all appropriate staff members directly associated with the inspection of operating power reactors.

including both Headquarters and Regional employees. To encourage candor and to comply with the Privacy Act, respondents were asked not to sign their names to the questionnaire and all responses were mailed directly to and compiled by Hay Associates.

Use of Site Rating Information

The views solicited in this questionnaire constitute predecisional opinions that are intended primarily to contribute to the development of a methodology for evaluating NRC licensees. For this and several other reasons, the site rating information may not be appropriate for release to the public. Although the information is untested, unvalidated, not directly related to licensee compliance with NRC requirements, and unreviewed by licensees, it may be of some use to IE management in gaining insights into the perceived safety at the 45 operating power reactor sites licensed by NRC. Some of the information may provide additional insights that will help identify inspection program improvements or form the basis for management conferences with licensees. For these latter purposes, the information should be used with some discretion and with an awareness of its limitations noted above.

The results of the site rating information are presented in summary form to adhere to the requirements of the Privacy Act by preventing the specific responses of any single individuals to be identified.

Survey Procedures

The questionnaire was distributed to all employees of IE associated with the inspection of operating power reactors. In rating specific sites, Regional respondents were asked to "rate sites you feel you know enough about to evaluate. Some of your knowledge of that site should have been gathered since January 1976."

Respondents were asked to rate each site they were familiar with by filling out a two-page section of questions (see Figures 1a and 1b on the following pages). The first eleven questions for each site asked the respondent to assess the safety of the site in terms of its overall safety (question 1) and in terms of ten additional factors (questions 2 - 11). For each of these eleven ratings, the respondents were asked to "draw a line indicating how safe you think this site is." A scale labeled "SAFETY" was provided for this purpose. The endpoints of this safety scale were labeled "ACCEPTABLE" and "EXCEPTIONAL." The following definitions were provided:

1. Safety - the degree to which the licensee protects the public against exposure to radiation resulting from the licensee's use of nuclear materials.

2. Acceptable - barely safe enough to be permitted to continue operating.

3. Exceptional - having a virtual absence of risk.

The "acceptable" endpoint was so labeled because all plants currently permitted to operate by NRC were presumed to be at least marginally satisfactory. In the event a respondent considered a plant less than "acceptable," a space for narrative comments about the safety of each site was provided (question 18).

Respondents were asked to describe their own level of knowledge of the site, identify whether they were the Principal Inspector for the site, and indicate how recently they had inspected the site (questions 12 - 14). Another question (15) asked for a comparison of the site's requirements

Figure 1a: Sample Rating Page 1

OPERATING SITE:

NAME: ______________________________

DOCKET NO.: ______________________________

Considering all you know about this site, what overall general safety rating would you give to it? Draw a line indicating how safe you think this site is.

	SAFETY	
	ACCEPTABLE	EXCEPTIONAL
1. Overall safety	|.	

Draw a line indicating how safe you feel this site is in terms of the following factors:

	SAFETY	
	ACCEPTABLE	EXCEPTIONAL
General attitude of plant personnel toward:		
2. Maintenance of safety	|.	
3. Cooperation with NRC	|.	
4. Technical competence of plant personnel	|.	
5. Quality of design, construction, components	|.	
6. Administrative controls	|.	
7. Operations	|.	
8. Emergency planning	|.	
9. Radiation protection and control	|.	
10. Safeguards	|.	
11. Quality assurance	|.	

Figure 1b: Sample Rating Page 2

12. How well do you know this site and its safety characteristics:

HARDLY AT ALL						EXTREMELY WELL
1	2	3	4	5	6	7

13. Are you the Principal Inspector for a reactor on this site?

Yes ☐
No ☐

14. About how many months ago did you last inspect this site? ________ months ago.

15. The NRC requirements that this site must follow are:

MUCH LESS DEMANDING THAN THOSE OF OTHER SITES						MUCH MORE DEMANDING THAN THOSE OF OTHER SITES
1	2	3	4	5	6	7

16. Have there been any changes in the overall safety of this site since January 1977, that have caused its safety level to change? (Check one)

1. ______ No change in safety at site
2. ______ Safety slightly improved
3. ______ Safety substantially improved
4. ______ Safety slightly worse
5. ______ Safety substantially worse
6. ______ Don't know

17. If a change in safety level occurred, please describe it briefly.

____

____

____

____

18. Are there other things we should consider about the safety of this site?

Yes ☐
No ☐

If yes, please explain: ____

____

____

____

If this is the last site you are rating, please turn to page 54 and complete the questionnaire.

with those of other sites in terms of being more of less demanding. Finally, respondents were asked to indicate whether site safety had changed since January 1977, to describe how it had changed, and to add any other comments relevant to the safety of the site (questions 16 - 18).

In addition to rating each site they were familiar with, all respondents were asked to assess the safety of a fictitious site that was defined as the average of the 45 operating nuclear reactor sites in the U.S. This section was included to provide a means of calibrating the responses of employees from different Regional Offices.

Computational Procedures

All site safety ratings were converted to digital scores by manual measurement and recording of the responses to the "draw a line" questions. The ratings of the "typical site" were similarly reduced to digital form. Mean rating scores for each site were then calculated. Comparability was a major concern in comparing ratings of sites in various Regions, because people in one Region were generally unfamiliar with and did not rate sites in Regions other than their own. To compensate for potential differences in ratings between Regions, each person's rating of each site was raised or lowered based upon how his rating of the "typical site" compared to the average of all respondents' rating of the "typical site." This linear "rubber band" transform makes the ratings of site safety comparable across all raters and Regions. These adjusted site ratings were reconverted to a graphical format for display.

Averages for the numerical responses to other site rating questions were also calculated, and responses to the narrative site rating questions were paraphrased.

Results

Ratings of the "typical site," shown in Figure 2, illustrate the format used to present site ratings. The top of the rating sheet depicts the safety ratings of the site in terms of overall safety and the other ten safety factors, all shown on a scale of "acceptable" to "exceptional." The squares shown on the scale for each factor represent the mean rating, and the two circles on each scale represent the high and low ratings for each factor. As shown in Figure 2, the typical site is rated somewhat more than halfway between acceptable and exceptional, and ratings of the ten individual safety factors are in the same range. The perceived weakest areas, by a small margin, are Quality Assurance, Emergency Planning, and Administrative Controls. For the typical plant, the range of responses covers the entire scale for almost every factor. The ratings of the typical site reflect the judgments of 94 persons.

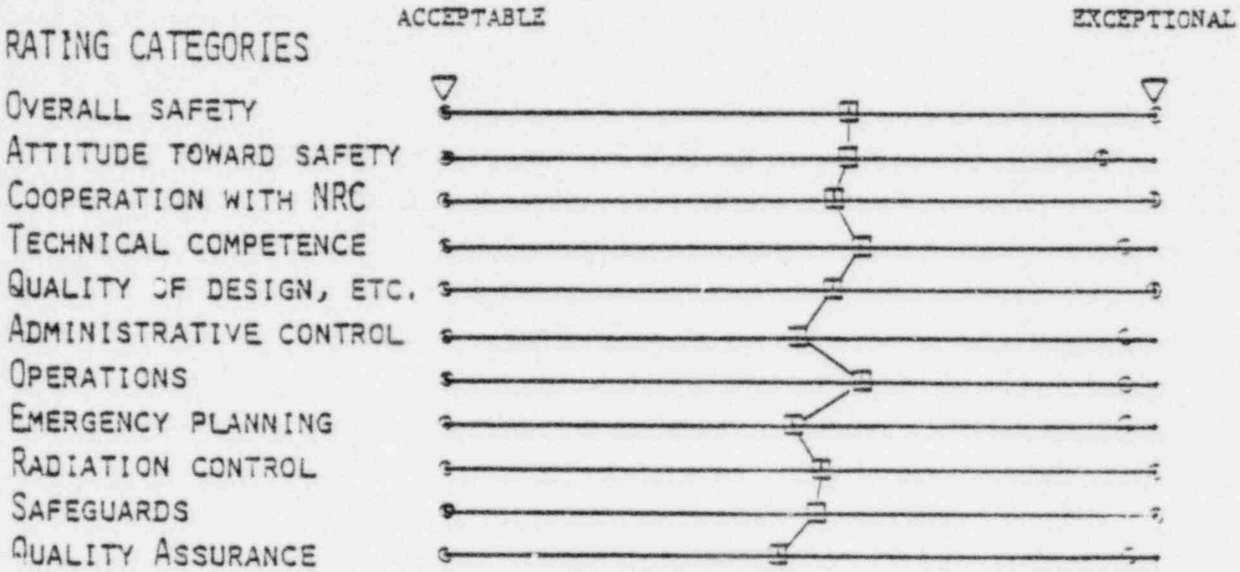
Because the typical site is fictitious, it did not receive ratings for the "familiarity of the raters with site," the "average number of months since raters' last inspection," or the "stringency of requirements for site."

Of the 94 persons rating the average site, 72 expressed opinions on the "change in site safety since January 1977." Most people felt that site safety had either improved slightly (39) or substantially (4), while about 40 percent (28 people) felt there was no change. Only one person thought that safety at the typical site had become worse since January 1977. There were no narrative comments solicited or offered about the safety of the typical site.

The means and standard deviations of the adjusted "overall safety" ratings are shown by Region in Figure 3. The mean adjusted safety rating for each Region is indicated by the square and the arrows represent the associated standard deviations. These results confirm that there are

SITE Typical Site in U.S.

Figure 2: Rating of the Typical Site DOCKET NUMBER 50-999



NUMBER OF PEOPLE RATING SITE = 94

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = N/A
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = N/A

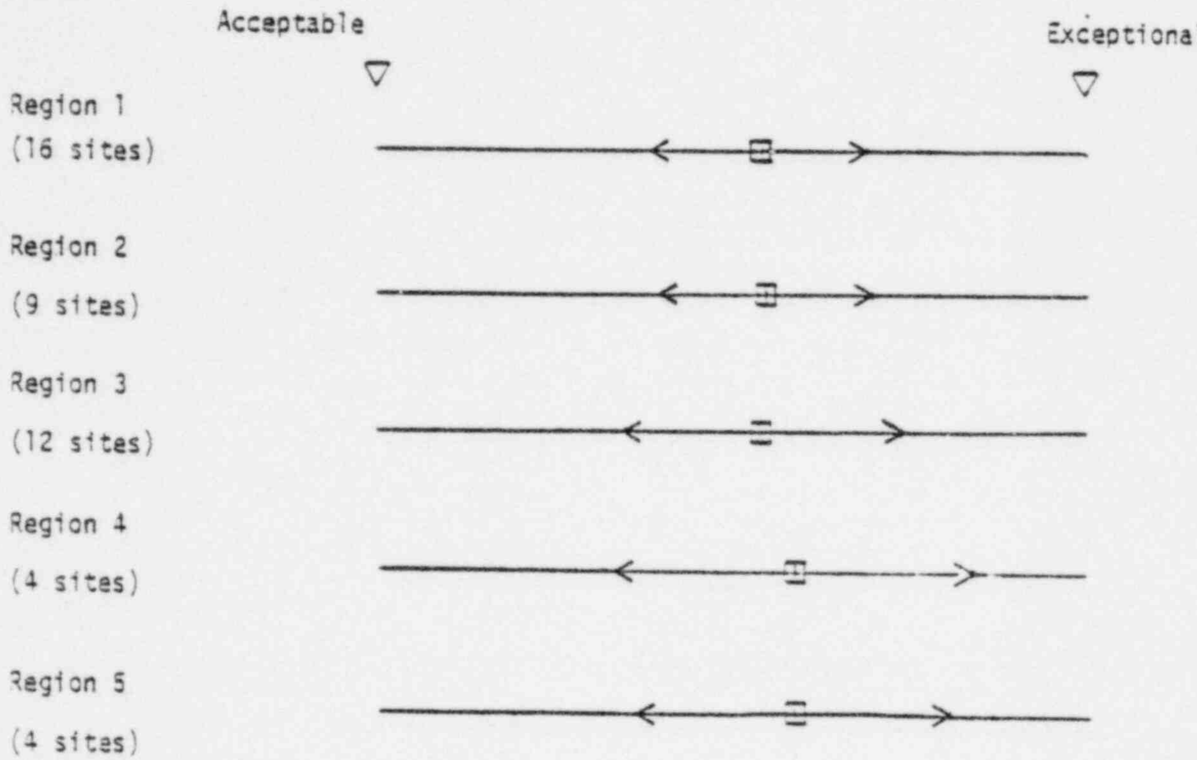
STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = N/A
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 28
2 = SAFETY SLIGHTLY IMPROVED............. 39
3 = SAFETY SUBSTANTIALLY IMPROVED........ 4
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

FIGURE 3: AVERAGE SITE RATINGS BY REGION*



*Squares indicate regional means of adjusted "overall safety" ratings. Arrows represent standard deviations.

no substantial differences in the average ratings between Regions after each individual's ratings are adjusted to account for his assessment of the typical site. The means for the three large regions (1,2, and 3) are virtually the same. Those for the smaller Regions (4 and 5) are slightly greater as are the standard deviations.

Rating information for each of the 45 sites is provided as Appendix A separate page is devoted to each site. As noted earlier, the squares on each safety scale indicate the mean rating, and the circles indicate the range of responses. The narrative comments represents a paraphrasing of observations from various persons which are not necessarily consistent with each other or with the quantitative rating information at the top of the form.

This information may be useful not only for developing evaluation methodology, but also for providing insights into the perceived levels of site safety, specific strengths and weaknesses at each site, overall trends toward improvement or degradation of performance, and possible improvements in inspection strategies.

APPENDIX A

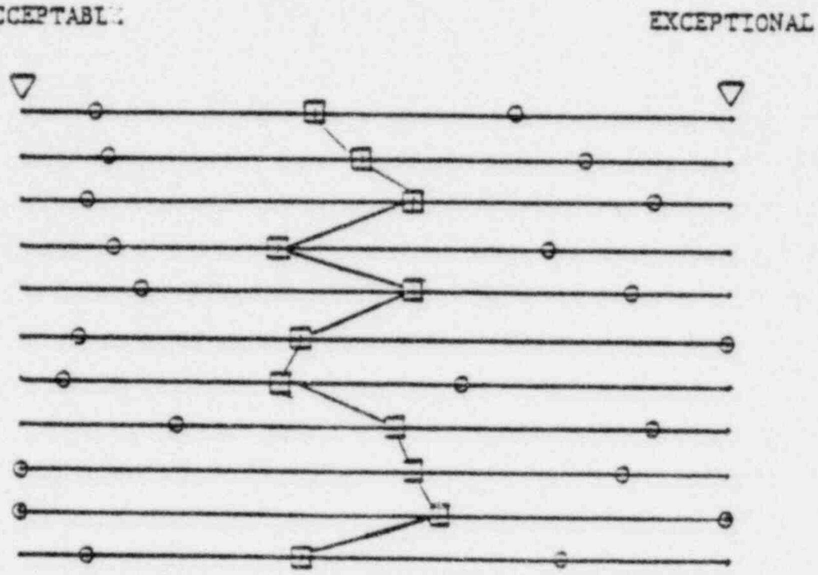
INDIVIDUAL SITE RATINGS

SITE Beaver Valley

DOCKET NUMBER 50-334

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 13

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.3
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 5.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 5.5
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 2
2 = SAFETY SLIGHTLY IMPROVED............. 6
3 = SAFETY SUBSTANTIALLY IMPROVED........ 1
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Staff is experienced. QA controls improved. Staff is improving. Bugs are being worked out of equipment and administrative controls. Plant management has improved. Security has improved with increased requirements. Staff still learning.

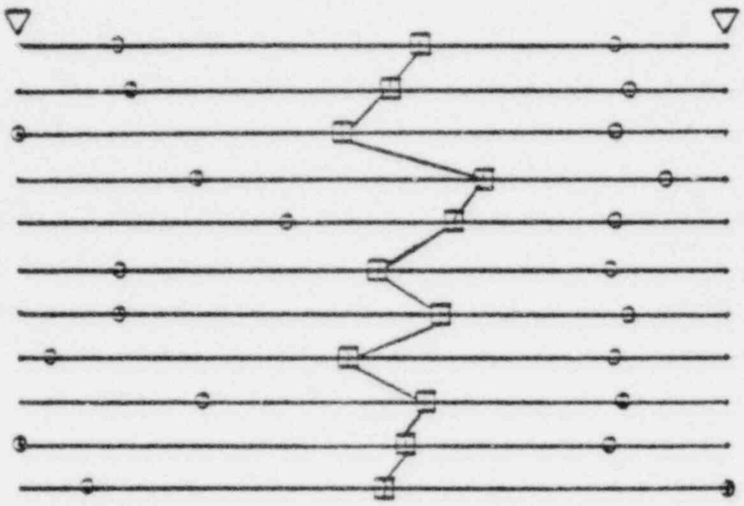
SITE Calvert Cliffs

DOCKET NUMBER 50-317

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 15

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.9
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 5.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 5.8
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 6
2 = SAFETY SLIGHTLY IMPROVED.............. 5
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................. 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

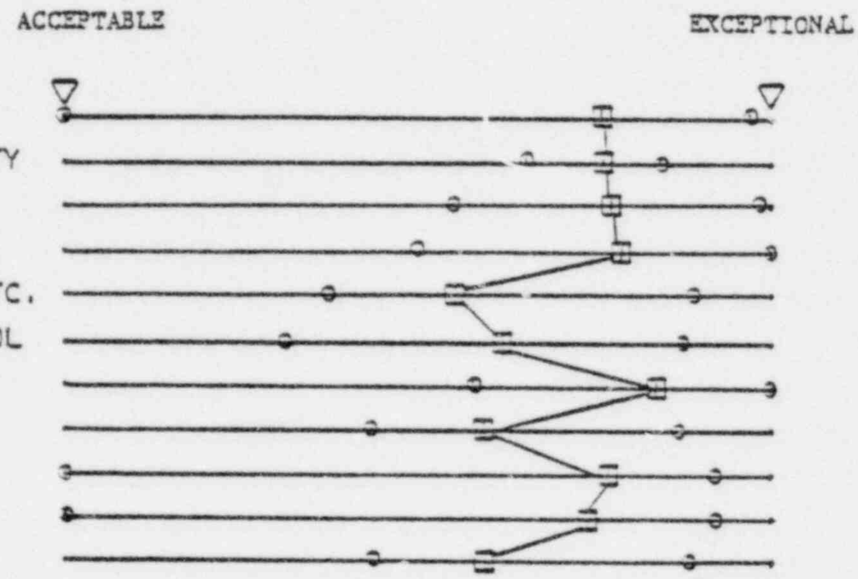
Management more attentive as a result of enforcement conference. An important staff member is anti-NRC and anti-QA. Security is improved. This site doesn't do more for safety than meet minimum requirements. Emphasis is upon commercial operation; attitude toward safety is that meeting NRC requirements literally is sufficient.

SITE Connecticut Yankee

DOCKET NUMBER 50-213

RATING CATEGORIES

- OVERALL SAFETY
- ATTITUDE TOWARD SAFETY
- COOPERATION WITH NRC
- TECHNICAL COMPETENCE
- QUALITY OF DESIGN, ETC.
- ADMINISTRATIVE CONTROL
- OPERATIONS
- EMERGENCY PLANNING
- RADIATION CONTROL
- SAFEGUARDS
- QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 9

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 7.6

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.7
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 7
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Overall safety should be improved at the complation of ongoing design requirement and license condition upgrading.

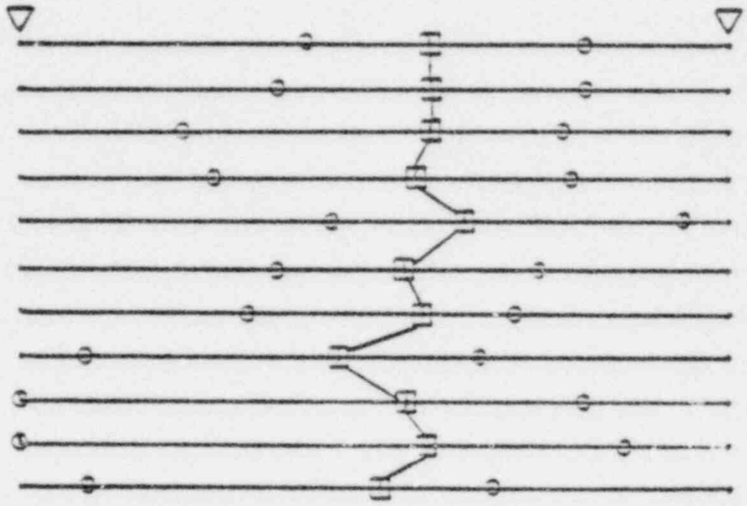
SITE Fitzpatrick

DOCKET NUMBER 50-333

ACCEPTABLE EXCEPTIONAL

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 11

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.0
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 5.5.

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.6
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................... 4
2 = SAFETY SLIGHTLY IMPROVED............. 6
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Plant has a new operator (PASNY) that appears to have made improvements. Is increased management attention to operations. New security procedures are in effect. New management has improved technical competence and management and administrative controls. Design has been modified to add safety systems. Excellent fire protection and security systems. Management improvements noted.

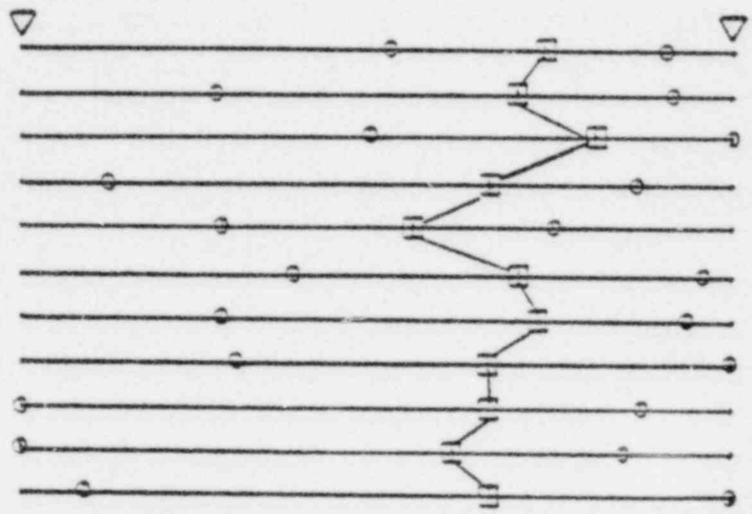
SITE Ginna

DOCKET NUMBER 50-244

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 11

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.4
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 8.5

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.4
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................... 9
2 = SAFETY SLIGHTLY IMPROVED............ 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

The plant is old, small, and run safely.

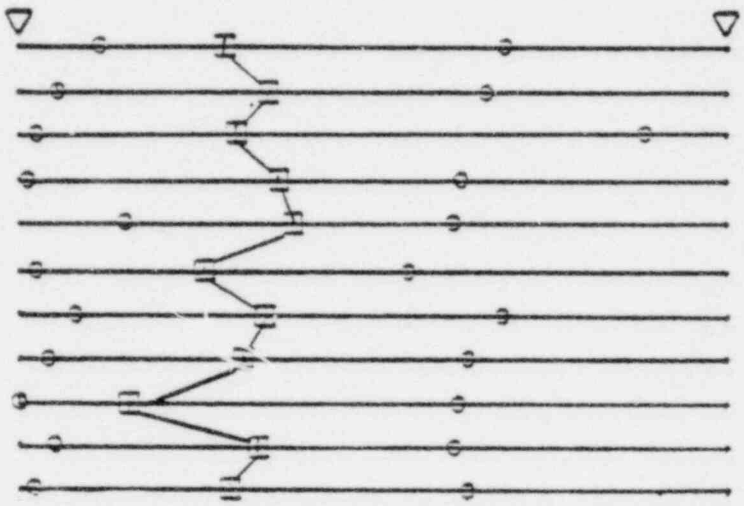
SITE Indian Point

DOCKET NUMBER 50-003

ACCEPTABLE EXCEPTIONAL

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 13

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.5
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 5.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.8
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

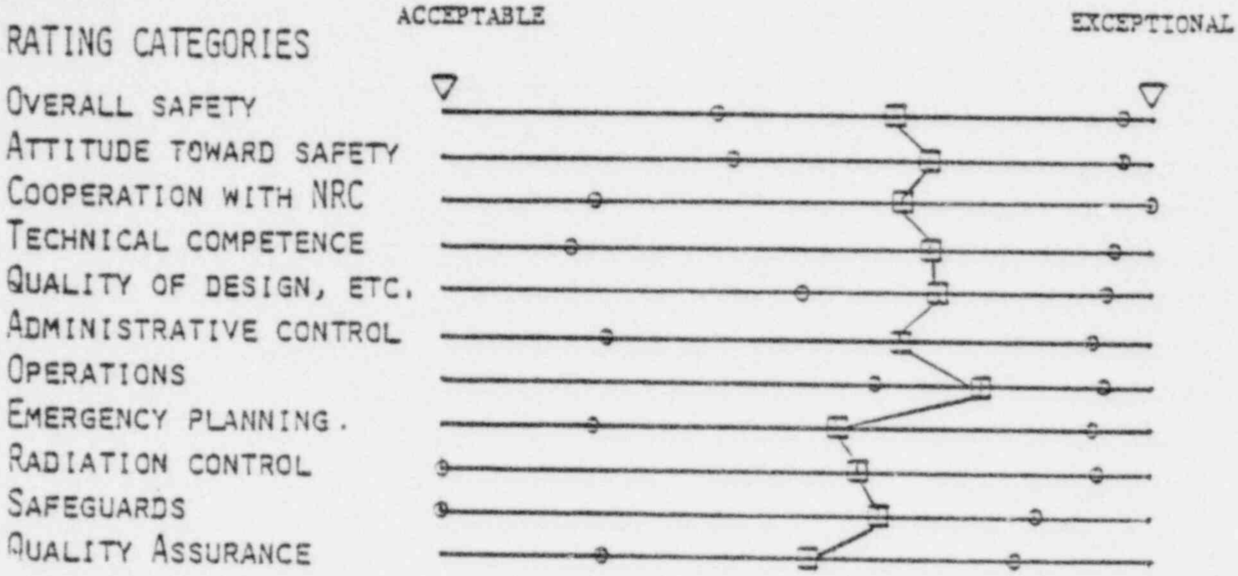
1 = NO CHANGE IN SAFETY.................. 4
2 = SAFETY SLIGHTLY IMPROVED............. 4
3 = SAFETY SUBSTANTIALLY IMPROVED........ 3
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Indian Point Unit 3 is superior in all respects to Unit 2, primarily because of its management controls and personnel. Considerable recent attention to HP, safeguards, and other areas of Unit 2 operations has resulted in considerable upgrading. Radiation health controls have improved. Recent problem with instrumentation. Does not have a QA plan meeting current requirements. Unit 3 rated higher than Unit 2 because PASNY management better than that of Con. Ed. Significant recent improvements in management control. Corporate management attitude continues to limit effectiveness of site management. Need to continue more frequent inspections by our best inspectors.

SITE Maine Yankee

DOCKET NUMBER 50-309



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.5
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.5
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 8
2 = SAFETY SLIGHTLY IMPROVED............ 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE............... 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

The cleanliness of this plant reflects a pride of ownership and indicates happy people working at a good plant. QA plan was recently upgraded.

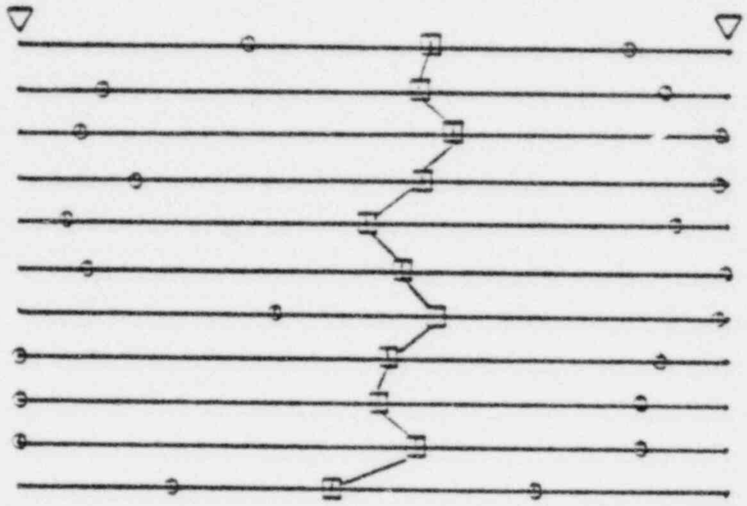
SITE Millstone

DOCKET NUMBER 50-245

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 20

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.0
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.8
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 9
2 = SAFETY SLIGHTLY IMPROVED............ 5
3 = SAFETY SUBSTANTIALLY IMPROVED....... 0
4 = SAFETY SLIGHTLY WORSE............... 0
5 = SAFETY SUBSTANTIALLY WORSE.......... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Unit 1, an old BWR, is rated lower than Unit 2. Awareness of safety has increased. The different units operate relatively independently, and each has a different vendor. Improved security arrangements. Plant lacks full separation and fire protection systems. Rad waste system undersized. New QA organization seems slightly better.

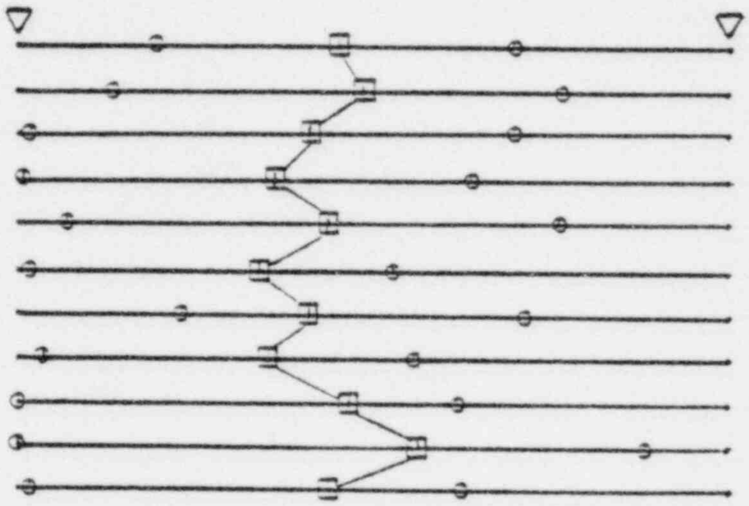
SITE Nine Mile Point

DOCKET NUMBER 50-220

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 13

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.5
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 10.0

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 11
2 = SAFETY SLIGHTLY IMPROVED............. 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Plant was operated by former fossil plant people; they have not yet become nuclear people. This is an old plant, but its engineering, layout, and construction are good. Do not have enough on-site plant support except in operations. Security program excellent. Plant deficient in system separation and high pressure inspection systems. Conservative approach to operations. Plant staff has been stable. Plant has experienced BWR operators.

SITE Oyster Creek

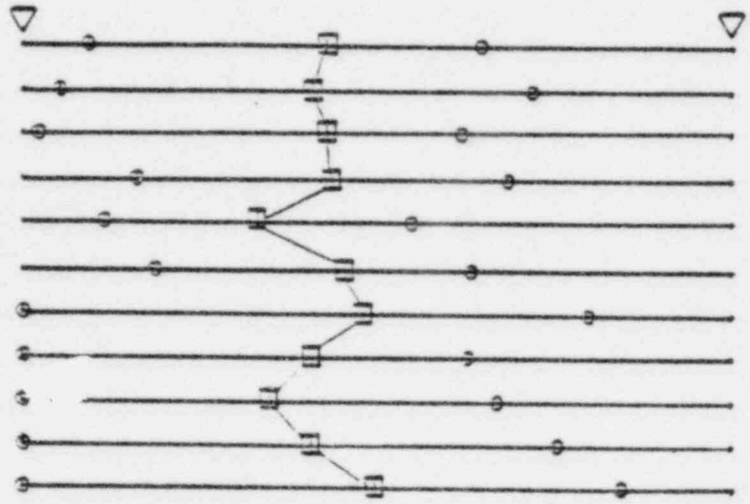
DOCKET NUMBER 50-219

RATING CATEGORIES

ACCEPTABLE

EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 14

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.5
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 10.0

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 6
2 = SAFETY SLIGHTLY IMPROVED............ 5
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE.............. 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Security should be upgraded (guard force and surveillance). New operating procedures and maintenance systems have improved safety. QA program has been more fully implemented. As an early BWR, plant has inherently different safety characteristics. Facility management has not endorsed in principle a comprehensive management control system. They tend to just meet the minimum requirements. Design review of this plant was deficient. Plant was built at minimum cost. Rad waste, fire protection, and system separation are inadequate. Corporate management has firsthand knowledge of plant.

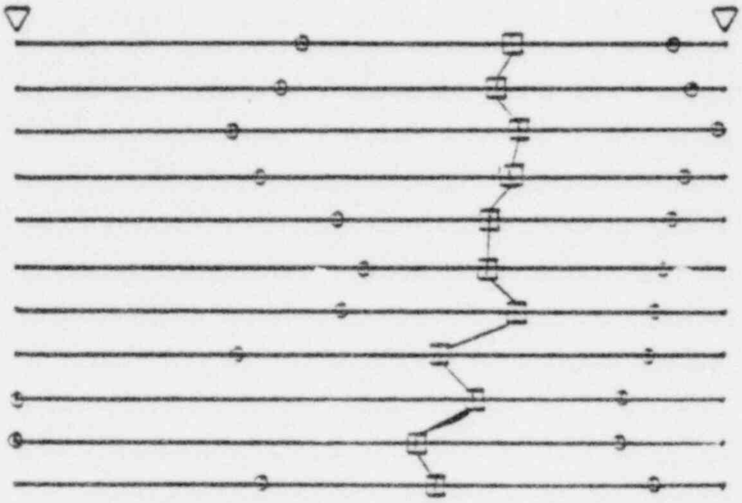
SITE Three Mile Island

DOCKET NUMBER 50-289

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 14

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.6

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.7
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 9
2 = SAFETY SLIGHTLY IMPROVED............ 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Station and unit superintendents are new. Security has improved. This is first B&W plant of current generation. Management control during construction was deficient. Management control: in operations is strong. Overall site safety may decrease because staff has become diluted with the licensing of Unit 2.

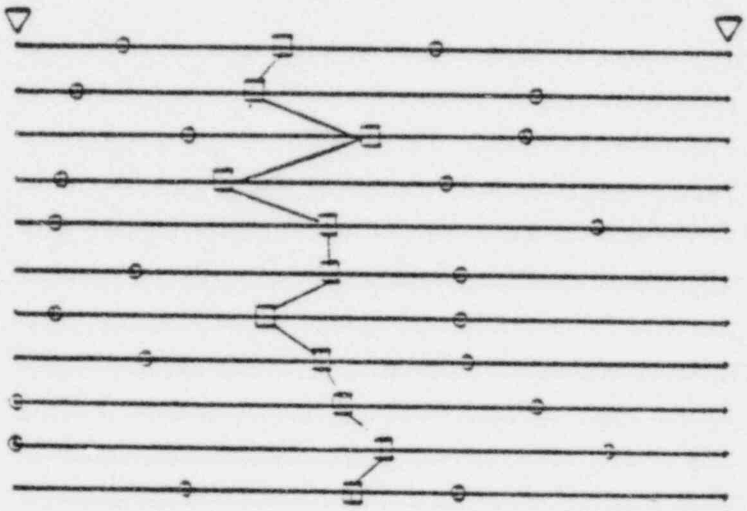
SITE Salem

DOCKET NUMBER 50-272

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.5
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 4.9

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 5.3
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................... 6
2 = SAFETY SLIGHTLY IMPROVED............... 1
3 = SAFETY SUBSTANTIALLY IMPROVED.......... 1
4 = SAFETY SLIGHTLY WORSE.................. 0
5 = SAFETY SUBSTANTIALLY WORSE............. 0

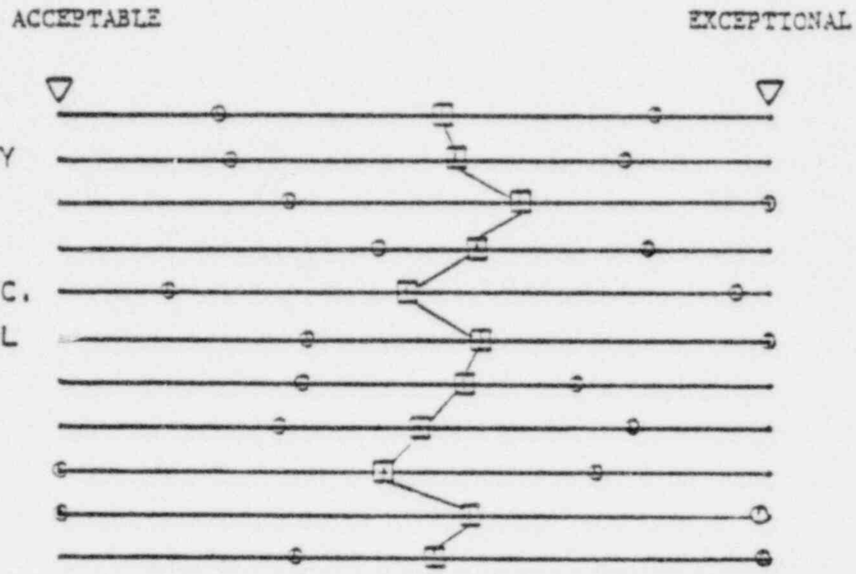
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

The plant control room is very poorly designed. This is a relatively new plant with growing pains. It needs close inspection attention to assure that appropriate improvements are made. Have had a number of problems in startup phase, which were corrected by management. Problems with operator controls.

SITE Pilgrim

DOCKET NUMBER 50-293

RATING CATEGORIES



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 13

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 8.6

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.2
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

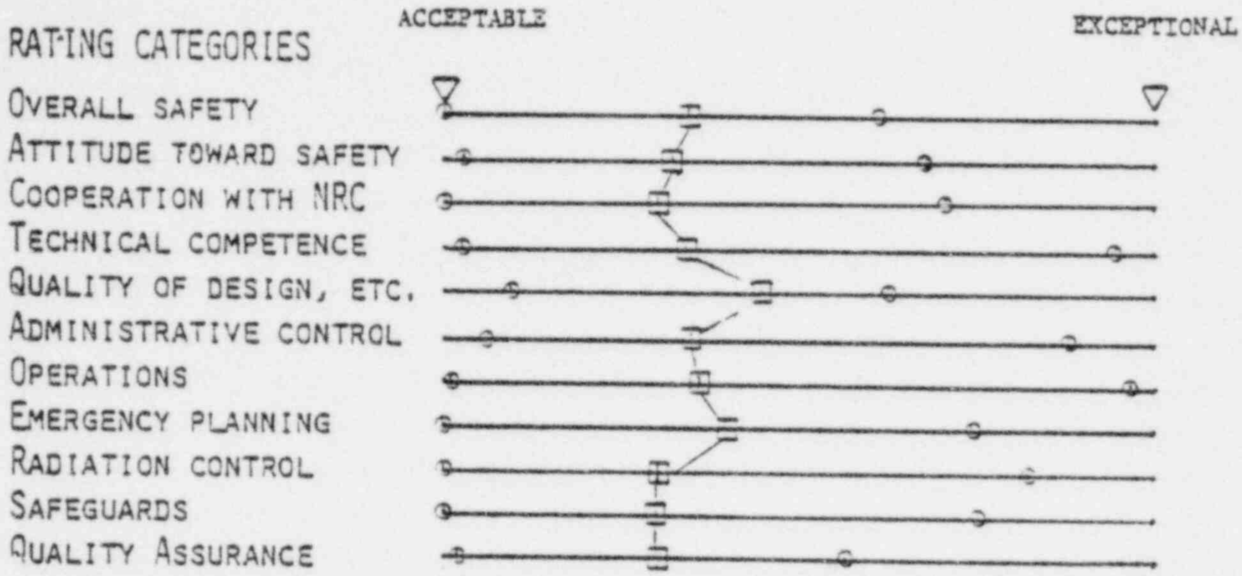
1 = NO CHANGE IN SAFETY.................. 7
2 = SAFETY SLIGHTLY IMPROVED.............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 1

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

The generation of its design may be an overriding factor for this early BWR. Corporate management improved. Radiation management improved. Frequent station manager changes. Significant reductions in effluents and worker exposures expected. Plant management has not been stable. This is the cleanest BWR in the country.

SITE Peach Bottom

DOCKET NUMBER 50-277



NUMBER OF PEOPLE RATING SITE = 19

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.0
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 5.5

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.3
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 8
2 = SAFETY SLIGHTLY IMPROVED............ 3
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE............... 3
5 = SAFETY SUBSTANTIALLY WORSE.......... 0

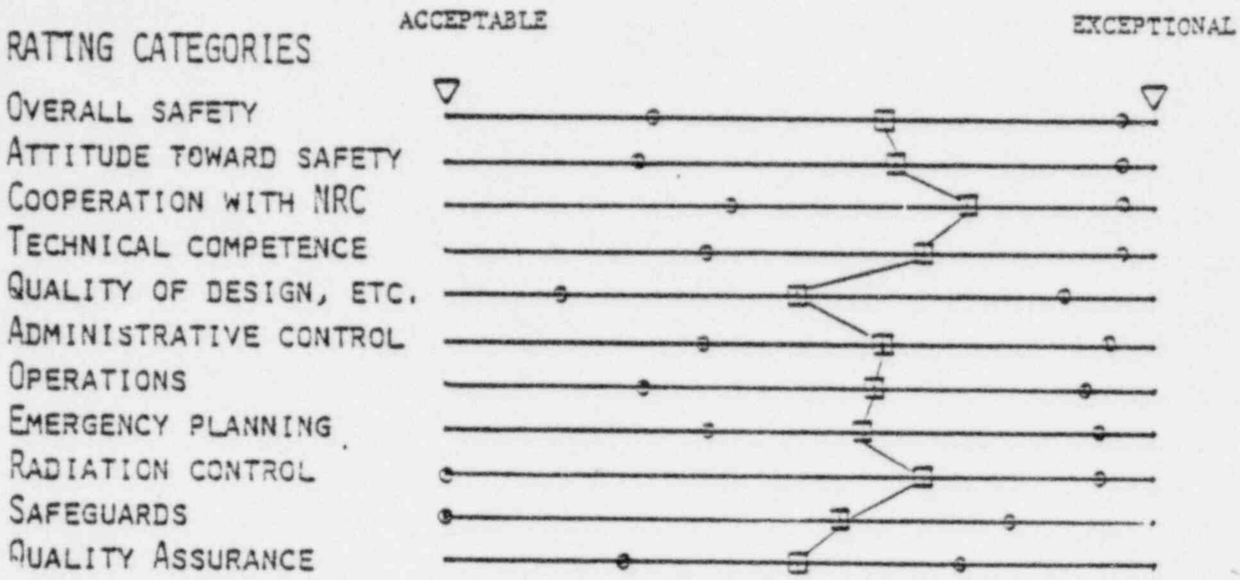
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

This is the least safe site in Region I and ha the poorest management. QA and securi are not upgraded to current standards. Many repe t items of noncompliance. Plant staff has appeared incapable of correcting increased plant radiation levels. Management is slow responding to problems. A greater inspection frequency is partially attributable to proximity to regional office. Expect improvements as a result of management meeting with company president. Operating staff presently error-prone due to back-to-back overhaul periods for Units 2 and 3. General attitude of plant appears to be compliance only as required. Careless operations and poor maintenance.

Over

SITE Vermont Yankee

DOCKET NUMBER 50-271



NUMBER OF PEOPLE RATING SITE = 12

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 10.2

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.8
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 9
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

QA plan has been upgraded. Management controls somewhat degraded by frequent changes in plant superintendent. Very clean plant. Management experience and depth is increasing.

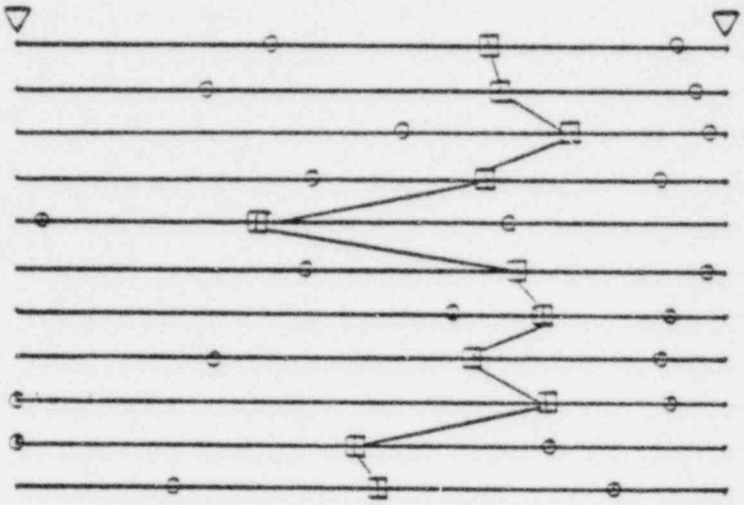
SITE Yankee Rowe

DOCKET NUMBER 50-029

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 10.1

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.4
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 6
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Plant is very small and very isolated. It presents virtually no health hazard to the public. Has old Tech Spec's. Upgraded QA program in 1977.

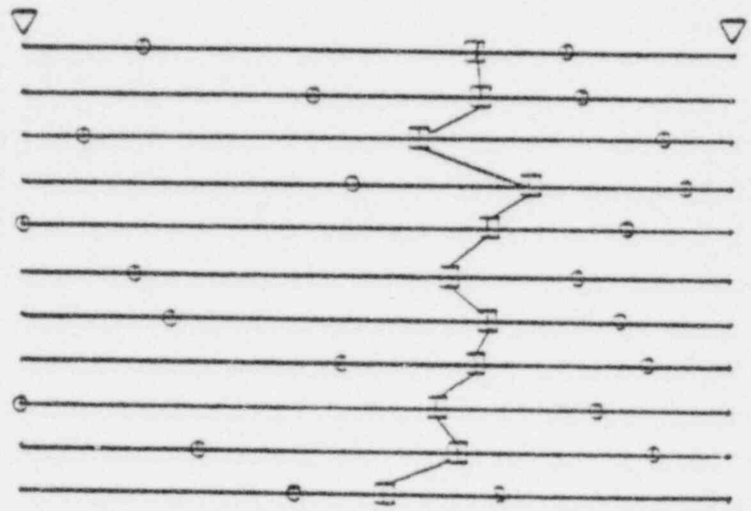
SITE Browns Ferry

DOCKET NUMBER 50-259

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.9
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.7
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 2
2 = SAFETY SLIGHTLY IMPROVED............. 4
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Attention to QA details has decreased slightly. Greater experience of plant personnel has contributed to improved safety and operations. More NRC inspections and plant management changes have also helped. Response to alarms has improved as a result of an enforcement meetings. Greater safety awareness. Fire protection improved.

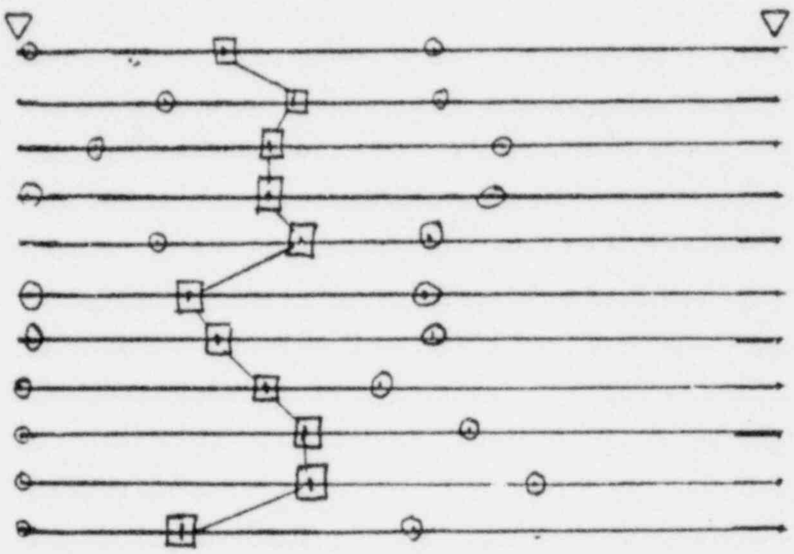
SITE Brunswick

DOCKET NUMBER 50-325

ACCEPTABLE EXCEPTIONAL

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.7
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.0

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 2
2 = SAFETY SLIGHTLY IMPROVED............. 4
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Site has reorganized and has new people in Key positions. Some improvement in adminstrative controls. Management seems to become more aware of events at plant. None of the top site management have had SRO training in BWR's. High personnel turnover rate. Plant management seems to believe that they are "over-regulated."

SITE Crystal River

DOCKET NUMBER 50-302

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL

OVERALL SAFETY

ATTITUDE TOWARD SAFETY

COOPERATION WITH NRC

TECHNICAL COMPETENCE

QUALITY OF DESIGN, ETC.

ADMINISTRATIVE CONTROL

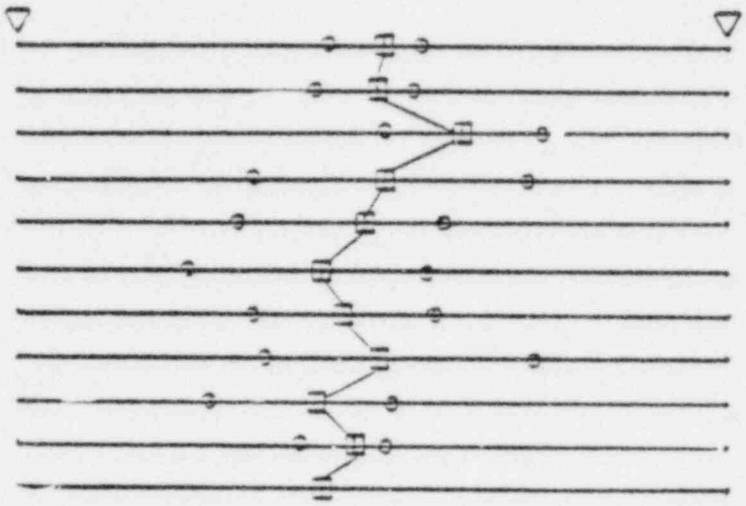
OPERATIONS

EMERGENCY PLANNING

RADIATION CONTROL

SAFEGUARDS

QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 3

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.0
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 5.0
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 1

2 = SAFETY SLIGHTLY IMPROVED............. 2

3 = SAFETY SUBSTANTIALLY IMPROVED........ 0

4 = SAFETY SLIGHTLY WORSE................ 0

5 = SAFETY SUBSTANTIALLY WORSE........... 0

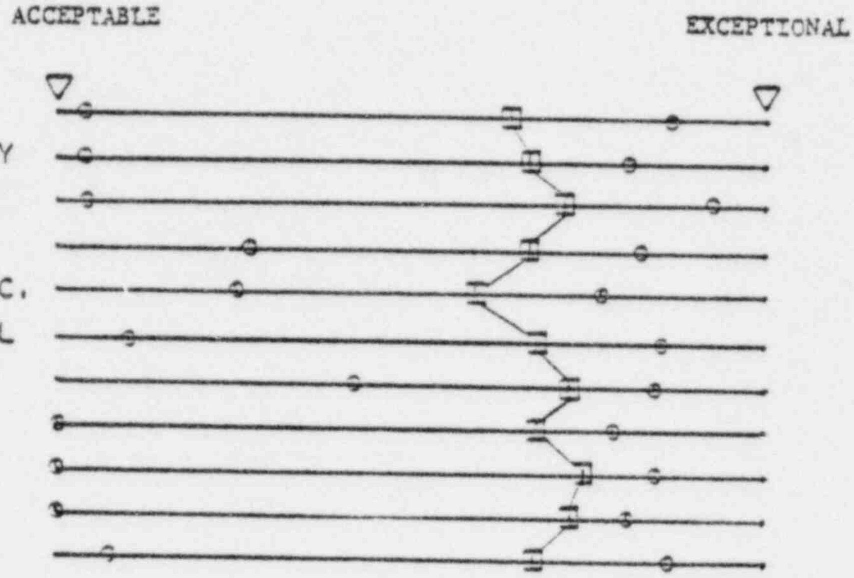
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety slightly improved because of more safety awareness. Operations and administrative controls improved.

SITE Hatch

DOCKET NUMBER 50-321

RATING CATEGORIES



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 9

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 4.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.5
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............. 4
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

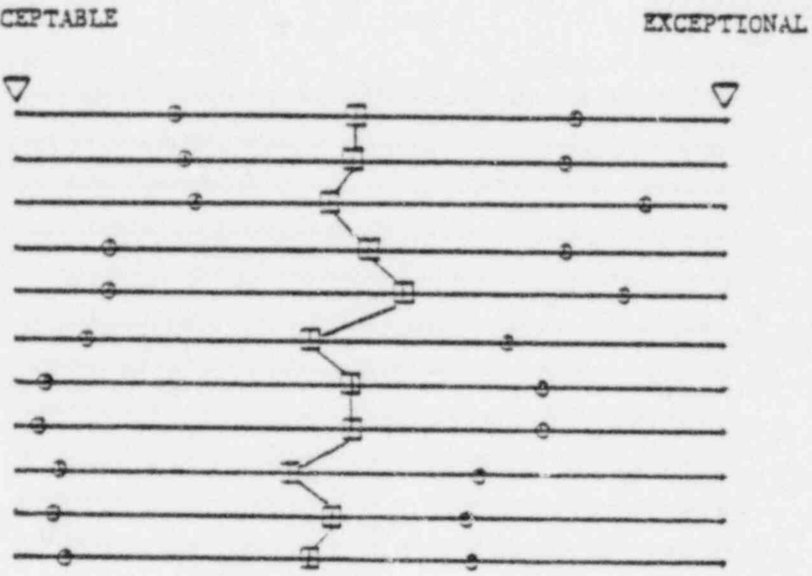
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Upgrading of administrative and QA controls is continuing.

SITE Oconee

DOCKET NUMBER 50-269

RATING CATEGORIES



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 6

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.3
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 17.0

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.0
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............ 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

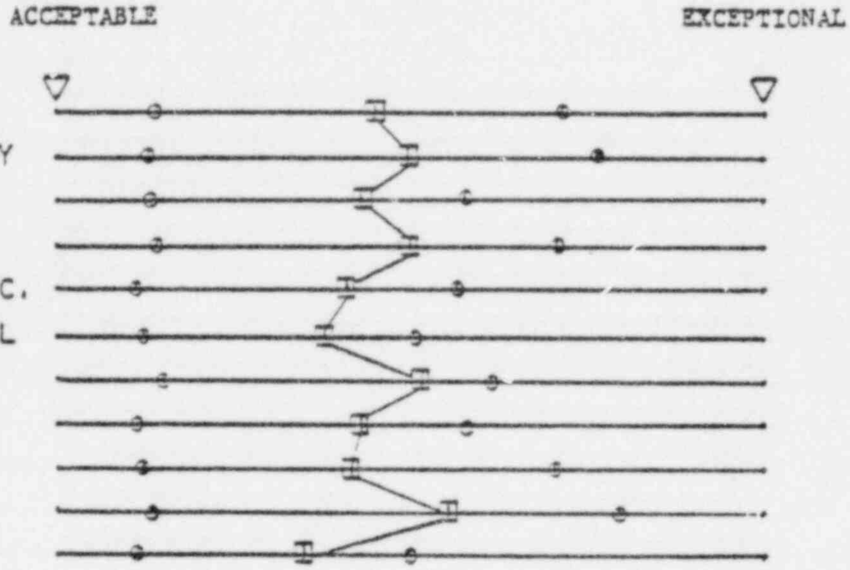
A change in the operating superintendent is expected to result in improvements.

SITE Robinson

DOCKET NUMBER 50-261

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 7

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.4
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.2

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.7
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 4
2 = SAFETY SLIGHTLY IMPROVED............ 2
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

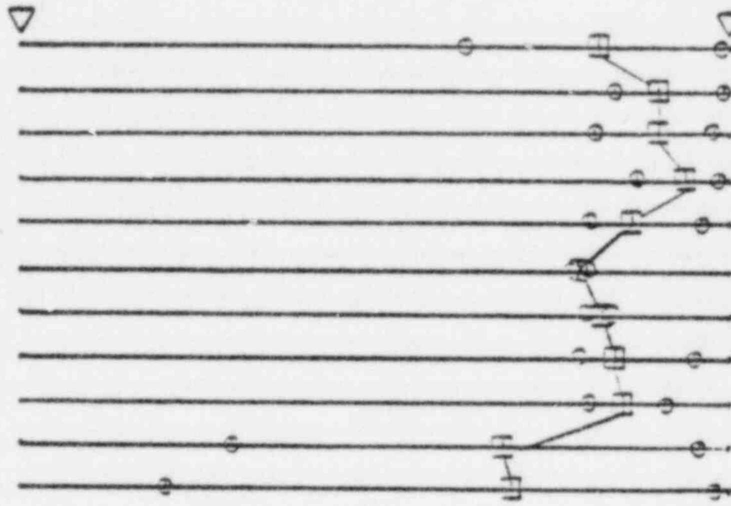
Licensee has made increased commitment to QA and QC. Licensee reports only those items that are conspicuously reportable. Licensee impedes inspector access and freedom of movement at site. No information freely given. Does only what is required.

SITE Saint Lucie

DOCKET NUMBER 50-335

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 3

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 6.3
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 2.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 6.0
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 1
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

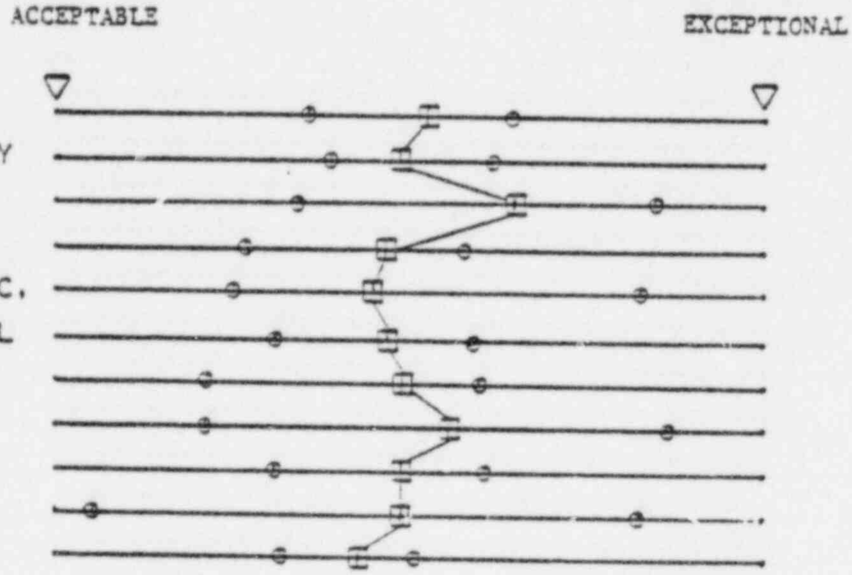
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety has improved due to increased experience of plant personnel. Plant's greater than average number of LERs is probably due to conscientiousness in reporting.

SITE Surry

DOCKET NUMBER 50-280

RATING CATEGORIES



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 6

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.2
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 14.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.0
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

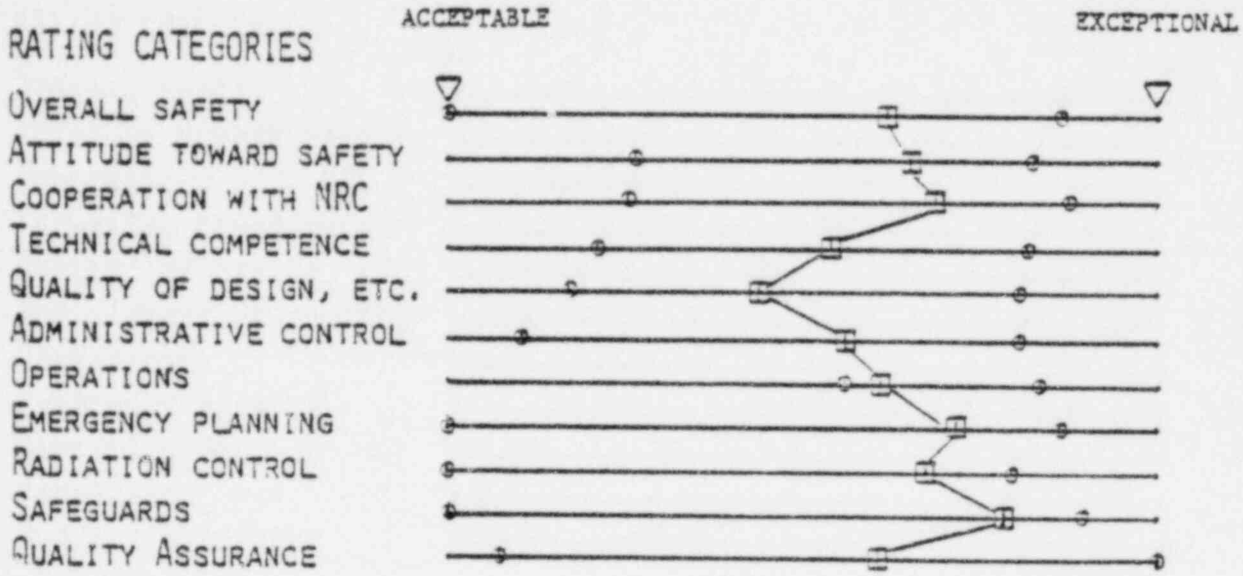
1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............. 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety slightly worse due to degradation of steam generator.

SITE Turkey Point

DOCKET NUMBER 50-250



NUMBER OF PEOPLE RATING SITE = 5

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.8
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 10.0

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.6
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

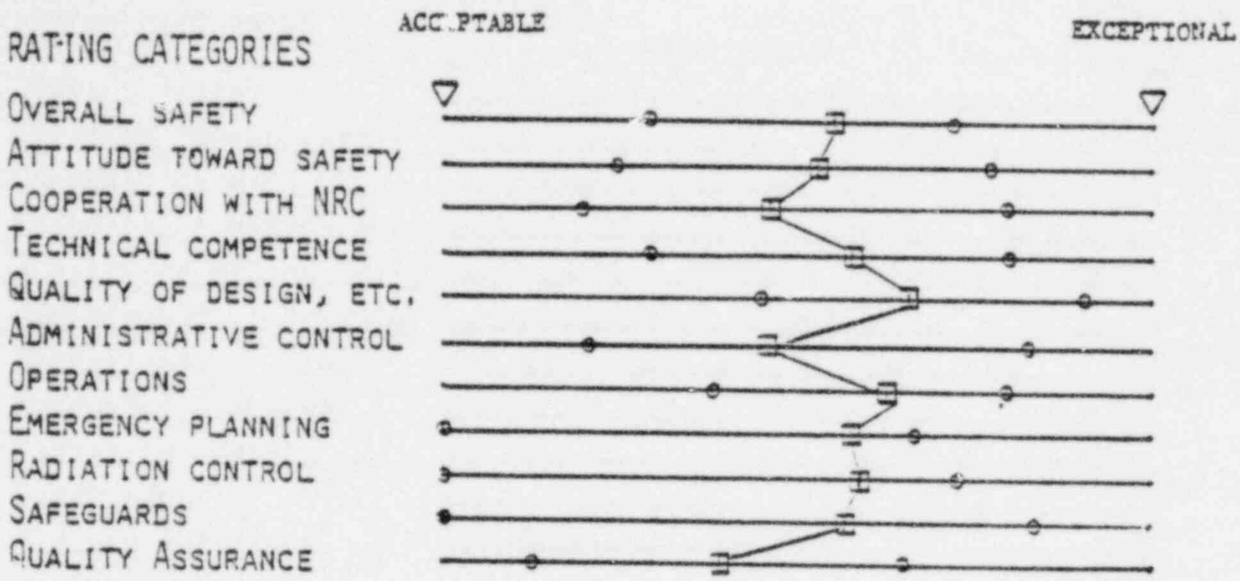
1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............. 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety may be slightly worse due to steam generator degradation.

SITE Arnold

DOCKET NUMBER 50-331



NUMBER OF PEOPLE RATING SITE = 9

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.3
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 10.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.6
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 0
2 = SAFETY SLIGHTLY IMPROVED............. 6
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety slightly improved due to improvements in QA and administrative controls, new plant superintendent, enforcement action, and increased inspection effort. Staff is more aware of significance of personnel error. Steady improvements in management controls, competence of staff, and attention from corporate office.

206

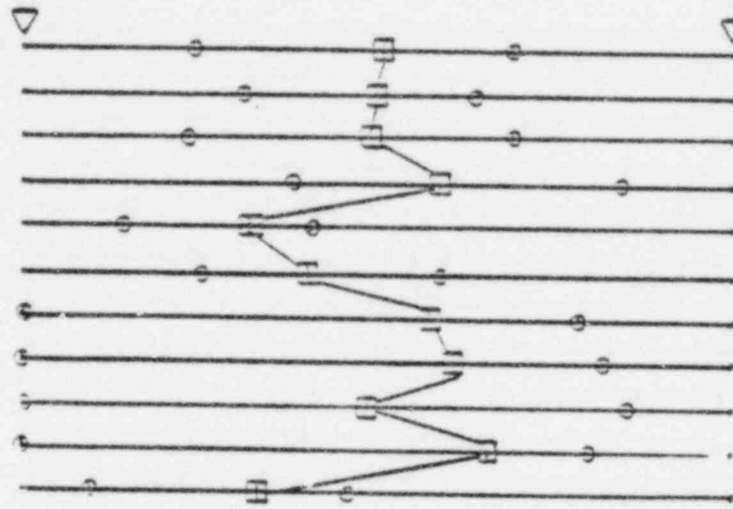
SITE Big Rock Point

DOCKET NUMBER 50-155

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 5

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.4
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 1
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

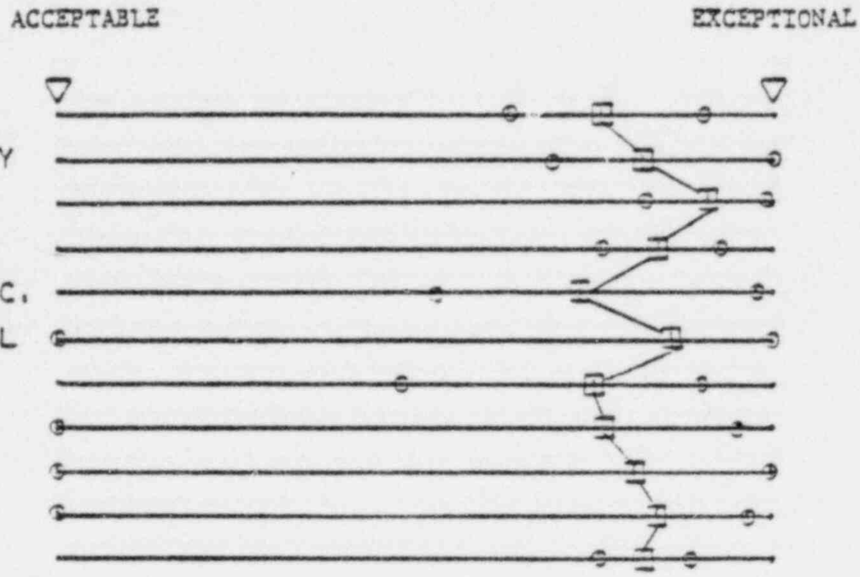
Design and operation of this early BWR are relatively uncomplicated. Plant safety improving due to continuing implementation of QA program and improving technical capability of staff.

SITE D. C. Cook

DOCKET NUMBER 50-315

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 7

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.7
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 7.5

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 5.1
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................... 3
2 = SAFETY SLIGHTLY IMPROVED............. 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 2
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Plant has standardized Technical Specifications. Resident inspector stationed site for some time. Plant has had increased personnel and procedural errors in 1977. Safety at Unit 1 is slightly worse because plant personnel and management have diverted attention to Unit 2 startup, fire protection, and security. Events are occurring that would not have a year ago.

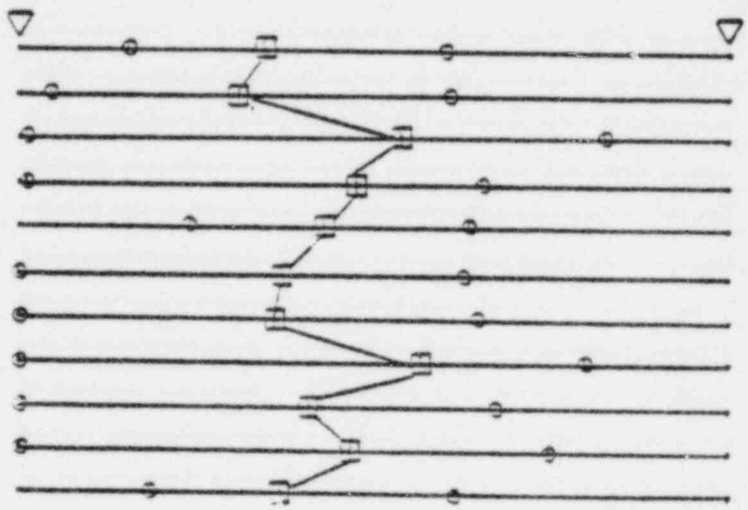
SITE Dresden

DOCKET NUMBER 50-010

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.1

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 5
2 = SAFETY SLIGHTLY IMPROVED............. 3
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 1

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Training and QA programs have improved. Unit 1, a smaller plant, does not receive the priority attention of Units 2 and 3. Manpower availability is a concern. Safety has improved due to better housekeeping and attention to detail. Safety is substantially worse due to poor operations and instrumentation problems.

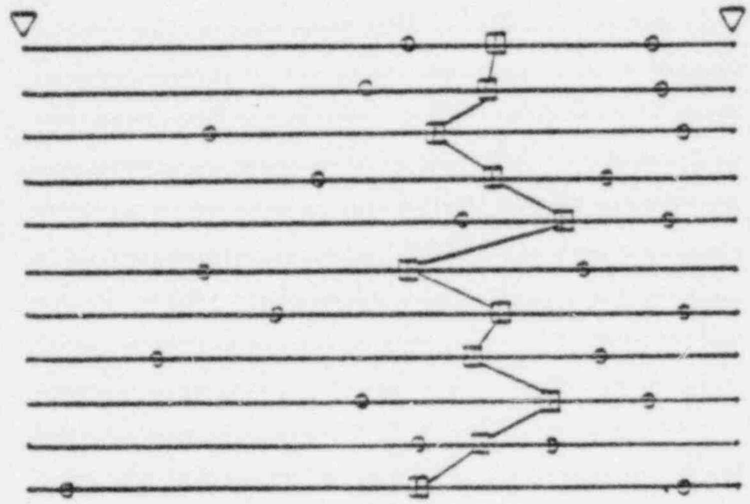
SITE Kewaunee

DOCKET NUMBER 50-305

ACCEPTABLE EXCEPTIONAL

RATING CATEGORIES

Overall safety
Attitude toward safety
Cooperation with NRC
Technical competence
Quality of design, etc.
Administrative control
Operations
Emergency planning
Radiation control
Safeguards
Quality Assurance



Number of people rating site = 6

Familiarity of raters with site (on 7 point scale) = 4.8
(1 = hardly at all, 7 = extremely well)

Average number of months since raters' last inspection = 6.5

Stringency of requirements for site (on 7 point scale) = 4.3
(1 = much less demanding than those of other sites,
7 = much more demanding than those of other sites)

Indications of change in site safety since January 1977

1 = No change in safety.................. 5
2 = Safety slightly improved............. 0
3 = Safety substantially improved........ 0
4 = Safety slightly worse................ 0
5 = Safety substantially worse........... 0

Narrative statements of changes in safety and other safety considerations

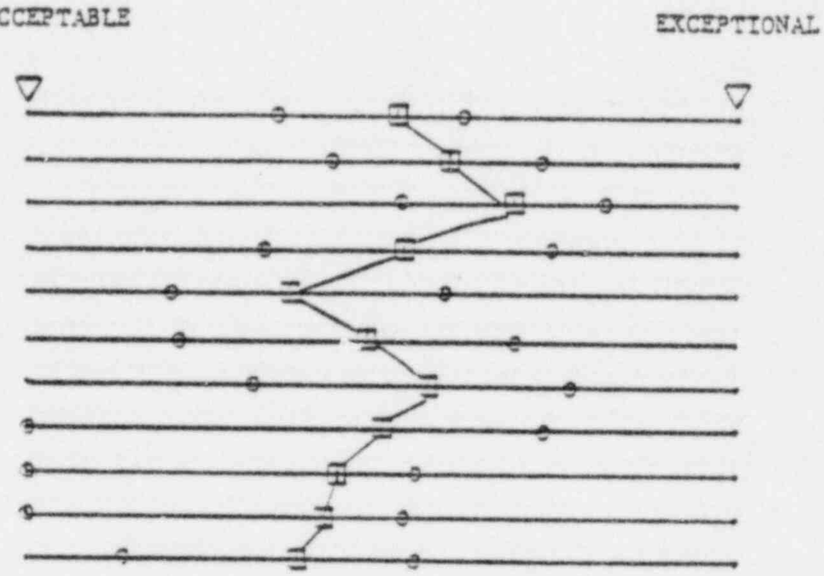
Resident inspector was assigned at this site. Plant management very stable and compent. Good attribute toward safety. Overall, the site has good operating performance.

SITE LaCrosse

DOCKET NUMBER 50-409

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 7

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.0
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 2.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 4
2 = SAFETY SLIGHTLY IMPROVED............. 2
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety slightly worse because of fuel degradation. Safety slightly better because of improved QA program. This plant is an AEC Developmental Reactor with a limited technical staff and minimal corporate backup. This small utility has difficulty absorbing the costs of NRC regulation.

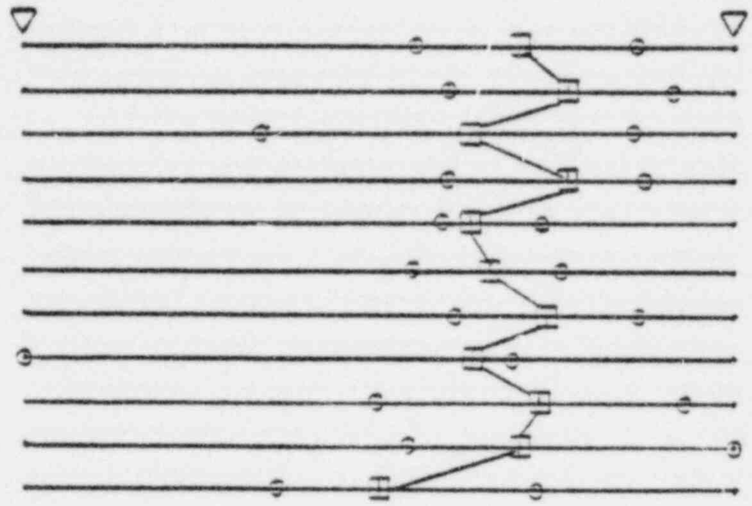
SITE Monticello

DOCKET NUMBER 50-263.

ACCEPTABLE EXCEPTIONAL

RATING CATEGORIES

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 8

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.1
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.5

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

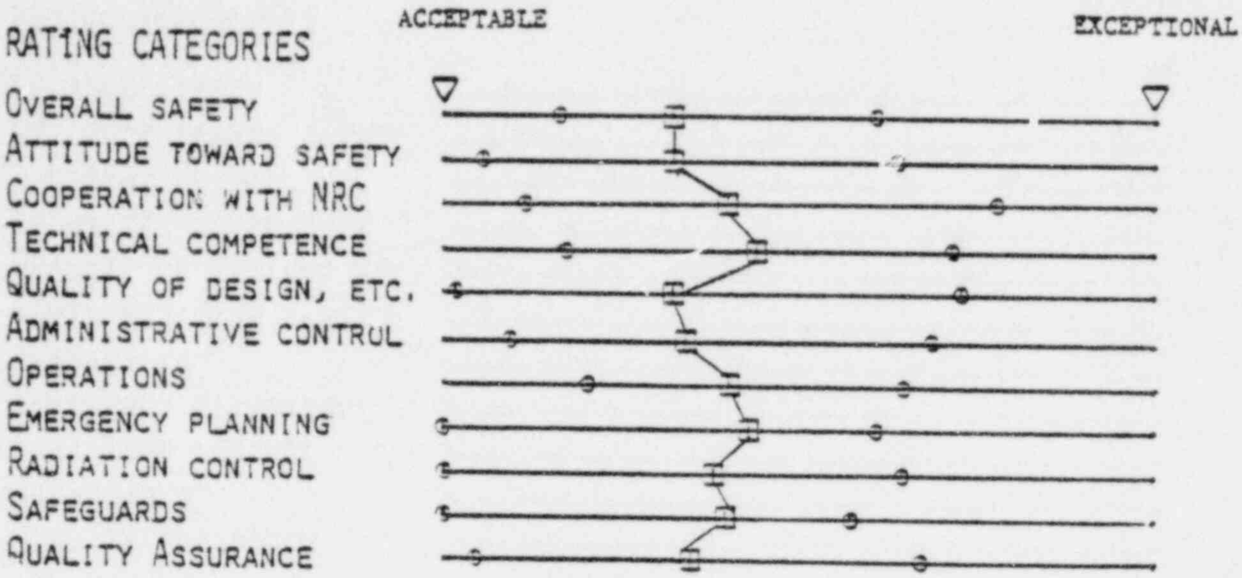
1 = NO CHANGE IN SAFETY.................. 4
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

No narrative comments.

SITE Palisades

DOCKET NUMBER 50-255



NUMBER OF PEOPLE RATING SITE = 8

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.8
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.4

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.8
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................... 2
2 = SAFETY SLIGHTLY IMPROVED............. 3
3 = SAFETY SUBSTANTIALLY IMPROVED....... 1
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE.......... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety is improved as a result of continuing QA program implementation. Management has been more attentive to the timely correction of problems. Resident inspector was assigned to site.

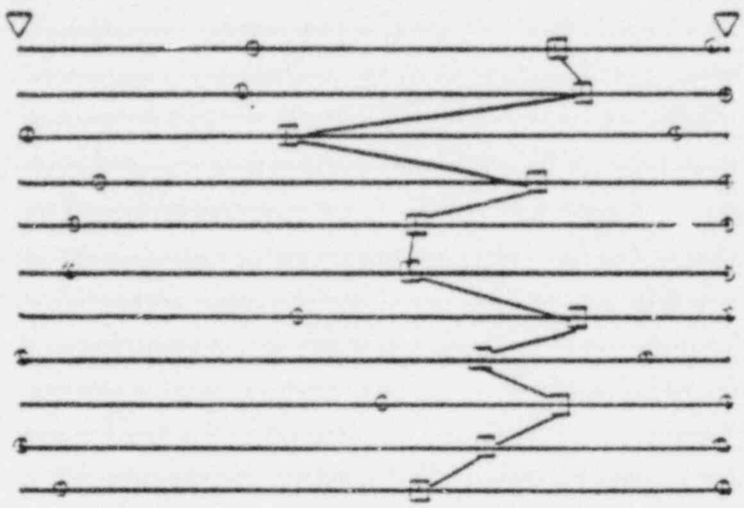
SITE Point Beach

DOCKET NUMBER 50-266

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

- OVERALL SAFETY
- ATTITUDE TOWARD SAFETY
- COOPERATION WITH NRC
- TECHNICAL COMPETENCE
- QUALITY OF DESIGN, ETC.
- ADMINISTRATIVE CONTROL
- OPERATIONS
- EMERGENCY PLANNING
- RADIATION CONTROL
- SAFEGUARDS
- QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 11.1

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.8
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 6
2 = SAFETY SLIGHTLY IMPROVED............. 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Plant is an older design attitude of plant management is extremely good. Staff is disciplined, well motivated, and proud of work. Staff offers constructive criticism of NRC. Plant management is strong in all areas, and has a total team effort from staff. Attitude on safety matters is excellent.

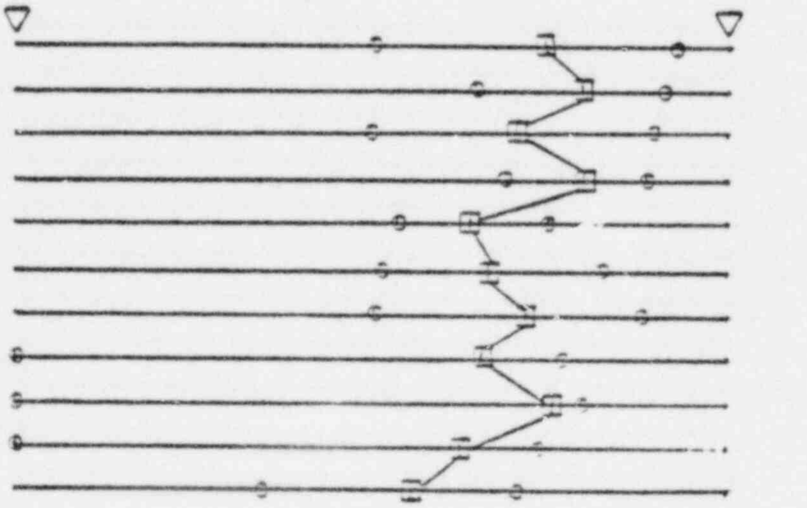
SITE Prairie Island

DOCKET NUMBER 50-282

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 8

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.8
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 8.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.3
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

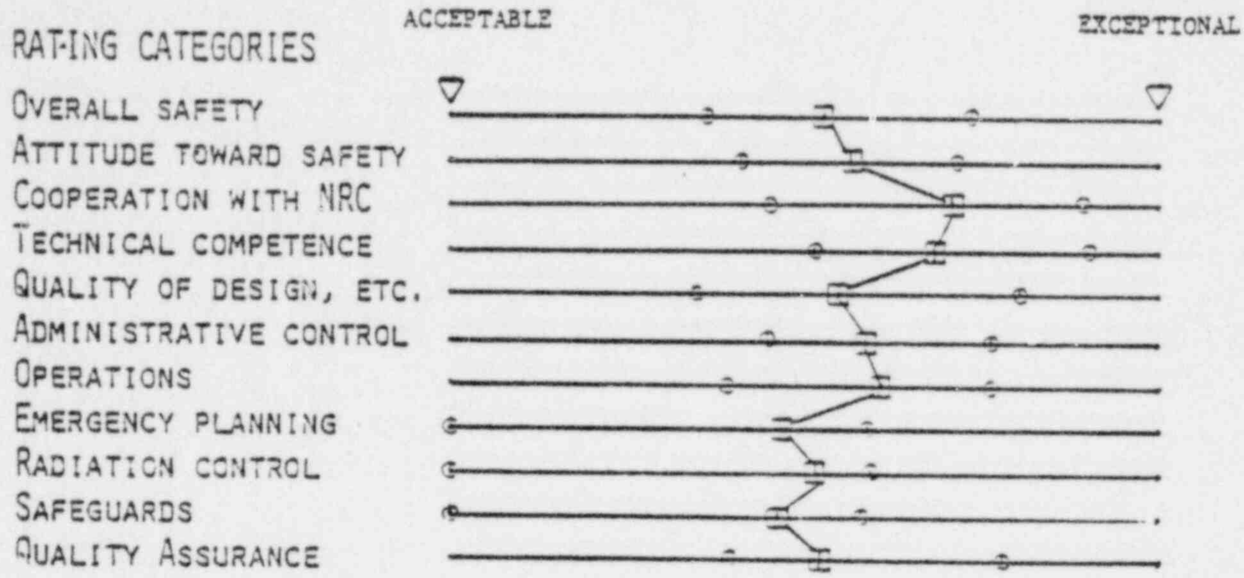
1 = NO CHANGE IN SAFETY.................. 5
2 = SAFETY SLIGHTLY IMPROVED............. 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

The technical staff is closely integrated with operations and maintenance; this helps prevent safety problems and provides good information.

SITE Quad Cities

DOCKET NUMBER 50-254



NUMBER OF PEOPLE RATING SITE = 10

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.3
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 15.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.9
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 4
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE.......... 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

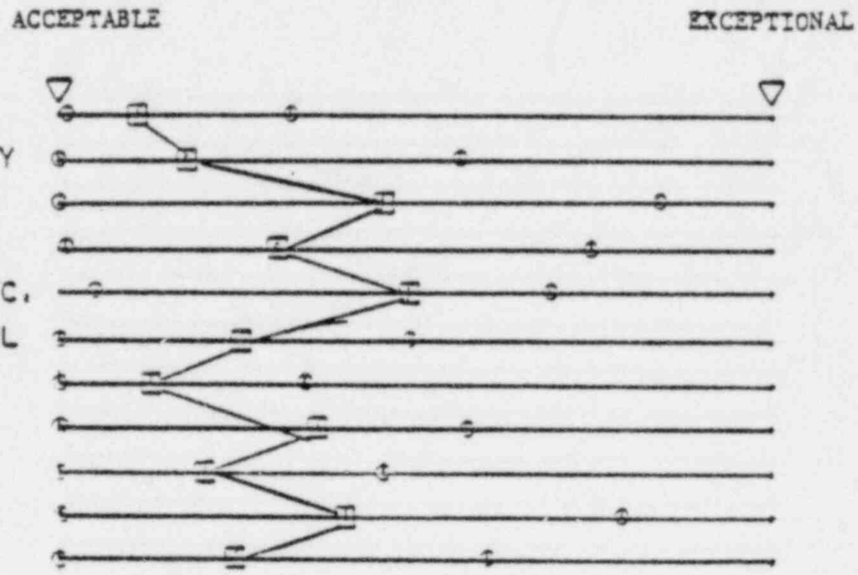
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Licensee has been "overinspected" by NRC and the state for several years. Plant not permitted by state to operate at design load; this affects operator attitudes. Safety slightly improved because of improvements in the training program, the QA program, and the radiological program.

SITE Zion

DOCKET NUMBER 50-295

RATING CATEGORIES



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 13

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 4.6
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.7

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.4
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 4
2 = SAFETY SLIGHTLY IMPROVED............ 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 1
5 = SAFETY SUBSTANTIALLY WORSE........... 3

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

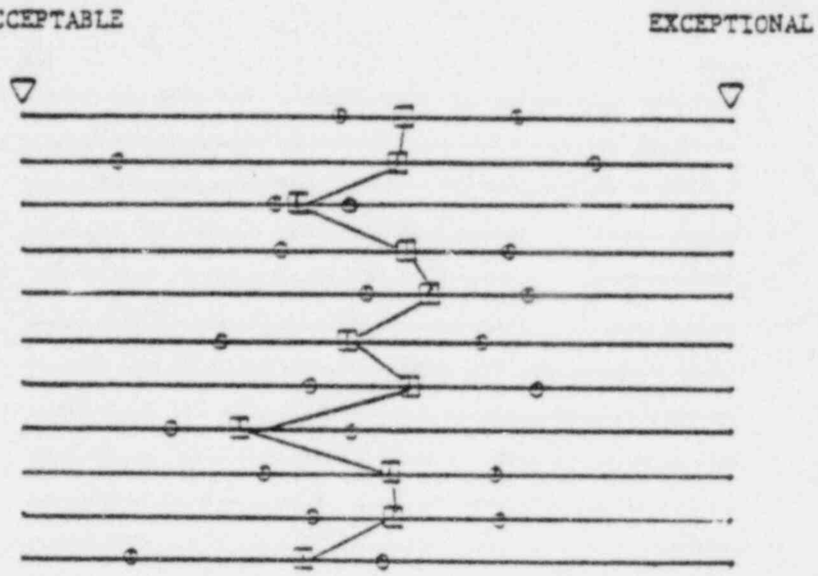
Safety is substantially worse because of poor attitude and marginal management. Inadequate management controls. Management lacks ability to discipline employees for operator errors and carelessness. Personnel selection and discipline may be adversely affected by union relations. Poor management attitude and followups. Size of Commonwealth Edison creates special management problems. Stability of staff a problem. Safety is substantially worse because of failures to conform to Tech Specs and administrative, operating, emergency, and test procedures. Attitude regarding safety is poor. Some improvements in procedures and training.

SITE Arkansas

DOCKET NUMBER 50-313

RATING CATEGORIES

Overall safety
Attitude toward safety
Cooperation with NRC
Technical competence
Quality of design, etc.
Administrative control
Operations
Emergency planning
Radiation control
Safeguards
Quality Assurance



Number of people rating site = 4

Familiarity of raters with site (on 7 point scale) = 5.3
(1 = hardly at all, 7 = extremely well)

Average number of months since raters' last inspection = 1.7

Stringency of requirements for site (on 7 point scale) = 3.5
(1 = much less demanding than those of other sites,
7 = much more demanding than those of other sites)

Indications of change in site safety since January 1977

1 = no change in safety.................... 3
2 = safety slightly improved............... 1
3 = safety substantially improved.......... 0
4 = safety slightly worse.................. 0
5 = safety substantially worse............. 0

Narrative statements of changes in safety and other safety considerations

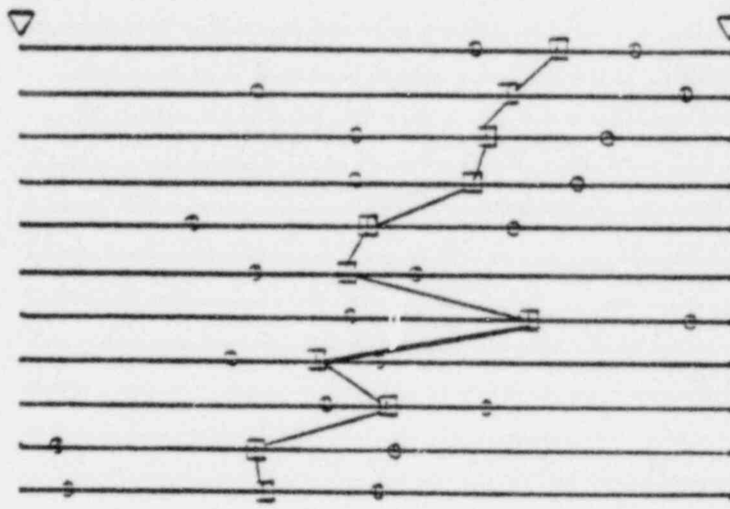
Management control of plant may be diluted when Unit 2 becomes operational. Safety slightly improved by upgrading of cable penetration barriers, fire protection, and procedural controls. Tech Specs should be upgraded to standard levels.

SITE Cooper

DOCKET NUMBER 50-298

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 4

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.0
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 6.5

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.0
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............ 0
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE............... 1
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

No narrative comments.

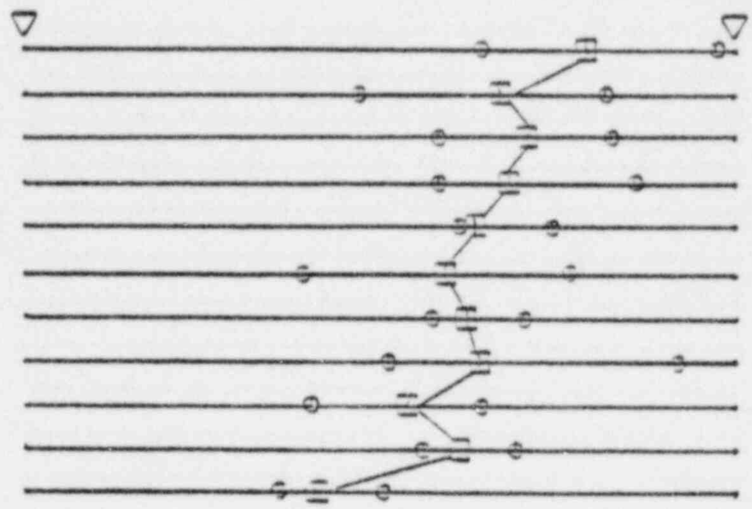
SITE Fort Calhoun

DOCKET NUMBER 50-285

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 3

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.3
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 1.0

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.7
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 2
2 = SAFETY SLIGHTLY IMPROVED.............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

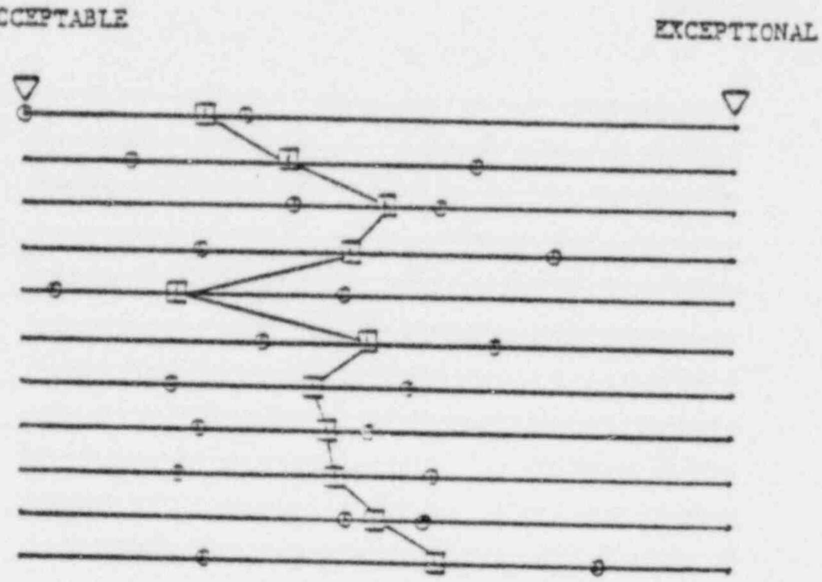
NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety at this plant is improving as management matures. Management recognizes its safety responsibilities. Employee morale could be affected by top utility attitudes about nuclear power.

SITE Fort St. Vrain

DOCKET NUMBER 50-267

RATING CATEGORIES



OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE

NUMBER OF PEOPLE RATING SITE = 5

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.2
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 4.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.0
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 1
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety substantially improved due to upgrading of cable separation, fire prevention, training program, and operating experience. Have been instrumentation improvements. This HTGR could be categorized as a demonstration plant. Plant safety characteristics are unique. Existing Tech. Specs. need revision.

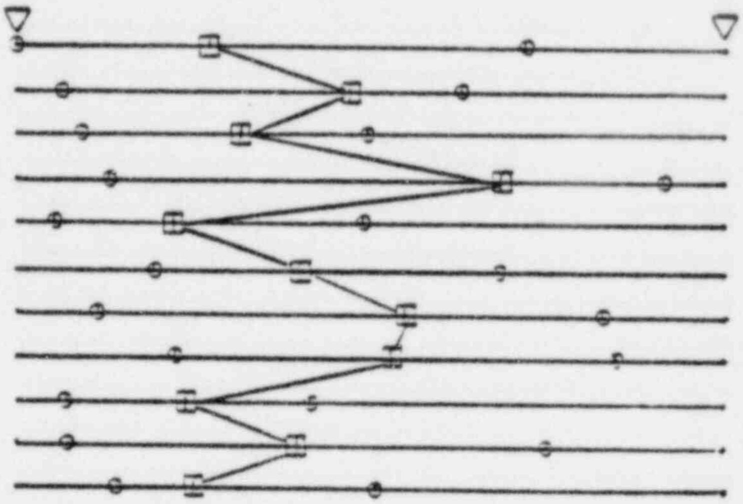
SITE Hu ldt Bay

DOCKET NUMBER 50-133

RATING CATEGORIES

ACCEPTABLE EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 7

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.9
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 9.3

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 2.6
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 3
2 = SAFETY SLIGHTLY IMPROVED............. 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 3
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety is substantially improved due to seismic modifications. Other safety-relevant matters are being pursued by NRR. The plant would be hard pressed to meet current safety criteria.

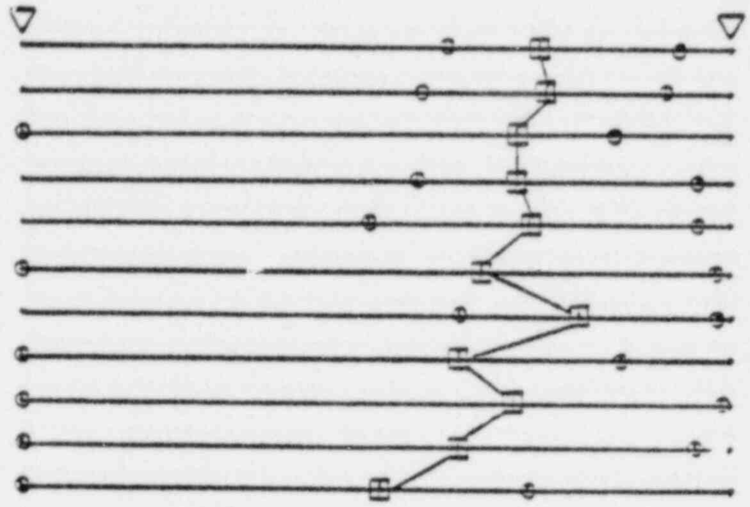
SITE Rancho Seco

DOCKET NUMBER 50-312

ACCEPTABLE EXCEPTIONAL

RATING CATEGORIES

- OVERALL SAFETY
- ATTITUDE TOWARD SAFETY
- COOPERATION WITH NRC
- TECHNICAL COMPETENCE
- QUALITY OF DESIGN, ETC.
- ADMINISTRATIVE CONTROL
- OPERATIONS
- EMERGENCY PLANNING
- RADIATION CONTROL
- SAFEGUARDS
- QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 7

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.1
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 2.8

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 4.1
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 5
2 = SAFETY SLIGHTLY IMPROVED............ 1
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety is slightly improved due to increasing operating experience and quality of plant management.

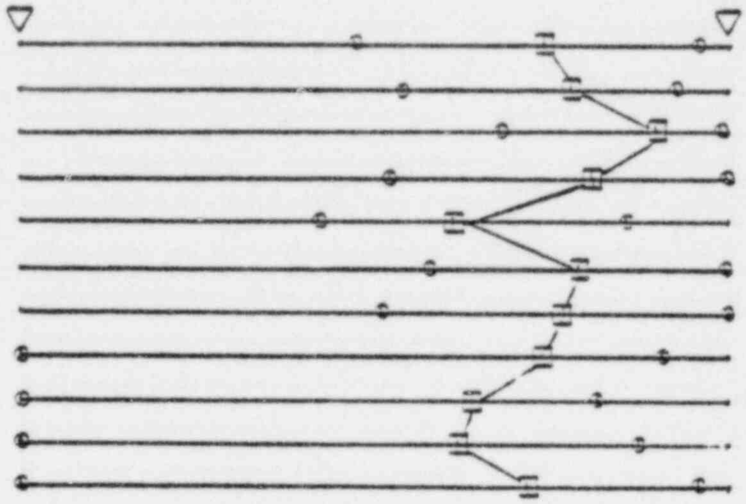
SITE San Onofre

DOCKET NUMBER 50-206

RATING CATEGORIES

ACCEPTABLE — EXCEPTIONAL

OVERALL SAFETY
ATTITUDE TOWARD SAFETY
COOPERATION WITH NRC
TECHNICAL COMPETENCE
QUALITY OF DESIGN, ETC.
ADMINISTRATIVE CONTROL
OPERATIONS
EMERGENCY PLANNING
RADIATION CONTROL
SAFEGUARDS
QUALITY ASSURANCE



NUMBER OF PEOPLE RATING SITE = 8

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.4
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 4.7

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 3.6
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

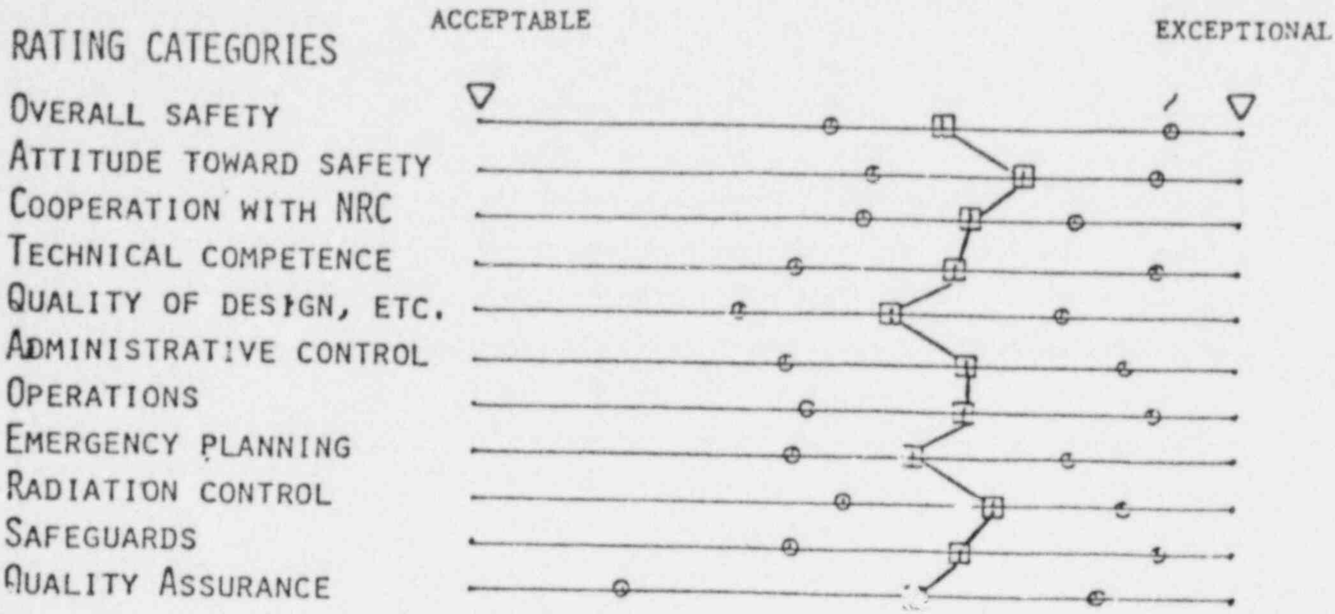
1 = NO CHANGE IN SAFETY.................. 2
2 = SAFETY SLIGHTLY IMPROVED............. 2
3 = SAFETY SUBSTANTIALLY IMPROVED........ 3
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety is slightly improved because of QA program improvements. Safety is substantially improved because of upgrading of the emergency power system. Utility management has been successful in instilling good safety attitudes and habits uniformly throughout the organization. Extensive ECCS and seismic modes have been completed.

SITE Trojan

DOCKET NUMBER 50-344



NUMBER OF PEOPLE RATING SITE = 8

FAMILIARITY OF RATERS WITH SITE (ON 7 POINT SCALE) = 5.4
(1 = HARDLY AT ALL, 7 = EXTREMELY WELL)

AVERAGE NUMBER OF MONTHS SINCE RATERS' LAST INSPECTION = 3.7

STRINGENCY OF REQUIREMENTS FOR SITE (ON 7 POINT SCALE) = 5.5
(1 = MUCH LESS DEMANDING THAN THOSE OF OTHER SITES,
7 = MUCH MORE DEMANDING THAN THOSE OF OTHER SITES)

INDICATIONS OF CHANGE IN SITE SAFETY SINCE JANUARY 1977

1 = NO CHANGE IN SAFETY.................. 1
2 = SAFETY SLIGHTLY IMPROVED............. 5
3 = SAFETY SUBSTANTIALLY IMPROVED........ 0
4 = SAFETY SLIGHTLY WORSE................ 0
5 = SAFETY SUBSTANTIALLY WORSE........... 0

NARRATIVE STATEMENTS OF CHANGES IN SAFETY AND OTHER SAFETY CONSIDERATIONS

Safety slightly improved by equipment upgrading and accumulation of operating experience. Active state regulation could affect safety through conflicting requirements. On site QA program implementation has improved. Fire protection program is being implemented. Attitude toward QA and prevention of recurring problems has improved.

ADDENDUM

TO

INDIVIDUAL SITE RATINGS

FROM THE

IE EMPLOYEE SURVEY ON

EVALUATION OF LICENSEES

APRIL 1978

The narrative statements provided in connection with the sheet for each site in the preceding section of this report were based on comments made by the inspectors regarding those sites. The actual comments made by the inspectors with respect to individual sites are contained in this addendum.

Changes in level of safety.

Site: Beaver Valley
Docket No.: 50-334

Plant is just completing startup testing and staff is more experienced.

QA controls slightly better.

Controls over explosive blow-out discs were established after identified by inspector.

Plant personnel are becoming more experienced, confident and competent. Bugs are gradually being worked out of equipment and administrative controls.

Plant management has improved.

Increased security requirements; i.e., additional guard force, increased surveillance, addition of mechanical search equipment (guard force doubled in last year).

New plant - only recently completed final testing - plant and management still learning of plant and design problems.

Site: Calvert Cliffs
Docket No.: 50-317

Management became more cognizant of plant operations following an enforcement meeting in early 1977.

Have a smaller "Q" list to which they apply their controls.

Improvements in security.

Completion of startup testing on Unit 2.

Increased attention to procedural adherence and plant cleanliness due to escalated enforcement action by IE.

Both plants, each operating. New upgraded T/S at both plants.

Site: Connecticut Yankee
Docket No.: 50-213

Review of inspection findings, LERs, and operating record supports this judgment.

Site: Fitzpatrick
Docket No.: 50-333

Take over by PANSY appears to be an improvement.

More management attention to operations. Change in operating licensee.

New security procedures.

Change in operating license from Niagara Mohawk to PANSY increased technical level of management and administrative controls.

Design changes to install additional safety systems.

Corporate management change NM to PANSY.

Site: Ginna
Docket No.: 50-244

None.

Site: Indian Point
Docket No.: 50-247, 286

Much recent IE and licensee management attention to IP-2 operations, health physics, safeguards, etc., has resulted in large overall licensee upgrading.

Improvements in radiation health controls.

Recently completed an intensive inspection program in rad protection - organizational changes were made, new procedures provided and a significant improvement in management control.

Inspection effort has improved management attention to factors affecting plant safety.

Applied considerable inspection effort and "talent" and convinced corporate management that they had to expand corporate resources.

Site: Maine Yankee
Docket No.: 50-309

None.

Site: Millstone
Docket No.: 50-245

More safety awareness.

New security fence and procedures.

Re-evaluations have been made and design changes implemented in plant power distribution and emergency power systems.

Review of inspection findings, LERs and operating record would support this judgment.

New QA organization seems to be slightly more effective.

Site: Nine Mile Point
Docket No.: 50-224

None.

Site: Oyster Creek
Docket No.: 50-219

Imposition of new operational procedures and facility record maintenance system has improved safety.

Installation of storage facility to house torus chromated water - and permit draining of torus.

QA program has been more fully implemented. New storage facilities, new document control center becomes operational.

Substantial upgrading of QA has been, and is, in progress.

Site: Peach Bottom
Docket No.: 50-277

Plant radiation levels have been increasing with time. Design and staffing of plant appear to have not been capable of handling this change. Management has been slow to take large step changes to correct problems.

Back to back overhaul/upkeep periods for units 2 & 3 appear to have produced a tired operating group prone to error.

Careless operations and poor maintenance.

Corrective action taken to repair core spray line cracks, feedwater spargers and nozzles and control rod drive return nozzle.

Licensee made significant effort to reduce routine radioactive release from reactor building vents through equipment repairs.

Site: Pilgrim
Docket No.: 50-293

Improved corporate management. Improved radiation management at site.

Due to instability in plant management. Drift due to lack of management direction.

Refueling outage.

Site: Salem
Docket No.: 50-272

Relatively new plant. Still has growing pains. Needs close attention (by IE) to assure appropriate improvements are made.

Power ascension testing revealed problems that were corrected by management, both in hardware and procedures.

Site: Three Mile Island
Docket No.: 50-289

Increased security by addition of fence surveillance, guard force and search equipment.

Site: Vermont Yankee
Docket No.: 50-271

Management experience and depth is increasing.

Site: Yankee Rowe
Docket No.: 50-029

Issuance of standard Technical Specifications.

Site: Browns Ferry
Docket No.: 50-259

Attention to QA principles seems somewhat less

a. More experience and exposure of plant personnel = improved safety and operation.

b. More inspections by NRC

c. Plant management changes

Improved response to alarms - enforcement meeting

More safety awareness

In fire protection

Site: Brunswick
Docket No.: 50-325

Some improvement in administrative controls. More experience by operating staff.

New management and experience.

Management seemed to become more aware of events at plant.

Site: Hatch
Docket No.: 50-321

Continued upgrading of Adm 8 QA control

Operating experience

Site: Oconee
Docket No.: 50-269

Change in Operating Superintendent should improve situation in next few months.

Site: Robinson
Docket No.: 50-261

Licensee has made increased site commitment to QA/QC.

Site: Saint Lucie
Docket No.: 50-335

Improved due to increased operations, etc., experience of plant personnel over time period involved.

Site: Surry
Docket No.: 50-280

One to degradation of steam generators.

Site: Turkey Point
Docket No.: 50-250

Safety may be slightly worse due to steam generator degradation.

Site: Crystal River
Docket No.: 50-302

More safety awareness

Improved Adm control. Improved Operations awareness.

Site: Arnold
Docket No.: 50-331

Improvement in administrative control and QA program.

New plant superintendent. Stronger enforcement action - increased inspection effort.

Management change.

More awareness regarding significance of personnel error.

Steady improvement in management controls and quality of onsite staff. Increased attention by engineering and corporate office.

Site: Big Rock Point
Docket No.: 50-155

QA program implementation continuing resulting in an improved plant safety level.

Site: D. C. Cook
Docket No.. 50-315

Increased number of personnel errors and procedural violations occurred during 1977.

Demands placed upon personnel and management due to Unit 2 startup, fire protection and security have brought a decrease in attention and review unit 1 is given. Events are occurring that would not have a year ago.

Site: Dresden
Docket No.: 50-010

Improved training program, and improved QA programs.

Better housekeeping, more attention to detail.

Poor operation, instrumentation problems.

Site: Kewanee
Docket No.: 50-305

None.

Site: LaCrosse
Docket No.: 50-409

Improved QA program

Site: Monticello
Docket No.: 50-263

None

Site: Palisades
Docket No.: 50-255

QA scope implementation continuing resulting in an improved plant safety level.

Improved attention by management toward more timely correction of problems.

Site: Point Beach
Docket No.: 50-266

None.

Site: Prairie Island
Docket No.: 50-282

None.

Site: Quad Cities
Docket No.: 50-254

Improvement in training program, improved QA program, improved radiological program.

Site: Zion
Docket No.: 50-295

Apparent PWR attitude of personnel resulting from marginal management.

Safety reduced as evidenced by loss of DC power and by passing all pressurizer level channels in 1977. Inadequate management controls.

Continued deterioration of management controls.

Nonconformance with technical specifications; failure to adhere to administrative procedure; failure to adhere to operating, emergency, and test procedures; inadequate procedure; operator error; poor overall operating performance; weak overall management.

Procedures improved; administrative procedures improved; better training.

Site: Arkansas
Docket No.: 50-313

Cable penetration barriers and fire proofing of essential and safety cables. Improvement in procedural controls.

Site: Cooper Station
Docket No.: 50-298

None.

Site: Fort Calhoun
Docket No.: 50-285

Site management at this plant is young and they are maturing and recognizing their safety responsibilities.

Site: Fort St. Vrain
Docket No.: 50-267

Cable separation, training program, penetration fire barriers and flammastic on essential and safety related cable. Experience of operating personnel as operation of plant continues.

Based on an IE inspection the licensee has recently had to review the setpoints of his safety systems to determine that instrument and calibration inaccuracies are adequately accounted for in the selected setpoints.

Site: Humboldt Bay
Docket No.: 50-133

Seismic modifications completed during past year.

Seismic modifications have been performed, feedwater sparger has been replaced.

Upgrading structures to new seismic criteria.

The plant has undergone an extensive outage to upgrade the structural integrity of the facility to limit seismic damage.

Plant shutdown for extensive modification in July 1976.

Site: Rancho Seco
Docket No.: 50-312

Overall plant safety increasing with experience of operations organization and management's understanding and knowledge of nuclear plant operations.

Site: San Onofre
Docket No.: 50-246

QA program improvement.

Completed outage which improved their emergency power capability substantially.

Installed emergency diesel generator capacity to carry LOCA load coincident with loss of off-site power. Also, constructed concrete shield around containment vessel.

Extensive ECCS and seismic modifications have been completed.

Installation of onsite emergency power capability.

Site: Trojan
Docket No.: 50-344

Equipment improvements in engineered safety features brought about by operating experiences.

Improving with experience as operating organization and management matures and gains nuclear experience.

QA program implementation onsite has substantially improved by identifying problems before they became issues or items of noncompliance detected by NRC inspectors.

Fire protection program is being implemented.

Improved attitude toward value of QA auditing and initiating corrective measures to correct recurring deficiencies identified from operating experience.

Other things relevant to safety of this site?

Site: Beaver Valley
Docket No.: 50-334

Technical competence of management personnel.

New plant - recently completed full power testing.

Site: Calvert Cliffs
Docket No.: 50-317

The Chief Engineer is anti-NRC, anti-QA.

The operation philosophy of this plant is 2.5 and survive - they don't do anything above that which is required toward plant safety.

This facility appears to place prime interest upon operating, to the extent of voluntary entrance into action statements. Its attitude toward safety appears to be that meeting literal NRC requirements is sufficient.

Management meeting held to impress President with our observations of the dedication of plant staff to "get the turbine on line" at the risk of not having assured that T/S requirements are met. Too early to determine the result of the meeting.

Site: Connecticut Yankee
Docket No.: 50-213

Age of plant.

NRR is backfitting CY in several areas. When this is completed, the design requirements and license conditions will be upgraded, and therefore, overall safety should be improved.

Site: Fitzpatrick
Docket No.: 50-333

Has a new operator (PANSY) for the plant, including new plant management.

Site: Fitzpatrick (Continued)
Docket No.: 50-333

Later design provides better safety systems, such as rod sequence control system, etc., but emergency diesel generators are not reliable and radioactive waste systems are underdesigned and marginally operated. Excellent fire protection system, excellent security program.

Station management recently changed from Niagara Mohawk to PANSY - improvements already noted - more anticipated.

Site: Ginna
Docket No.: 50-244

The plant is old, small, and run safely---the small aspect is important because of the relative lack of danger to the public.

Recent change in station superintendent - no significant change noted.

Site: Indian Point
Docket No.: 50-247, 286

The ratings indicated are for Indian Point 2 in that Indian Point 3 is highly superior in all aspects as related to Unit 2 due primarily to management controls and personnel.

Facility operation at full power with question on calibration of nuclear instruments and resolution of read-out available to operations. Management is aware of problem and IE is following.

Do not have accepted QA plan meeting current requirements. Should be approved soon. Unit 3 would be better rated because PANSY does better than Con Ed.

Upper management (corporate) attitudes continue to limit effectiveness of site management.

Continue to inspect and observe with highly competent and experienced inspectors. The trend toward more inspections with less competent inspectors is dangerous. Also, continue design reviews by highly competent NRR personnel - also tighten standards and codes, and operator license examinations.

Site: Maine Yankee
Docket No.: 50-309

The plant is very clean - it shows pride in ownership and is indicative of happy people working at a good plant.

Have recently approved QA plan - upgraded to current standards. Became effective 8/16/77.

Recent change in station superintendent - no significant changes in safety expected.

Site: Millstone
Docket No.: 50-245

Large public interest in events taking place at this facility. Have a new plant superintendent.

Unit 1 is a BWR which is old - these items combine to cause a lower rating for Unit 1 than Unit 2.

Millstone site has three reactors, operating BWR, operating PWR, under construction PWR - all are by different vendors - all of different "era" - the operating reactors are, relatively, independent (as compared to a multiple unit site with the same generation of reactor from the same vendor) in their inherent safety characteristics.

Reliability of emergency gas turbine, acceptance of the feedwater injection system as a high pressure ECCS system. Plant lacks a lot of separation and fire protection systems. Rad waste system undersized.

Inter-relationship between diverse units at single site.

Site: Nine Mile Point
Docket No.: 50-224

There were some old fossil people managing and operating this plant - they don't have the nuclear ethic yet.

This is a plant of older design but the early engineering was of a high quality and excellent plant layout and construction. Onsite plant support (other than operations) lacking in numbers of people. Plant lacks system operation and a real high pressure inspection system. Excellent security program.

Site: Nine Mile Point (Continued)
Docket No.: 50-224

Approach to operations of plant have been conservative. Plant staff has been stable.

Nine Mile also has considerable operating experience, and a reservoir of experienced BWR operators (from Fitzpatrick which has until recently been operated by the Nine Mile licensee and which "leases" its operators from Niagara Mohawk until it trains its own).

Corporate engineering role in maintenance activities.

Site: Oyster Creek
Docket No.: 50-219

Security should be upgraded, i.e., increase capabilities of guard force and surveillance equipment.

Upgrading of requirements, imposition of environmental T.S.

An early generation BWR - its age and generation made it different in inherent safety from facilities - and facility management has been less than willing to endorse in principle a comprehensive management control system - they conform as required rather than aggressively prosecute.

This plant received a poor design review as demonstrated by logic system inadequacies, recently found. Plant was built at minimum cost. Radioactive waste and fire protection are inadequate. Plant lacks system separation.

Management at corporate level has a first-hand technical and working level knowledge of the plant.

Site: Peach Bottom
Docket No.: 50-277

QA program not upgraded to current standards. Security not upgraded. Many repeat items of noncompliance. Least safe plant in RI! Poorest management!

Site: Peach Bottom (Continued)
Docket No.: 50-277

Quality of people (i.e., technical educational level) that are operating a plant and the type of organizational structure they are placed in can have a significant impact on safety.

Higher number of inspections due to proximity to regional office.

Recent management meeting with the President - expect to determine by scheduled inspections in the next 30 days if significant improvements were made.

Plant management exhibits an appearance of attempting to "control" NRC inspector access thru continual escort - general attitude appears to be one of compliance as required instead of an aggressive prosecution of management controls.

The problem with this plant is that it is a big BWR - by definition, they will have problems unless they have a good operating staff. PB does...

Upgrading of requirements upon this license, particula ly in cases of security and QA.

Site: Pilgrim
Docket No.: 50-293

Generation of design may be the overriding factor for this early generation BWR.

Have experienced a number of station manager changes.

Recent change in corporate radiological protection and all old fuel is being removed. Significant improvements in reducing effluents and worker exposure expected.

Several changes in upper level management, some instability because of changes.

The cleanest BWR in the country.

Site: Salem
Docket No.: 50-272

The plant control room was designed in-house - it is a disaster waiting to happen.

In startup phase. Have had a number of problems. This can be due either to poor system or poor management or the "normal" failures when new systems are placed into service.

Design of controls with back-lighted pushbuttons results in operator data assessment problems, especially when lights are burned out. Management is aware of problem and IE is following up.

New plant - recently completed full power testing - plant still in early operating phases.

Site: Three Mile Island
Docket No.: 50-289

Pay close attention to performance of newly assigned station and unit superintendents.

This is the first designed B&W plant of this generation. Construction was largely accomplished without aggressive prosecution of nuclear management control. Operation is conducted under strong management control.

The licensing of Unit 2 i ..,77 will have an impact on the site/corporate staffs. In all probability the overall safety may become worse over the year due to this increased workload.

2nd plant in startup places some additional "drag" on operating facility equipment and manpower.

Site: Vermont Yankee
Docket No.: 50-271

Have upgraded QA plan which became effective 8/16/77.

Frequent changes in plant superintendent - has resulted in slight degradation of management controls.

Very clean.

Public interest in events at site.

Site: Yankee Rowe
Docket No.: 50-029

Plant is very small and very isolated - virtually no health hazard to the public exists.

Old plant Tech Specs. New, upgraded QA program became effective 8/16/77.

Site: Browns Ferry
Docket No.: 50-259

Core performance analysis, qualifications of technicians and mechanics who maintain safety equipment.

Site: Brunswick
Docket No.: 50-325

The training or experience of senior site management - none of the top three have had SRO training in BWRs. The plant has had a very high personnel turnover rate. Consequently, the staff is young for the responsibilities needed. Corporate management apparently still has not faced up to what this inexperience costs in safety and efficiency. They appear to believe they are being over-regulated.

Qualifications of technicians and mechanics that maintain safety equipment.

All pre-op testing must be completed prior to licensing.

Site: Hatch
Docket No.: 50-321

Qualifications of technicians and mechanics that maintain safety equipment.

Site: Oconee
Docket Number: 50-269

Qualifications of technicians and mechanics that maintain safety equipment. Maintenance of test equipment.

Site: Robinson
Docket No.: 50-261

Low number of LERs reflects attitude of reporting only items that are conspicuously reportable. Licensee impedes IE freedom of movement and access at site. No information freely given. Definite attitude of do only what is required.

Qualifications of technicians and mechanics that maintain safety systems.

Site: Saint Lucie
Docket No.: 50-335

This plant has more than average number of LER's. I believe this is due to Licensee's determination to report all possibly reportable items rather than poor perf[illegible]ance.

Site: Surry
Docket No.: 50-280

None.

Site: Turkey Point
Docket No.: 50-250

Qualifications of technicians and mechanics that maintain safety equipment.

Site: Crystal River
Docket No.: 50-302

None.

Site: Arnold
Docket No.: 50-331

Site: Big Rock Point
Docket No.: 50-155

On original BWR - Design, operation relatively uncomplicated. Closeness of operating staff.

Site: Big Rock Point (Continued)
Docket No.: 50-155

General safety of older plants.

Plant personnel qualifications have improved (technical capability) monthly.

Site: D. C. Cook
Docket No.: 50-315

One plant in operations the other in startup. Plant using standardized Tech Specs.

Resident inspector stationed thru 74-77.

Design, its newer with greater indepth protection.

Site: Dresden
Docket No.: 50-010

See Zion commen

U-1 is a 200 MWe plant while U-2&3 are 800 MWe each - U-1 will never receive priority at the management level - One should also consider the manpower availability on site.

Site: Kewanee
Docket No.: 50-305

Resident Inspector assigned 74-76.

Very stable and competent plant management; overall good operating performance; strong safety attitude.

Site: LaCrosse
Docket No.: 50-409

Part 115 plant (AEC developmental reactor) small utility - limited technical staff with minimal corporate backup - difficult to absorb costly NRC regulations.

Site: Monticello
Docket No.: 50-263

None

Site: Palisades
Docket No.: 50-255

Utility constantly confronted by intervention - legal challenge from the outside.

Effectiveness of management controls. Resident inspection assigned 74-77.

Site: Point Beach
Docket No.: 50-266

This plant is of older design with its management attitudes it would be above exceptional if designed like present day.

Disciplined staff, well motivated, pride which includes their ability to positively criticize the NRC in matters which distract from their ability to conduct their plant operations.

The exceptional strength of plant management in all areas. The total team effort in all matters - the excellence of all personnel attitude in regard to safe plant operation.

Site: Prairie Island
Docket No.: 50-282

The technical staff is closely integrated with operations and maintenance. This helps resolve problems before safety concerns develop and provides good information where failures have occurred.

Site: Quad Cities
Docket No.: 50-254

See Zion comments.

The licensee has been "overinspected" by NRC and state for the past 2 or 3 years. The plant cannot operate at design load because of an agreement with the state to operate with a closed cycle cooling canal, after the plant was built as designed for once thru cooling. This affects plant operation and also attitudes of operators.

Site: Zion
Docket No.: 50-295

Lack of management. Ability of discipline employees for operators error/carelessness.

Management - union interface and its effect on selection of personnel and disciples. Attitude and support from support engineering and corporate management to resolve operating equipment problems. Corporate management involvement in plant operations. Corporate management attitudes and followup.

Part of a complex nuclear commitment which carries with it the management problems associated with "bigness." Stability of staff a continuous problem.

Overall attitude regarding safety is not strong. Lax operating performance and attitude.

Adequacy of training program; number of personnel errors resulting in significant problems.

Site: Arkansas
Docket No.: 50-313

Unit 2 which is soon to be operational will be managed by the same size management as that which controls Unit 1. I feel this practice considerably dilutes management's control over these plants.

Upgrade technical specifications to standard T/S.

Site: Cooper Station
Docket No.: 50-298

None.

Site: Fort Calhoun
Docket No.: 50-285

Top management (Board of Directors) have an anti-nuclear attitude which is upsetting to site personnel and management. Since there is a correlation between morale and job satisfiers, I am concerned about this situation. This concern is due to the fact that morale affects employee safety practices more than production.

Site: Fort St. Vrain
Docket No.: 50-267

First of a kind - fits the category of a demonstration site.

The basic design and configuration of the HTGR introduces a completely different set of parameters and accidents to be considered in plant safety.

This is a one of a kind HTGR. The existing regulatory guides, standards, etc., do not apply to this plant. The existing Technical Specifications need to be completely revised.

Site: Humboldt Bay
Docket No.: 50-133

The other matters I feel are necessary to consider are being pursued by NRR - they include adequacy of ECCS, single failure design and gaseous effluent treatment.

The plant would be hard pressed to meet any of today's criteria for nuclear plant safety.

No opinion.

New Technical Specifications.

Adequacy of seismic design, ECCS and reactor protection system. Results of analyzing these safety questions could change (significantly) my rating of overall plant safety.

Site: Rancho Seco
Docket No.: 50-312

Not that I'm aware of.

Site: San Onofre
Docket No.: 50-246

This company should be studied to determine how and why their management has been so successful in instilling good safety attitudes and habits so uniformly thru their organization.

Site: Trojan
Docket No.: 50-344

Active role of State of Oregon in attempting to regulate this plant could have an effect on safety - possibility of contradictory requirements and demands of federal/state agencies.

ADDENDUM

TO

INDIVIDUAL SITE RATINGS

FROM THE

IE EMPLOYEE SURVEY ON

EVALUATION OF LICENSEES

APRIL 1978

The narrative statements provided in connection with the sheet for each site in the preceding section of this report were based on comments made by the inspectors regarding those sites. The actual comments made by the inspectors with respect to individual sites are contained in this addendum.

Docket No.: 50-334
Site: Beaver Valley

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Plant is just completing startup testing and staff is more experienced.

QA controls slightly better.

Controls over explosive blow-out discs were established after identified by inspector.

Plant personnel are becoming more experienced, confident and competent. Bugs are gradually being worked out of equipment and administrative controls.

Plant management has improved.

Increased security requirements; i.e., additional guard force, increased surveillance, addition of mechanical search equipment (guard force doubled in last year).

New plant - only recently completed final testing - plant and management still learning of plant and design problems.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

New plant - recently completed full power testing.

Technical competence of management personnel.

Docket No.: 50-317
Site: Calvert Cliffs

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Management became more cognizant of plant operations following an enforcement meeting in early 1977.

Have a smaller "Q" list to which they apply their controls.

The (blank) is anti-NRC, anti-QA.

Improvements in security.

Completion of startup testing on Unit 2.

Increased attention to procedural adherence and plant cleanliness due to escalated enforcement action by IE.

Both plants, each operating. New upgraded T/S at both plants.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Management meeting held to impress President with our observations of the dedication of plant staff to "get the turbine on line" at the risk of not having assured that T/S requirements are met. Too early to determine the result of the meeting.

The operational philosophy of this plant is 2.5 and survive - they don't do anything above that which is required, towards plant safety.

This facility appears to place prime interest upon operating, to the extent of voluntary entrance into action statements. Its attitude toward safety appears to be that meeting literal NRC requirements is sufficient.

Docket No.: 50-213
Site: Connecticut Yankee

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Review of inspection findings, LERs, and operating record supports this judgment.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Age of plant.

NRR is backfitting CY in several areas. When this is completed, the design requirements and license conditions will be upgraded, and therefore, overall safety should be improved.

Docket No.: 50-333
Site: Fitzpatrick

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Take over by PANSY appears to be an improvement.

More management attention to operations. Change in operating licensee.

New security procedures.

Change in operating license from Niagara Mohawk to PANSY increased technical level of management and administrative controls.

Design changes to install additional safety systems.

Corporate management change NM to PANSY.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Has a new operator (PASNY) for the plant, including new plant management.

Later design provides better safety systems, such as rod sequence control system, etc., but emergency diesel generators are not reliable and radioactive waste systems are underdesigned and marginally operated. Excellent fire protection system, excellent security program.

Station management recently changed from Niagara Mohawk to PASNY. Improvements already noted - more anticipated.

Docket No.: 50-244
Site: Ginna

Answers to Questions 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

The plant is old, small, and run safely---the small aspect is important because of the relative lack of danger to the public.

Recent change in station superintendent - no significant change noted.

Docket No.: 50-003
Site: Indian Point

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Much recent IE and licensee management attention to IP-2 operations, health physics, safeguards, etc., has resulted in large overall licensee upgrading.

Improvements in radiation health controls.

Recently completed an intensive inspection program in rad protection - organizational changes were made, new procedures provided and a significant improvement in management control.

Inspection effort has improved management attention to factors affecting plant safety.

Applied considerable inspection effort and "talent" and convinced corporate management that they had to expand corporate resources.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

The ratings indicated are for Indian Point 2 in that Indian Point 3 is highly superior in all aspects as related to Unit 2 due primarily to management controls and personnel.

Facility operation is full power with question on calibration of nuclear instruments and resolution of read-out available to operators. Management is aware of problem and IE is following.

Do not have accepted QA plan meeting current requirements. Should be approved soon. Unit 3 would be better rated because PASNY does better than Con Ed.

Upper Management (corporate) attitudes continue to limit effectiveness of site management.

Continue to inspect and observe with highly competent and experienced inspectors. The trend toward more inspections with less competent inspectors is dangerous. Also, continue design reviews by highly competent NRR personnel - also tighten standards and codes, and operator license examinations.

Docket No.: 50-309
Site: Maine Yankee

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

The plant is very clean - it shows pride in ownership and is indicative of happy people working at a good plant.

Have recently approved QA plan - upgraded to current standards. Becomes effective 8/16/77.

Recent change in station superintendent - no significant changes in safety expected.

Docket No.: 50-245
Site: Millstone

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

More safety awareness.

New security fence and procedures.

Re-evaluations have been made and design changes implemented in plant power distribution and emergency power systems.

Review of inspection findings, LERs and operating record would support this judgment.

New QA organization seems to be slightly more effective.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Large public interest in events taking place at this facility. Have a new plant superintendent.

Unit 1 is a BWR which is old - these items combine to cause a lower rating for Unit 1 than Unit 2.

Millstone site has three reactors, operating BWR, operating PWR, under construction PWR - all are by different vendors - all of different "era" - the operating reactors are, relatively, independent (as compared to a multiple unit site with the same generation of reactor from the same vendor) in their inherent safety characteristics.

Reliability of emergency has turbine, acceptance of the feedwater injection system as a high pressure ECCS system. Plant lacks a lot of separation and fire protection systems. Radwaste system undersized.

See answer to Question 28.

See p. 23.

Inter-relationship between diverse units at single site.

Docket No.: 50-224
Site: Nine Mile Point

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

There were some old fossil people managing and operating this plant - they don't have the nuclear ethic yet.

See question 69.

This is a plant of older design but the early engineering was of a high quality and excellent plant layout and construction. Onsite plant support (other than operations) lacking in numbers of people. Plant lacks system separation and a real high pressure injection system. Excellent security program.

Approach to operations of plant have been conservative. Plant staff has been stable.

Nine Mile also considerable operating experience, and a reservoir of experienced BWR operators (from Fitzpatrick which has until recently been operated by the Nine Mile licensee and which "leases" its operators from Niagara Mohawk until it trains its own).

Corporate engineering role in maintenance activities.

Docket No.: 50-219
Site: Oyster Creek

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Imposition of new operational procedures and facility record maintenance system has improved safety.

Installation of storage facility to house torus chromated water - and permit draining of torus.

QA program has been more fully implemented. New storage facilities, new document control center becomes operational.

Substantial upgrading of QA has been, and is, in progress.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Security should be upgraded, i.e., increase capabilities of guard force and surveillance equipment.

Upgrading of requirements, imposition of environmental T.S.

See question 69.

An early generation BWR - its age and generation made it different in inherent safety from facilities - and facility management has been less than willing to endorse in principle a comprehensive management control system - they conform as required rather than aggressively prosecute.

This plant received a poor design review as demonstrated by logic system inadequacies, recently found. Plant was built at minimum cost. Radioactive waste and fire protection are inadequate. Plant lacks system separation.

Management at corporate level have a first-hand technical and working level knowledge of the plant.

Docket No.: 50-277
Site: Peach Bottom

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Plant radiation levels have been increasing with time. Design and staffing of plant appear to have not been capable of handling this change. Management has been slow to take large step changes to correct problems.

Back to back overhaul/upkeep periods for units 2 & 3 appear to have produced a tired operating group prone to error.

Careless operations and poor maintenance.

Corrective action taken to repair core spray line cracks, feedwater spargers and nozzles and control rod drive return nozzle.

Licensee made significant effort to reduce routine radioactive release from reactor building vents through equipment repairs.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

See question 69 and 28.

QA program not upgraded to current standards. Security not upgraded. Many repeat items of noncompliance. Least safe site in Region I! Poorest management!

Quality of people (i.e., technical educational level) that are operating a plant and the type of organizational structure they are placed in can have a significant impact on safety.

Higher number of inspections due to proximity to regional office.

Recent management meeting with the President - expect to determine by scheduled inspections in the next 30 days if significant improvements were made.

Plant management exhibits an appearance of attempting to "control" NRC inspector access thru continual escort - general attitude appears to be one of compliance as required instead of an aggressive prosecution of management controls.

The problem with this plant is that it is a big BWR - by definition, they will have problems unless they have a good op. staff. PB does...

Upgrading of requirements upon this licensee, particularly in cases of security and QA.

Docket No.: 50-293
Site: Pilgrim

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Improved corporate management. Improved radiation management at site.

Due to instability in plant management. Drift due to lack of management direction.

Refueling outage.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Generation of design may be the overriding factor for this early generation BWR.

Have experienced a number of station manager changes.

Recent change in corporate rad protection and all old fuel is being removed. Significant improvements in reducing effluents and worker exposure expected.

Several changes in upper level management, some instability because of changes.

The cleanest BWR in the country.

Docket No.: 50-272
<u>Site: Salem</u>

<u>Answers to Question 17 (If a change to safety level occurred, please describe it briefly):</u>

Relatively new plant. Still has growing pains. Needs close attention (by IE) to assure appropriate improvements are made.

Power ascension testing revealed problems that were corrected by management, both in hardware and procedures.

<u>Answers to Question 18 (Are there other things we should consider about the safety of this plant?):</u>

The plant control room was designed in-house - it is a disaster waiting to happen.

In startup phase. Have had a number of problems. This can be due either to poor system <u>or</u> poor management <u>or</u> the "normal" failures when new systems are placed into service.

Design of controls with back-lighted pushbottons results in operator data assessment problems, especially when lights are burned out. Management is aware of problem and IE is following up.

New plant - recently completed full power testing - plant still in early operating phases.

Docket No.: 50-289
Site: Three Mile Island

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Increased security by addition of fence surveillance, guard force and search equipment.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Pay close attention to performance of newly assigned station and unit superintendents.

This is the first designed B&W plant of this generation. Construction was largely accomplished without aggressive prosecution of nuclear management control. Operation is conducted under strong management control.

The licensing of Unit 2 in 10/77 will have an impact on the site/corporate staffs. In all probability the overall safety may become worse over the next year due to this increased workload.

2nd plant in startup placed some additional "drag" on operating facility equipment and manpower.

Docket No.: 50-271
Site: Vermont Yankee

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Management experience and depth is increasing.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Have upgraded QA plan which becomes effective 8/16/77.

Frequent changes in plant superintendent - has resulted in slight degradation of management controls.

Very clean.

Public interest in events at site.

Docket No.: 50-029
Site: Yankee Rowe

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Issuance of standard Technical Specifications.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Plant is very small and very isolated - virtually no health hazard to the public exists.

Old plant Tech Specs. New, upgraded QA program doesn't become effective until 8/16/77. See questions 67 & 69.

Docket No.: 50-259
Site: Browns Ferry

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Attention to QA principles seems somewhat less.

a. More experience and exposure of plant personnel = improved safety and operation.

b. More inspec 'ons by NRC.

c. Plant management changes.

Improved response to alarms - enforcement meeting.

More safety awareness.

In fire protection.

Answers to Question 18 (Are there other things we should c -sider about the safety of this plant?):

Core performance analysis, qualifications of technicians and mechanics who maintain safety equipment.

Docket No.: 50-302
Site: Crystal River

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

More safety awareness.

Improved Adm. control. Improved Opera. awareness.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

None.

Docket No.: 50-325
Site: Brunswick

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Some improvement in administrative controls. More experience by operating staff.

New management and experience.

Management seemed to become more aware of events at plant.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

The training or experience of senior site management - none of the top three have had SRO training in BWRs. The plant has had a very high personnel turnover rate. Consequently, the staff is young for the responsibilities needed. Corporate management apparently still has not faced up to what this inexperience costs in safety and efficiency. They appear to believe they are being over-regulated.

Qualifications of technicians and mechanics that maintain safety equipment.

All pre-op testing must be completed prior to licensing.

Docket No.: 50-261
Site: Robinson

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Licensee has made increased site commitment to QA/QC.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Low number of LERs reflects attitude or reporting only items that are conspiciously reportable. Licensee impedes IE freedom of movement and access at site. No information freely given. Definite attitude of do only what is required.

Qualifications of technicians and mechanics that maintain safety systems.

Docket No.: 50-321
Site: Hatch

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Continued upgrading of Adm 8 QA control

Operating experience.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Qualifications of technicians and mechanics that maintain safety equipment.

Docket No.: 50-269
Site: Oconee

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Change in Operating Superintendent should improve situation in next few months.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Qualifications of technicians and mechanics that maintain safety equipment. Maintenance of test equipment.

Docket No.: 50-335
<u>Site: Saint Lucie</u>

<u>Answers to Question 17 (If a change to safety level occurred, please describe it briefly):</u>

Improved due to increased operations, etc., experience of plant personnel over time period involved.

<u>Answers to Question 18 (Are there other things we should consider about the safety of this plant?):</u>

This plant has more than average number of LER's. I believe this is due to Licensee's determiration to report all possibly reportable items rather than poor performance.

Docket No.: 50-280
Site: Surry

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Due to degradation of steam generators.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

None.

Docket No.: 50-250
Site: Turkey Point

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Safety may be slightly worse due to steam generator degradation.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Qualifications of technicians and mechanics that maintain safety equipment.

Docket No.: 50-331
Site: Arnold

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Improvement in administrative control and QA program.

New plant superintendent. Stronger enforcement action - increased inspection effort.

Management change.

More awareness regarding significance of personnel error.

Steady improvement in management controls and quality of onsite staff. Increased attention by engineering and corporate office.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

See page 13.

Docket No.: 50-155
Site: Big Rock Point

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

QA program implementation continuing resulting in an improved plant safety level.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

An original BWR - Design, operation relatively uncomplicated. Closeness of opera.

General safety of older plants.

Plant personnel qualifications have improved (technical capability) recently.

Docket No.: 50-315
Site: D. C. Cook

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Increased number of personnel errors and procedural violations occurred during 1977.

Demands placed upon personnel and management due to Unit 2 startup, fire protection and security have brought a decrease in attention and review unit 1 is given. Events are occurring that would not have a year ago.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

One plant in operation the other in startup. Plant using standardized Tech Specs.

Resident inspector stationed there 74-77.

Design, its newer with greater indepth protection.

Docket No.: 50-010
Site: Dresden

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Improved training program, and improved QA programs.

Better housekeeping, more attention to detail.

Poor operation, instrumentation problems.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

See page 13.

See Zion comments.

U-1 is a 200 MWe plant while U-2&3 are 800 MWe each - U-1 will never receive priority at the management level - One should also consider the manpower availability on site.

Docket No.: 50-305
Site: Kewanee

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Resident inspector assigned 74-76.

Very stable and competent plant management; overall good operating performance; strong safety attitude.

Docket No.: 50-409
Site: LaCrosse

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Fuel degradation.

Improved QA program.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

See page 13.

Part 115 plant (AEC developmental reactor) small utility - limited technical staff with minimal corporate backup - difficult to absorb costly NRC regulations.

Docket No.: 50-263
Site: Monticello

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

None.

Docket No.: 50-255
Site: Palisades

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

QA program implementation continuing resulting in an improved plant and safety level.

Improved attention by management toward more timely correction of problems.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Utility constantly confronted by intervention - legal challenge from the outside.

Effectiveness of management controls. Resident inspector assigned 74-77.

Docket No.: 50-266
Site: Point Beach

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

This plant is of older design with its management attitudes it would be above exceptional if designed like present day.

Desciplined staff, well motivated, pride which includes their ability to positively criticize the NRC in matters which distract from their ability to conduct their plant operations.

The exceptional strength of plant management in all areas. The total team effort in all matters - the excellence of all personnel attitude in regard to safe plant operation.

Docket No.: 50-282
Site: Prairie Island

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

The technical staff is closely integrated with operations and maintenance. This helps resolve problems before safety concerns develop and provides good information where failures have occurred.

Docket No.: 50-254
Site: Quad Cities

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Improvement in training program, improved QA program, improved radiological program.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

See Zion comments.

The licensee has been "overinspected" by NRC and state for the past 2 or 3 years. The plant cannot operate at design load because of an agreement with the state to operate with a closed cycle cooling canal, after the plant was built as designed for once thru cooling. This affects plant operation and also attitudes of operators.

See page 13.

Docket No.: 50-295
Site: Zion

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Apparent PWR attitude of personnel resulting from marginal management.

Safety reduced as evidenced by loss of DC power and by- passing all pressurizer level channels in 1977. Inadequate management controls.

Continued deterioration of management controls.

Nonconformance with technical specifications; failure to adhere to administrative procedure; failure to adhere to operating, emergency, and test procedure; inadequate procedure; operator error; poor overall operating performance; weak overall management.

Procedures improved; administrative procedures improved; better training.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Lack of management. Ability to discipline employees for operator error/ carelessness.

Management - union interface and its effect on selection of personnel and discipline. Attitude and support from support engineering and corporate management to resolve operating equipment problems. Corporate management involvement in plant operations. Corporate management attitudes and followup.

Part of a complex nuclear commitment which carries with it the management problems associated with "bigness". Stability of staff a continuous problem.

Overall attitude regarding safety is not strong. Lax operaring performance and attitude.

Adequacy of training program; number of personnel errors resulting in significant problems.

Docket No.: 50-313
Site: Arkansas

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Cable penetration barriers and fire proofing of essential and safety cables. Improvement in procedural controls.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Unit 2 which is soon to be operational will be managed by the same site management as that which controls Unit 1. I feel this practice considerably dilutes management control over these plants.

Upgrade technical specifications to standard T/S.

Docket No.: 50-298
Site: Cooper

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

None.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

None.

Docket No.: 50-267
Site: Fort St. Vrain

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Cable separation, training program, penetration fire barriers and flammastic on essential and safety related cable. Experience of operating personnel as operation of plant continues.

Based on an IE inspection the licensee has recently had to review the setpoints of his safety systems to determine that instrument and calibration inaccuracies are adequately accounted for in the selected setpoints.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

First of a kind - fits the category of a demonstration site.

The basic design and configuration of the HTGR introduces a completely different set of parameters and accidents to be considered in plant safety.

This is a one of a kind HTGR. The existing Regulatory Guides, standards, etc., do not apply to this plant. The existing Technical Specifications need to be completely revised.

Docket No.: 50-133
Site: Humboldt Bay

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Seismic modifications completed during past year.

Seismic modifications have been performed, feedwater sparger has been replaced.

Upgrading structures to new seismic criteria.

The plant has undergone an extensive outage to upgrade the structural integrity of the facility to limit seismic damage.

Plant shutdown for extensive modification in July 1976.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

The other matters I feel are necessary to consider are being pursued by NRR - they include adequacy of ECCS, single failure design and gaseous effluent treatment.

The plant would be hard pressed to meet any of today's criteria for nuclear plant safety.

No opinion.

New Technical Specifications.

Adequacy of seismic design, ECCS and reactor protection system. Results of analyzing these safety questions could change (significantly) my rating of overall plant safety.

Docket No.: 50-312
Site: Rancho Seco

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Overall plant safety increasing with experience of operations organization and management's understanding and knowledge of nuclear plant operations.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Not that I'm aware of.

Docket No.: 50-246
Site: San Onofre

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

QA program improvement.

Completed outage which improved their emergency power capability substantially.

Installed emergency diesel generator capacity to carry LOCA load coincident with loss of off-site power. Also, constructed concrete shield around containment vessel.

Extensive ECCS and seismic modifications have been completed.

Installation of onsite emergency power capability.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

This company should be studied to determine how and why their management has been so successful in instilling good safety attitudes and habits so uniformly thru their organization.

Docket No.: 50-344
Site: Trojan

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Equipment improvements in engineered safety features brought about by operating experiences.

Improving with experience as operating organization and management matures and gains nuclear experience.

QA program implementation onsite has substantially improved by identifying problems before they became issues or items of noncompliance detected by NRC inspectors.

Fire protection program is being implemented.

Improved attitude toward value of QA auditing and initiating corrective measures to correct recurring deficiencies identified from operating experience.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Active role of State of Oregon in attempting to regulate this plant could have an effect on safety - possibility of contradictory requirements and demands by federal/state agencies.

Docket No.: 50-285
Site: Fort Calhoun

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Site management at this plant is young and they are maturing and recognizing their safety responsibilities.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Top management (Board of Directors) have an anti-nuclear attitude which is upsetting to site personnel and management. Since there is a correlation between morale and job satisfiers, I am concerned about this situation. This concern is due to the fact that morale affects employee safety practices more than production.

Docket No.: 50-285
Site: Fort Calhoun

Answers to Question 17 (If a change to safety level occurred, please describe it briefly):

Site management at this plant is young and they are maturing and recognizing their safety responsibilities.

Answers to Question 18 (Are there other things we should consider about the safety of this plant?):

Top management (Board of Directors) have an anti-nuclear attitude which is upsetting to site personnel and management. Since there is a correlation between morale and job satisfiers, I am concerned about this situation. This concern is due to the fact that morale affects employee safety practices more than production.

LICENSEE PERFORMANCE EVALUATION

Teknekron, Inc.

WASHINGTON, D.C.

NUREG/CR-0110

LICENSEE PERFORMANCE EVALUATION

PHASE I REPORT

H.E. Chakoff

D.M. Speaker

S.R. Thompson

S.C. Cohen

Manuscript Completed: May 1978

Teknekron, Inc.
4701 Sangamore Road
Washington, D.C. 20016

Prepared for
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Under Contract No. NRC-05-78-302

Teknekron, Inc.

ABSTRACT

A model was developed for the analysis of the performance of NRC's licensees. The model is based on identifying the distinctions between the licensee's facility, personnel, and management and the interrelationships between them. The application of this model and related methodology to available NRC licensee data permits the display of licensee performance in terms of temporal patterns that provide an understanding of performance quality and furnish an insight into the causal factors underlying this quality. In principle, the analytic methodology derived from the model can be applied to any licensee class; at present, except for operating power reactors, available data are relatively sparse. On the basis of the LER and 766 files, three nuclear power licensees in Region 3 were analyzed with the result that previously suspected differences in performance quality became evident through the displays generated by the analysis. Management attitude and capability were found to play major roles in determining performance.

EXECUTIVE SUMMARY

In order to assist NRC's Office of Inspection and Enforcement in ensuring the safety of licensee operations, we developed a methodology to analyze licensee performance. This methodology utilizes an initial conceptual "model" of a licensee, in which the physical facility, the operating personnel, and the management are clearly identified as distinct entities. The model also explicitly defines the interrelationships among these elements by characterizing flows of information and control signals among the elements. Applying the model produces profiles of licensee performance. These performance patterns, which are displayed as a function of time, not only reflect the character of performance (relatively good or relatively poor), but also provide insight into the causal factors that underlie performance quality.

The model is applicable to all licensee classes. However, feasible application is limited by the data available for each licensee category. Currently, the data that exist in NRC files are most complete for operating power reactors. Because power reactors are the most complex of all licensees and because substantial data describing their operation are available, we initially tested the methodology on this category of licensees.

In the case studies, we analyzed three operating power reactors from NRC Region 3, including one considered to be a "good" performer and one a "poor" performer. All three were alike in terms of age and type of equipment. The analysis showed substantial differences between the performance patterns of the "good" and "poor" performers, especially in the clustering of causally related events. In both cases, it was clear that the willingness and ability of management to institute prompt and generic remedial measures was a major factor in performance quality.

A major finding of this study was that the content (not the quantity) of Licensee Event Reports (LERs) proved to be of considerable value as a performance indicator in the context of our licensee model. Testing the noncompliance data produced by NRC's inspection process against the model provided insight into how the content of the noncompliance data could be improved to enhance its value to licensee performance evaluation.

TABLE OF CONTENTS

Sectión | Page

TABLE OF CONTENTS (cont.)

LIST OF FIGURES

List of Figures (Cont.)

LIST OF TABLES

1.0 INTRODUCTION

In approaching this project, we tried to focus on significant aspects of "licensee performance" and how their analysis could best support NRC's goals. We concluded that "performance" is fundamentally grounded in the structure and operation of the licensee; to provide insight into why one licensee is different from another, we had to devise a way to examine the licensees' ability and willingness to operate the facility to carry out the public safety intent of NRC's regulations. Therefore, the first step was to develop a general concept of a licensee - a "model" - and then examine the available data to see what information could illuminate the elements of that model. We began with a concept of a licensee's operation and structure, not with the data that the operation and structure produce.

Two types of data - licensee event reports and inspector-reported non-compliances - give two views of how a licensee conducts his operations. By using the structure of our licensee model to analyze the content of the data, a picture of that licensee's capability and attitude emerged. We began to see apparent causes underlying the data. Because poor behavior does not always have severe consequences, we made no attempt to weight data elements. Neither did we count data elements, nor normalize them in any way. Using the content of the data as a source of attitude and behavior information made counting and normalizing unnecessary.

The results of this methodology take a non-numeric form. The licensee model and the way we used the data to illuminate the model's interrelationships suggested graphic profiles that show behavior over a period of time. We believe these profiles show the differences between licensees while still preserving their uniqueness and that they lend themselves to NRC's setting a "threshold band" above which performance is adequate and below which it is not. The methodology makes it possible to examine specific areas of a licensee's operation to pinpoint problem areas; it also enables a more comprehensive picture to be seen. Further, using licensee event reports and inspector-reported noncompliances to create separate profiles makes it possible to see the interaction between NRC and the licensee.

We believe that this report presents a valid and insightful performance analysis method. NRC needs a tool to analyze the performance of its licensees so that it can determine where to place its inspection emphasis to improve that performance. For this reason, we have used the term "licensee performance analysis." We think this name accurately reflects NRC's need for and use of such a tool.

Section 2.0 sets the stage for licensee performance analysis by linking it to NRC's mission and goals. Section 3.0 presents the FPM model and our methodology for using available data to analyze licensee performance. The fourth section shows that the methodology meets the requirements of the NRC Request for Proposal.

Section 5.0 sets out our proposed plan of action for Phase II of this program. The final section identifies a number of work areas addressing needs that became evident during the course of this study. Appendix A presents three case studies in their entirety. Reading the details of these case studies will give a full appreciation of the meaning of the performance profiles and the use of our methodology.

2.0 RATIONALE FOR LICENSEE PERFORMANCE ANALYSIS

This section discusses the factors involved in NRC's decision to develop a tool to analyze the performance of its licensees. We define "performance" and then discuss NRC's objectives in analyzing performance. NRC staff perceptions are closely interrelated with NRC objectives, and those perceptions will influence the ways in which NRC will use a performance analysis tool. Finally, we discuss prior performance measurement efforts.

2.1 DEFINITION OF PERFORMANCE

In this study, "licensee performance" is specifically related to elements that affect the level of risk presented by the licensee's operation. One assumption, basic to any program that regulates hazardous activities, is that compliance with regulations will maintain the risk at or lower than a level "acceptable" to NRC. Because of this assumption, one of our early definitions of performance included "...demonstrated compliance... with the regulations and the conditions of the license."

That early definition also included "the ability of the licensee to comply" as well as the "attitude of the licensee toward compliance." These two factors influence performance rather than being essential components thereof, but their inclusion recognized that unless attention were given to motivation and ability to perform, NRC could not fully understand the reasons for inadequate performance. NRC's Request for Proposal made it clear that the methodology developed must be able to distinguish between "good" and "poor" performers as well as provide insights into the "whys" of performance. NRC must have a tool with both these dimensions if it is to successfully remedy poor performance.

While "good" and "poor" performance are relative terms, we can say that a "poor performer" is a licensee who has more noncompliances or safety-related events than NRC feels he should have. This must be a subjective definition, since there can be no fixed threshold of noncompliances or events above which performance threatens public health and safety. But excessive noncompliances or LERs can indicate a lack of management controls, which, if widespread, could eventually threaten public health and safety.

Therefore, although the concept of performance remains closely linked to regulatory compliance, we did not restrict it to that criterion. In fact, we found that safety-related performance is more accurately analyzed and more meaningfully interpreted when seen as a multidimensional behavioral pattern rather than a numerical record of lapses from regulatory grace.

Thus, over the first phase of this study, Teknekron's working definition of performance has been:

> PERFORMANCE: Those patterns of behavior that show the ability and willingness of the licensee to conduct his operation to minimize the risk to public health and safety and to the environment.

2.2 OBJECTIVES OF LICENSEE PERFORMANCE ANALYSIS

During the early part of this study, the tentative objective of performance analysis was to identify "those licensees whose level of performance (as measured principally, but not solely, by compliance) may require improvement." As the study evolved, no findings of the case studies contradicted or were inconsistent with this objective. But the objective appeared incomplete: it did not include understanding the behavioral differences among licensees nor did it include identifying their levels of performance.

The methodology Teknekron developed makes it possible to compare behavior patterns of one licensee against those of another. A comparison might be expressed as: "Licensee A has been more effective than licensee B in eliminating facility conditions that can induce recurrent and causally connected events. It is clear that A's management is the more alert and responsive of the two, and that, on the whole, the potential risk presented by A is substantially less than that associated with B's operations."

We must emphasize that our methodology does not attempt or intend to rank licensees (within a given class) on any sequential or numerical basis. The method does, however, allow the relatively good and the relatively poor performers to be identified in a way that gives NRC insight into the reasons why these licensees are different.

2.3 PERCEPTIONS OF LICENSEE PERFORMANCE ANALYSIS

As part of Phase I, Teknekron met with a variety of people who will be affected in some way by licensee performance analysis. The perceptions and feelings of these people should be recognized and accounted for as much as possible if this program is to be most useful.

The perceptions of NRC personnel are critical. We met with headquarters staff and each of the regional directors; we sifted through several documents that expressed NRC viewpoints and concerns. Several of these concerns were related to earlier NRC attempts at performance evaluation; Section 2.5.1 briefly discusses one of these earlier attempts. The view of headquarters and regional personnel toward licensee performance analysis are discussed separately below.

NRC's licensees will obviously be affected. To obtain the licensees' views, and what they perceive such an assessment might mean, we met with the secretary of the Atomic Industrial Forum (AIF) and also with the AIF's Ad Hoc Committee on Inspection Practices, where representatives of four power companies and two NSSS suppliers were present.

Finally, to complete the spectrum of perceptions, we obtained the intervenors' viewpoint in discussions with the Natural Resources Defense

Council. While intervenors are not directly affected by licensee performance analysis, they may be interested in its potential use in their representation of one public viewpoint, a factor that may affect the form taken by public release of performance analysis results.

2.3.1 Headquarters Staff

As is natural in any group of people, the aims and inclinations of individuals vary. But there was more agreement than disagreement on a number of major points. First, some analysis of performance will be conducted, because it is basic to focusing the resources of the inspection program efficiently and effectively, and it may also provide a way to link enforcement action to the weak spots in the licensees' behavior. If it is properly structured, performance analysis may also help to improve relations between NRC and the licensees, so that the goal of adequately protecting public health and safety can be more easily attained. These basic feelings about the purposes of the program influence its form, and a majority of the headquarters staff lean toward the idea of NRC-established "thresholds" of acceptable performance rather than classifying licensees into groups. The "threshold" concept is consistent with the NRC's regulatory mandate to require levels of safety that adequately protect the public.

Nearly everyone agreed that licensee capability and attitude are important indicators of performance - if data can be obtained that reflect those qualities. "Management inspections" are to be reinstated, and they may help provide this data. The actions a licensee takes to investigate his own problems, the actions he takes to correct them, and the effectiveness of those actions are indicators that reflect both attitude and capability. Some of the staff felt that the perceptions of the regional personnel should be a potential indicator, and others felt that occupational exposure and effluent release data should be included.

A few other views were less widely held, but they indicate that the staff feels a need to move ahead in devising a workable analysis tool. Nearly all agreed that numerical counts of noncompliances and reported licensee events are not valid performance indicators, because counting implies the need for a weighting factor related to severity levels. There has been no agreement that any weighting scheme devised so far is completely satisfactory. Similarly, most Headquarters staff believe that the issue of normalization (by inspection hours, modules completed, or inspectable requirements) is difficult; that issue may well dissipate with the advent of the resident inspector program. Since normalization was an attempt to handle regional differences and variations in time spent with different licensees, the need for normalization may disappear if the analyses for each region are kept separate.

2.3.2 Regional Staff

Teknekron held separate discussions with each regional director and his staff. Despite our attempts to follow a similar format, each conversation took a slightly different turn, and not all topics were covered in all discussions. But the perceptions on a core of topics that were covered in all the discussions show some views that are quite similar to those of the headquarters staff as well as a few that are quite different.

All the regions stated that some sort of performance analysis should be performed. But a number of regions felt that they "know" which licensees are "good" performers and which are not. They also agreed that regional differences are substantial, including style of management. The regional personnel feel that they are closest to the day-to-day operation of the licensees, and that any method that is developed must accomodate regional differences and not be simply a tool for use by headquarters.

Regional feelings on performance indicators varied, but they centered around the idea of management responsibility. All but one of the regions mentioned that counting LERs and noncompliances was inappropriate. Uneasiness about counting stemmed from the feeling that human errors and adequate management response in correcting those errors are more important factors. Most of the regions stressed that ability and attitude of the plant manager was a major force in shaping the plant's performance. More than half the regions said that some form of subjective evaluation should be included; more than half also felt that repeated noncompliance was a good indicator because it revealed poor management response.

A majority of regions supported the concept of performance thresholds, but the idea of ranking licensees produced several negative reactions. We could find no agreement on normalization of noncompliances. Some felt inspection hours should be used, and other regions had no fixed opinion. Three regions stated that normalization may be unnecessary, particularly in light of the resident inspection program.

2.3.3 Licensees

It is safe to say that the nuclear industry is nervous and suspicious about NRC's reasons for wanting an analysis tool. Their feelings have two bases. First, the industry feels beleaguered by a negative attitude toward nuclear power as expressed in public reaction, legal intervention, and in press coverage. They feel that this negative public attitude will almost certainly result in the possible misuse and misrepresentation of _any_ assessment method, and because of this, no method can receive a fair trial. Second, they assume that an ability to determine where emphasis is needed implies ranking or comparatively rating licensees. The strong feeling against ranking, even in such terms as "A, B, and C" or in quadrants as used in the TRW* report - not to mention a 1-60 list with attached scores - is intimately linked with industry's fear of public reaction and public (mis)use.

*Discussed in Section 2.5.1.

On the more positive side, licensees enthusiastically welcome the concept of NRC-established thresholds for acceptable performance. The threshold concept clarifies the relationship between the NRC and the licensee and potentially offers a clear goal to be achieved. If the thresholds are mutually acceptable, the licensees realize that they should perform at an acceptable level both for their own good and for the good public perception of the industry.

A few other comments illuminate the current relationship between NRC and licensees. The licensees perceive strong differences in management approach among the NRC regions, and in some cases they feel that the inspection process results in little if any increase in safety. But they also feel that reduced inspection effort by NRC would have little or no effect on safety although it could function as an incentive.

Licensees also feel that in many cases the inspection program does not help them find particular areas of weakness because it seldom helps locate the causes of noncompliance.

Finally, the licensees are concerned about the possible impact of licensee performance analysis on the licensing process. If the analysis process were applied to a reactor under construction, licensees feel that a poor level of performance in the construction stage could make it difficult for that reactor to be licensed to operate. Increased difficulty in obtaining an operating license places in jeopardy the time and money already spent in construction.

2.3.4 Intervenors

The Natural Resources Defense Council's (NRDC's) feelings about licensee performance analysis must be placed in the context of its position on nuclear power.* Broadly stated, its position is that nuclear power plants should not be built or operated, first because licensees cannot be trusted to build and operate plants safely by themselves, and second, because the regulatory system does not adequately oversee the licensees to assure that they meet specifications and license conditions. Since NRDC can deal more directly with NRC's regulatory role than it can with a multiplicity of licensees, the thrust of many of its comments was directed at evaluating the effectiveness of the inspection program. NRDC feels that measurement of I&E effectiveness is basic to encouraging adequate licensee behavior.

In NRDC's view, a fundamental question is not whether performance analysis is feasible, or what method should be used, but whether the public will believe the results if they show that licensee X is good. This stems from its perception that no licensee is performing adequately, at least in part because the regulatory program cannot make him do so. On the other side of the coin, NRDC will not attack an analysis methodology because it feels that adequate regulatory control is lacking.

2.4 USES OF LICENSEE PERFORMANCE ANALYSIS

The primary user of a performance analysis tool will of course be the NRC. Based on the perceptions of NRC personnel and on the objectives of identifying those licensees whose performance must be improved and analyzing why one licensee differs from another, we believe that licensee performance analysis can be effectively used to:

- Allocate I&E Resources

  The case studies we have performed (all in Region 3) demonstrate an extremely wide range of licensee

*We contacted two intervenor groups but held discussions with only one. We felt that the intervenor's viewpoint should not be ignored, because public perception is a factor of concern to the licensees; we also feel that the intervenor's view should not be a major factor in shaping the final product. But a *caveat* is necessary: NRDC's views may not be those of other intervening groups.

performance quality. The managerial quality of the best performer strongly suggests that this licensee is highly motivated to maintain an excellent operation (responsible and highly compliant) and would do so even if the NRC inspection program did not exist. Poorer performers obviously require more of NRC's attention.

By analyzing the relative quality of operation of licensees in a given class, I&E can then allocate its inspection and other resources to focus on upgrading the poorer performers, while possibly devoting less inspection effort to those licensees who are more self-motivated. Further, in the case of the poorer performers, licensee performance analysis will permit NRC to identify those facility systems that have experienced repeated causally related events and to concentrate on those systems that have the greatest safety implications. Using this method of analysis, NRC can identify major organizational causes of system breakdown, and the onsite inspector can concentrate his efforts on the cause rather than the effect.

- Assess the Likelihood of Future Events

A sustained sequence of causally linked events in a single system suggests a higher probability of future events occurring in the same system (within a given period) than does the absence of such a sequence. The reason for this rests primarily in the quality of facility management that a sequence of events implies. In well-managed operations, repetitive events occur in smaller numbers because the cycle is truncated by generic correction of the problem. (For example, if seal leaks have occurred in similar equipment on two or three occasions, management will order all such equipment to be inspected and all questionable seals replaced.) Thus, a low incidence of causally linked events suggests good management; good management, in turn, characteristically designs and carries out effective inspection and maintenance programs that reduce the likelihood of event occurrence. In less well managed facilities, where the probability of future events is relatively greater, it does not necessarily follow that the event, if it indeed occurs, is causally linked to the sequence of past events in the same system. It may be causally linked to a sequence of past events or it may be unrelated. Causal linkage supports the earlier remarks about management quality.

- Support Enforcement Action

  The imposition of sanctions against a licensee can legally take place only if the licensee is not in compliance with legitimate requirements. Therefore, his performance patterns, as developed through the FPM methodology, cannot themselves be used as the basis for enforcement action. But once NRC has decided to bring an enforcement action on regulatory grounds, licensee performance analysis can be used as a guide for determining the severity of this action. For example, a large number of causally related events occurring within a given time period might suggest a more severe penalty than would the occurrence of a small number of random events within the same period.

- Identify I&E Regional Differences

  Some aspects of our analysis are particularly sensitive to the ways in which I&E inspection actions are implemented and to the ways in which reactive inspections are triggered. We believe that further case studies will identify and define significant regional differences in the inspection process.

2.5 BRIEF ANALYSIS OF RELATED WORK

As part of Phase I, Teknekron examined other NRC efforts dealing directly or indirectly with analysis of licensee performance. Three documents are particularly pertinent to this project, since they have helped to focus the views and attitudes of I&E personnel on the acceptability and usefulness of various methods of analyzing licensee performance and, to some degree, on the role the inspection process itself. These three documents are:

- "A Statistical Evaluation of the Nuclear Safety-Related Management Performance of NRC Operating Reactor Licensees During 1976." This is an NRC-generated report dated February 1977.
- "Phase I Report: Utility of Incentive Systems for Licensees." This report was prepared by TRW under NRC sponsorship and is dated October 1977.
- "Benefit Cost Analysis of the Trial Inspection Program Involving Statistical Sampling Inspection Techniques Conducted at Metropolitan Edison Company's Three Mile Island Unit 1 during the Period July 1, 1975 to June 30, 1976." This is an NRC-generated report, dated January 1977.

This discussion briefly summarizes Teknekron's views on these efforts and shows how they influenced our work.

"A Statistical Evaluation of the Nuclear Safety-Related Management Performance of NRC Operating Reactor Licensees During 1976"

This report describes a licensee performance assessment methodology based on the statistical treatment of noncompliance counts by category, numbers of LER's submitted, and other measures that are ultimately combined into a single index (Z score). Its intent is to arrive at a numerical rating that realistically reflects licensee performance, since the better performer is assumed to incur fewer noncompliances and issue fewer LERs. This statistical methodology defines one view of "licensee performance." This report has stimulated considerable comment within NRC, much of which has focused on certain specific issues, including:

- The problem of developing a broadly acceptable relative weighting system for the various noncompliance categories: violations, infractions and deficiencies.
- The question of whether differences in the stringency of technical specifications applicable to different licensees may in themselves affect performance quality. This factor could prevent uniform application of the methodology.
- Licensee performance evaluations expressed as single numbers (as aggregates of several factors) inherently lend themselves to the relative ratings of licensees. NRC I&E generally feels that relative rankings of licensees are likely to generate misleading impressions and are therefore undesirable in terms of the interests of both industry and the public.
- A relatively high number of LERs may not necessarily indicate poorer performance: it could mean that the licensee is overly conscientious in his interpretation of what is considered reportable.

Overall, NRC's development of a statistical methodology has proven valuable in illuminating factors specific to this approach, as well as others that are largely independent of the particular evaluative method used. One of these latter factors is the effect of performance assessment on the licensee (will it motivate him to improve the quality of his performance, or might it have the reverse effect?). Another is the clear recognition that any evaluative approach should, to the degree possible, be based only on those performance factors that are within the licensee's control.

Review of both the NRC statistical approach and the commentary generated by it within the agency influenced the direction we took in developing our own licensee assessment methodology. It appeared that even if the statistical method could be refined to the point at which most of the specific issues were resolved, it was not designed to provide the insight into licensee performance (an understanding of the reasons for performance quality, as well as performance assessment) required by the RFP. This led us to a different approach.

"Phase I Report: Utility of Incentive Systems for Licensees"

This TRW report ably identifies several aspects of the NRC enforcement process that seem to offer less-than-optimum incentive to improve performance. But the concept of the TRW report of great value to our study was that licensee performance reflects a combination of attitude (willingness/desire to comply with NRC regulatory requirements or to improve the quality of operation), and capability (managerial and technical ability) to achieve compliance and improved operating quality. The first factor - attitude - relates to licensee motivation; the second - capability - relates to his capacity to translate his motivation into action.

The TRW report presents a graphic display classifying licensees who (at least theoretically) possess different attitude/capability combinations into four quadrants of "performance space." One quadrant represents good attitude/high ability, another good attitude/low ability, and so forth. In TRW's study context, this classification helps identify the forms of NRC enforcement/incentive actions that are appropriate to the attitude/capability combinations licensees exhibit. TRW's classification is of considerable interest to us because our methodology analyzes performance through its controlling causal factors. We were able to build on TRW's "performance space" concept by attempting to use performance indicators to discover causes, not only as measures of performance.

"Benefit Cost Analysis of the Trial Inspection Program Involving Statistical Sampling Inspection Techniques Conducted at Metropolitan Edison Company's Three Mile Island Unit 1 During the Period July 1, 1975 to June 30, 1976"

In Section 3.4.2, we consider statistical sampling as a possible means of analyzing the performance of classes of licensee for which the existing data are too sparse to permit individual analysis (materials licensees). For this reason, this report is of interest to us.

The Statistical Sampling Program (SSIP) was conducted as an experimental project to determine whether it was feasible, through the use of a statistical sampling inspection methodology, to establish confidence levels for licensee compliance with all requirements. Three strata of inspectable regulatory requirements were established, based on how closely the requirements were related to safety.

The authors of the report argue against further development of the SSIP on several grounds:

- Since the SSIP relies primarily on record audits and hardly at all on direct observation, an inspector might miss an important safety-related noncompliance item.
- Random sampling does not give the inspector an adequate overview of the quality of the licensee's operation.

- The SSIP is not cost effective. The average number of man-days required to identify a noncompliance are about 50% higher than under the regular inspection program.

Although the report does not favor extending the SSIP effort, we do not believe that sampling techniques should be completely dismissed. They could, for example, be independently applied in conjunction with the MC-2515 process as a check of the regular inspection program. Also, inspectable categories could be established on a system rather than a modular basis to ensure that no system having significant safety implications is ignored. This would require that samples be drawn from each system population of inspectables.

3.0 METHODOLOGY OF LICENSEE PERFORMANCE ANALYSIS

This section is central to our report. It presents a model of licensee structure and operation and describes how we use that model to analyze the patterns of a licensee's performance. We discuss in detail the types of data available, and how our methodology uses the data. There is a brief discussion of how performance analysis may be related to the performance appraisal team program. The section concludes with a determination of the licensee classes for which performance analysis is feasible.

3.1 GENERAL CRITERIA

When this study was first planned, before any analysis method had been developed, we felt that any approach to analyzing licensee performance must satisfy certain key criteria in order to be both practical - meaning that it can be readily implemented and that the results can be easily interpreted - and useful - meaning that the results will support NRC's safety-related mission. These criteria are:

Practicality

- The methodology should use available data where possible and should permit other data to be readily obtained.
- The methodology should be easy to apply.
- The methodology should be free from ambiguity, both in using data and in interpreting results.
- The methodology should use data that are related to or reflect safety factors.
- The methodology should not strain NRC's resources.

Usefulness

- The methodology should produce results that permit both absolute and relative analysis.

- The methodology should permit improvement - for both the licensee and NRC - to be assessed from one analysis to the next.

- The methodology should reveal patterns of compliance and non-compliance.

The criteria for practicality generally concern whether a method is feasible to use. But as this study proceeded, it became clear that the results of the analyses will be released in some form to the public; feasibility must also consider whether the analysis method is acceptable to the nuclear industry and to intervenors. A licensee analysis methodology may be highly useful to the NRC, but if it is inherently unacceptable to major interest groups, NRC's credibility as an objective agency will be impaired and any benefit of applying the methodology might well be outweighed by adverse public reaction. Potential public reaction was one of several factors that led us to adopt an approach geared to licensee structure and operation. This method permits licensees in a given class to be compared on the basis of "better" or "worse," but it is not designed to provide relative numerical ratings.

3.2 THE FPM MODEL

Performance is fundamentally grounded in the structure and operation of the licensee. We developed a licensee model to distinguish between "good" and "poor" performers and to gain insight into why one licensee differs from another. The structure of this licensee performance analysis model - the FPM model - is comprehensive and applies to the most complex category of NRC licensees, the operating power reactors. It can be modified to apply to other licensee classes as discussed in Section 3.4.1.

The conceptual design of the FPM model meets the general criteria for practicality and usefulness outlined in Section 3.1. The FPM model offers a reliable presentation of the licensee's performance pattern and an understanding of why this pattern has assumed its particular form. Understanding <u>why</u> provides insight into the causal factors underlying the performance of a given licensee and, when used on a comparative basis, identifies the reasons for performance differences among licensees in the same category. In addition, the FPM model shows licensee performance over time fo- two reasons: (1) the temporal relationships among events, inspection findings, and licensee responses provide significant insights into the nature and quality of licensee performance,* and (2) licensee performance is a potentially dynamic function that may improve or deteriorate with time.

FPM Model Structure

The model explicitly differentiates between two sets of parameters:

- Intrafacility relationships and interactions, such as those between management** and personnel.** These are critical determinants of licensee performance.

- <u>External indicators</u> of performance quality, such as inspector-reported noncompliances, other inspection findings, and LERs.

---

*In many instances, the meaning of certain patterns in these relationships may become clear only when viewed over an appreciable interval, such as year or two. We use a two year period in this analysis. But the model must also be sensitive to abrupt changes in the licensee's operation that may have significant implications.

**These terms, as employed in the model, have been assigned specific meanings that are defined later.

Figure 1 shows the structure of the model. The three circles designated "F", "P" and "M" represent the facility, personnel and management respectively. The arrows designated "1" through "5" symbolize the relationships among these entities. The arrows outside the rectangle and pointing away from it represent the external indicators of performance quality - noncompliances, LERs, and other inspection findings. In causal terms, the interrelationships within the rectangle are essentially within the licensee's control, and performance deficiencies traceable to these interrelationships can validly be attributed to licensee action or inaction. However, we recognize that some performance deficiencies could arise from causes that are not within the control of the licensee. These include certain external causes - a highly extreme case would be impact on the plant by a meteor - and inherently faulty components - components that are truly defective as opposed to those that became so through negligent or improper maintenance. Causes of these kinds are represented by the arrows to the left of the rectangle.

In this model, the terms facility, personnel and management have precise meanings:

Facility

This means the physical plant *in toto*, including not only the reactor and auxiliary plant, but also all instrumentation and test equipment. Thus the facility includes all physical components and structures relating to the licensed operation, but excludes associated human beings.

Personnel

This means all individuals who have a routine "hands on" relationship with any part of the facility. Personnel generally do not establish the procedures they implement.

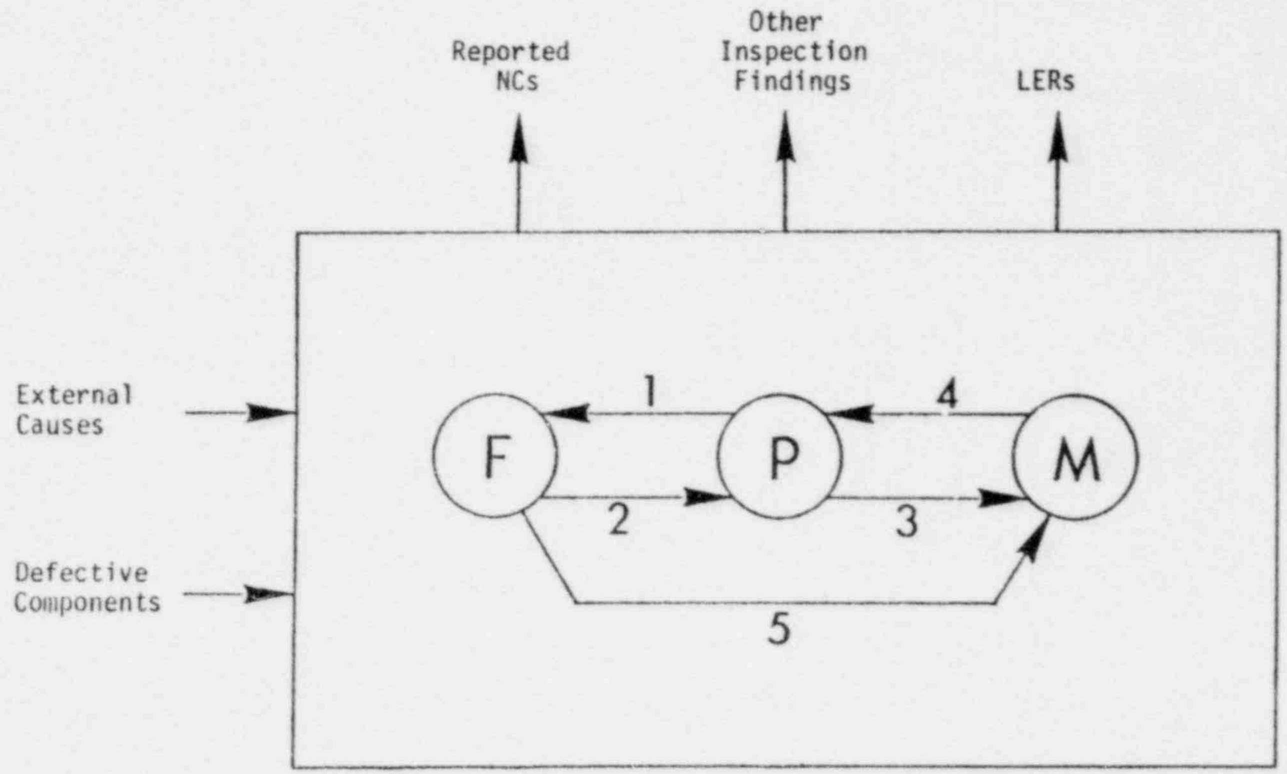


FIGURE 1
The FPM Model

Management

This means all individuals who are responsible for establishing policy, technical design, developing procedures, and training and supervising of personnel. These responsibilities implicitly include the assurance of facility safety. Management generally does not have a "hands on" relationship to the facility.

As stated earlier, the arrows within the rectangle represent direct interrelationships among the facility, personnel, and management. These interrelationships act as information channels, with messages flowing in the directions shown by the arrowheads. The message content varies considerably among the arrows. Briefly,

Arrows 1 and 2 are channels between the Personnel and the Facility

Arrow 1 represents all procedures and actions performed by personnel for the "hands on" operation, control, and maintenance of the facility.

Arrow 2 represents all information and data originating from the facility of which personnel should be aware; it includes all information and data that requires a "hands on" response by personnel.

Arrows 3 and 4 are channels between Personnel and Management

Arrow 3 represent personnel's reporting function with respect to management.

Arrow 4 represents the supervisory and administrative functions of management with respect to personnel. Note that this relationship is the sole avenue through which management can implement its responsibilities for acceptable facility operation.

Arrow 5 is the channel from the Facility to Management

This arrow represents all the information and data originating from the facility that makes management directly aware of normal operation and deviations from normal operation. The relationship between management and the facility is represented by only one arrow, because management control of the facility is normally exercised through personnel rather than through direct "hands on" operation.

This brief discussion simply identifies the broad character of the interrelationships and messages symbolized by the arrows. Our structural model is essentially simple; but a great deal of information about licensee performance is represented by the arrows themselves. A more detailed discussion of the interrelationships will help to understand the detail they can contribute to the analysis of performance.

Arrow 1

This arrow represents all the "hands on" activities that personnel perform in their operation of the facility. It includes both routine and nonroutine actions. These actions may be triggered by information and data that come from the facility via Arrow 2 or by directives to personnel from management via Arrow 4.

Arrow 2

Because it represents all information that the facility transmits to personnel, this arrow symbolizes routine data and also unscheduled or undesirable events or conditions. These non-routine events may reflect spontaneous failure within the facility, but they may also result from improper personnel action or the absence of appropriate action transmitted via Arrow 1. These two types of events directly represent the NRC LER Proximate Cause Code Categories of "component failure" and "personnel error."

Arrow 3

Arrow 3 represents the flow of information from personnel to management. Much of this information relates to the state of the facility as originally transmitted via Arrow 2. In addition to providing an information transfer route to management, Arrow 3 is also the channel through which personnel seek information from management.

Arrow 4

This information flow channel from management to personnel carries several types of communication, including written and verbal expressions of policy, intangible expressions of management attitudes, descriptions of administrative practice and procedure, and facility operating and other instructions. Arrow 4 also permits management to question personnel about the facility.

Arrow 5

This arrow carries facility information and data directly to management. In general, the information transmitted via Arrow 5 is included in the information carried by Arrows 2 and 3; Arrow 5 represents the independent check that management should have on the operation of the facility. It also reflects the awareness that good management should have. For example, management will sometimes observe significant facility operating indications that personnel has overlooked. Conversely, management may overlook those indications in some cases.

Using the FPM Model

In theory, the performance of a licensee can be analyzed and the reasons for his performance determined by examining only the portion of the FPM model inside the rectangle, if all the required internal data are available.

In most instances, the primary cause of a performance defect or deficiency can be assigned to one of the FPM circles, although it may first appear as an incorrect or missing component of the information flow along one of the arrows. Suppose, for example, that management had developed an incomplete or erroneous procedural plan for some operation and that this plan was transmitted to personnel via Arrow 4. Examining the plan as a component of the total information flow proceeding along Arrow 4 would immediately identify management error as the primary cause of whatever consequences stemmed from the use of the defective procedure. As another example, assume that personnel has transmitted to management (via Arrow 3) some significant information about facility operation that requires immediate management decision and response. The delay time, as measured by the interval between the transmittal via Arrow 3 and the management response via Arrow 4, as well as the appropriateness and adequacy of the response, provide an indication of management performance in this particular situation.

Unfortunately, complete and detailed internal information and data are generally not available to those outside the rectangle in the FPM model diagram (to NRC, for example); a reliable assessment of licensee performance cannot currently be made on the basis of these alone.* Because of this, performance analysis must depend, at least at present, on indicators that are external to the rectangle in the FPM model diagram, such as LERs, reported noncompliances and other accessible data. Other approaches to licensee performance analysis have stressed numerical counts of these indicators

*During the inspection process, some degree of awareness and understanding of this type of information may be acquired by observation. When the resident inspection program is established and operating, it is very likely that the inspectors will gain more insight into licensee performance in terms of the internal structure of the FPM model through more continuous exposure to the facility and its staff.

over defined periods of time. The FPM methodology emphasizes analyzing the content of LERs and noncompliance reports. When keyed to the integral portion of the FPM model, this content analysis provides insight into the nature of the licensee's performance pattern and the causal factors underlying it. We have presented the analytic results in a graphic form that permits immediate visual comparison of licensee performance patterns. The differences between the profiles of "good" performers and "poor" performers are clearly evident.

How we use the available data and analyze licensee performance are discussed in the next section. But we should note here that we have not used the severity of reported events and noncompliances in this evaluation. The discussion (in Section 2.5) of the statistical methodology developed within NRC pointed out the difficulty of finding a widely-acceptable weighting scheme, and we have chosen to weight violations, infractions, and deficiencies equally for the sake of simplicity in devising and initially testing the FPM methodology. This equal weighting is consistent with the fact that numbers of events or noncompliances are not central to the FPM approach.

While the numbers and magnitudes of events and noncompliances play no role in this analysis, we place considerable emphasis on the patterns of events and noncompliances over sufficiently long periods of time. Important pattern elements include event frequency, distribution, assigned cause, the occurrence of events that appear to have a common cause, and the number of repetitions of such events. Based on the limited number of case studies we have performed, these patterns appear to provide considerable insight into the quality of the licensee's operation and also into the personnel and management behavior that underlie that quality. We believe that the licensee performance patterns can be directly correlated with management and personnel actions symbolized in the FPM model, even though virtually no data on the information flowing along the numbered arrows is available for direct examination.

The design concept of the FPM model guided the analysis of the external data; this analysis preceded the construction of the graphic performance patterns. The FPM model also aids in understanding the implications of the performance patterns, once these patterns have been developed. The next section of this report details the procedures we used to analyze the external data (LERs and noncompliances), to construct the graphic performance patterns, and to interpret those patterns.

The decision to portray the results of licensee performance analysis through graphic patterns, rather than to attempt statistical manipulations of these results, was made soon after the model concept was first developed. We referred the question of graphic or statistical display to our consultant statistician before making a final decision. His view was that graphic patterns are inherently more revealing than numbers, particularly when a perspective of licensee performance as a function of time provides insight into the factors that determine performance. He felt that statistical treatment would tend to blur causal relationships that could be readily inferred from graphic displays. Furhter, the perceptions of NRC, licensees, and intervenors, discussed in Section 2.3, made it clear that ranking of licensees, made easier by numerical results, could threaten the acceptability of licensee performance analysis.

3.3 AVAILABLE DATA AND ITS USE

This section describes the data available for performance evaluation and how two kinds of data are used in the FPM methodology. First, we summarize the major types of data, the extent to which they are potentially available for each class of licensees, and the reasons for choosing LER and noncompliance data for use with the FPM model. Then, the type and extent of data contained in the LER file is discussed, followed by a thorough description of how we use LER data in licensee performance analysis, Noncompliance data is treated in a similar fashion. Potential problems in using each type of data a- discussed where appropriate.

The three case studies on which we tested the FPM performance analysis methodology are contained in Appendix A, but the introduction and conclusions drawn from the case studies are presented in this section to show the type of performance analysis produced by the FPM methodology. The section concludes with a brief discussion of the potential relationship of licensee performance analysis and NRC's Performance Appraisal Team.

3.3.1 Why Licensee Event Reports (LERs) and Noncompliances were Selected for Use in the FPM Model

Data describing the information that flows along the arrows of the FPM model are not readily available. But the NRC collects and makes available a variety of external data on its licensees. Occupational exposures, effluent releases, inspection findings, and events falling outside technical specifications are reported to NRC; Table 1 summarizes the type of data collected for each class of licensee.

Data on licensee events and on inspection findings in the form of noncompliances* are available in either written or computerized form for all classes of licensees. Effluent release and occupational exposure data are less widely available and in most cases are dupliacted in licensee event information. Thus, we believe that the information on noncompliances and licensee events is most useful in analyzing the performance of NRC's licensees, especially since this information covers a broad spectrum of licensee activities. Even more importnat, these data are computerized for three of the four major classes of licensees, an essential aid when analyzing substantial amounts of information for a sizable number of licensees. Computerization also places the data in a standard format, an advantage for ready comparison, and an evaluation methodology that can to some extent be computerized provides an element of uniformity in an evaluation process that must be sensitive to individual differences.

---

*As discussed in Section 3.2, we have weighted violations, infractions, and deficiencies equally. The term "noncompliance" covers all three categories.

TABLE 1

DATA COLLECTED FOR EACH LICENSEE CLASS

	Non-Compliance Data	Licensee Event Data	Effluent Release Data	Occupational Exposure Data
POWER REACTORS				
Construction	766 file	region; some in LER file		
Operation	766 file	LER file	HQ (file)	REIRS file
TEST & RESEARCH REACTORS	766 file	LER file	HQ (written)	REIRS file
FUEL FACILITIES	766 file	LER file		REIRS file
MATERIALS LICENSEES				
Special Nuclear Materials	766 file	region*		
Manufacturing & Distribution	766 file	region		REIRS file
Radiography	766 file	region		REIRS file
Waste Disposal & Collection	766 file	region		
Industrial	766 file	region		
Academic	766 file	region		
Medical	766 file	region		
Environmental	766 file	region		
Source Material Operations	766 file	region		
Shipping Casks & Transportation	766 file	region		
All Other	766 file	region		

*Not required to report to the Office of Management Information and Program Control (OMIPC); may be in LER file if the region sends report to OMIPC. This note applies to all materials licensees.

3.3.2 Licensee Event Reports (LERs)

3.3.2.1 Type and Extent of LER Data

Each licensee is required by law to report actual happenings that fall outside the bounds of his applicable technical specifications and license conditions. Since the summer of 1973, information extracted from these reports has been gathered in a computerized file of information known as the Licensee Event Report (LER) file, maintained by the Office of Management Information and Program Control (OMIPC). Operating power reactors and other production and utilization facilities report events directly to OMIPC, using an LER form. Other classes of licensees (including reactors under construction) report to the regional offices, which may or may not send the reports to OMIPC for coding and entry into the computer. The file is designed to accommodate events reported by all licensees, but the file currently contains data primarily submitted by power reactors since the beginning of 1969: for 1976 and 1977, only 137 LERs are in the file for the 93 test and research reactors, the 38 fuel facilities, and the more than 9,600 materials licensees, 78 construction deficiency reports are included for 28 construction sites in the same time period.

Instructions for completing the LER form were updated in July 1977, mainly to improve the specificity of information provided and to add new information on the licensee's reaction to an event. The LER form is shown in Figure 2. The 1977 revision added a cause subcode (item 13) and subcodes for components and valves (items 15 and 16). Codes were added to describe action taken immediately and in the future (items 18 and 19), the effect on the plant, the method used to shut down the plant (if required), and the length of time the plant was shut down (items 20, 21, and 22). A code was provided to indicate whether the event was publicized, together with a brief description of the event (items 44 and 45).* Little of the data now in the file is in this new format, since most licensees began using the new form at the beginning of 1978. But the new cause subcodes and the codes for action taken and planned will soon make it possible to sort the file more easily for data of particular interest, since it is relatively easy to sort a data file on a coded field.

LER Data Elements Used in Licensee Performance Analysis

Three major types of data elements now in the LER file contribute to the analysis of a licensee's (or a group of licensees') performance. First, and most basic, is information that identifies a licensee. Referring to the LER form in Figure 2, a licensee can be identified by code (item 2), by license type (item 4) or by docket number (item 7). The information provided by docket number and licensee code is duplicative; either can conveniently be used as the key element when searching the LER file for events pertaining to a particular licensee. License type is potentially useful in extracting data for a group of licensees for aggregate rather than individual evaluation.

The second set of data elements describes the event, an actual occurrence that results in activity outside the bounds set by license conditions and technical specifications. The event date (item 8) places the event in its chronological order in the eventual profile of licensee activities. The system code (item 11) identifies the system in which the event

*This revision also deleted a coded block used to identify whether an event was a "violation." The term "violation" was not specifically defined, and received varying interpretations by licensees. A licensee-reported noncompliance was not entered in the 766 file after October of 1977. Since our study period was 1976-1977, most licensee-reported noncompliances are included.

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: 1 6 (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

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

CON'T

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

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2

0 3

0 4

0 5

0 6

0 7

0 8

7 8 9 80

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

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

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

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0

1 1

1 2

1 3

1 4

7 8 9 80

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

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

PERSONNEL EXPOSURES

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

PERSONNEL INJURIES

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

LOSS OF OR DAMAGE TO FACILITY (43)

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

PUBLICITY

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

NAME OF PREPARER ______________________ PHONE: ______________________

FIGURE 2

LER Form

Teknekron, Inc.

occurred.* Seventy system codes are provided for reactors, as well as a code for "other systems" and a code for use when an event is not system-related. The system codes are the first two letters of the Nuclear Plant Reliability Data System codes, providing a potential linkage between this system and the LER file.

Content of LERs - not their potential consequences or quality - is of major importance in revealing licensee action and attitude. Item 10 on the LER form is a 504-character field containing a description of the event. This description includes the activity in progress when the event occurred, the circumstances leading to the event, the event itself in terms of which technical specification or license requirement was not met, any significant occurrences resulting from the event, and a further discussion of related or similar events if applicable. Only the concise 504-character description is entered in the computer, but more complete descriptions may be attached to the form and are available at OMIPC. Since data can be retrieved from the LER field by word search, only generally accepted terminology, abbreviations, and acronyms should be used. Where possible, an even greater degree of standardized wording will in the future make similar events easier to identify through a word search of the descriptive field.

The LER file also provides information on the cause of the event. The proximate cause code (item 12), the cause subcode (item 13), and a 360-character field (item 27) in which the cause and corrective action taken are described provide the major portion of the data for analysis of the cause of an event. Six cause codes are provided, covering (1) personnel error, (2) design, manufacturing, construction/installation, (3) external causes, (4) defective procedures, (5) component failure, and (6) other

*In any facility, systems are the common point of origin of events. Events in the same system may have a common cause. Causally-linked groups of events and repeated events are important elements in a licensee's performance pattern. Section 3.3.2.2 discusses these points more fully.

causes, for use when no other category is applicable or the cause cannot be determined. The cause subcode defines the cause more specifically when the proximate cause of the event is personnel error; design, manufacturing, construction/installation; or component failure. The cause subcode is a new item and little of the existing file data includes it, but it should substantially improve the ease of searching the file for events with particular stated causes.

The descriptive field (item 27) is essential to determining the actual cause of an event. The description includes the root cause of the problem, if known, expanded information on the personnel or components involved, and the immediate action taken and action planned to prevent recurrence. If a licensee cannot immediately determine the cause of an event, the description so states and the licensee must file an updated LER when the information becomes available. Attachments may be submitted for the physical LER file, but only 360 characters can be entered into the computer. As with the event description, more and improved information could be gained from a word search if wording were standardized.

Two new items will permit information on action taken and future action planned to prevent recurrence to be obtained more easily. Items 18 and 19 provide coded fields for this information, which must now be extracted from the cause description. The description must expand upon the information in the coded fields; the coded fields will not lessen the usefulness of the descriptive field.

Codes for the component, its supplier, and its manufacturer (items 14, 15, 16, 25, and 26), while not an essential part of the data needed for performance evaluation, make it possible to use the LERs for a far-flung statistical evaluation of components, manufacturers, and vendors.

Area of Concern: Quality of LER Data

The amount of data in the LER file for most operating power reactors is certainly sufficient for use in evaluating their performance. Other classes of licensees are substantially less well represented: as mentioned earlier, only 137 LERs are in the file for test and research reactors, fuel facilities, and materials licensees for 1976 and 1977. The quality of the data and people's perception of both the quality and the quantity deserve some comment.

Quality has two aspects: how well the data in the LER file matches the written LERs (data "goodness") and how well events are reported by the licensee. Two mechanisms are used to assure that the data are "good." First, OMIPC personnel check each licensee-coded LER form against the written description that accompanies practically all LERs (only very minor events that can be completely described in the descriptive fields need not be accompanied by a description). This check ensures that all required data are on the LER form, that there is a reasonable match between the attached description and the concise description in the LER form, and that there are no obvious errors, such as stating that the event occurred after the date of the report. The OMIPC staff generally does not question the coding of causes or the licensee responses because it lacks the technical expertise to do so. (The regional office sometimes does "change" the cause coding for its own use in focusing its inspection effort for a particular licensee; these "changes" in no way affect the data in the LER file.) This procedure is repeated as a manual "audit" after the data is keypunched but before the file is updated.

The second measure that assures "good" data is a mechanized edit check, which duplicates to some extent the check performed by OMIPC personnel and also catches keypunch errors. The LER check program has two levels. The simplest and first check is for the presence of the correct type of data: is there an entry in all required places and is it of the correct form (alpha or numeric). Next a check is made to see if the data entered are internally consistent (if item A is present, then item B must be present).* Only then is the file actually updated to include the new entry.

The second aspect of quality involves how the licensees report events, both in accuracy and quantity. NRC personnel feel that licensee reporting of events is not "uniform." One feeling is that some personnel errors are reported as component failures, because component failure "looks better" - is somehow more acceptable from the point of view of competency - than personnel error. We believe that repeated or similar events reasonably related in time may indicate either the failure of personnel to follow the established procedures, the absence of those procedures, or that plant management's QA program permitted the installation of inadequate components in the first place. The FPM model's stress on the <u>content</u> and <u>common origin</u> of events eliminates the problem of reporting personnel and management error as component failure.

<u>Area of Concern: Differing Technical Specifications</u>

Some NRC personnel also feel that certain licensees report more events than do others because their technical specifications are more numerous or more stringent. This quantitative difference is sometimes cited as a reason for discounting the information present in the LERs. Technical specifications do differ from one licensee to the next, and by type and age of plant. In general, failure to either follow procedures or to establish proper procedures as required by the technical specifications will result in their violation. But since we analyze the content of LERs, rather than counting them, this issue pales. First, <u>violations</u> of the technical specifications and license conditions are to be reported rather than compliance with them - a factor that reduces numerical difference rather than exaggerating it. Stringency and quantity of technical specifications have changed, but at

*A complete edit check includes a third level, in which the new entry is matched against the previous file entry to assure that the new entry is consistent with the other data in the file (for example, the date of the newest entry must be later than the date of the previous entry). The nature of the LER data makes this third check unnecessary.

each point in time, an applicant engaged in the NRC licensing process must be able to operate within the bounds of those specifications. And a licensee who does not report events that occur has violated the terms and conditions under which he received his license, and is highly likely to be reprimanded by NRC.

Three features of the case studies were directed toward evaluating the sensitivity of the FPM methodology to differences in licensee reporting and differences in technical specifications. First, we selected two similar facilities (Prairie Island Unit 1 and Zion Unit 1) with similar technical specifications as verified by the NRC regional management and one facility with less stringent technical specifications (Point Beach Unit 1).

Second, when we reviewed inspection reports associated with items of non-compliance identified in the 766 File, we noted the number of LERs reviewed by the inspector and whether the inspector agreed with the adequacy of the licensee's reporting of each LER. This established the quality of the reported LER data. Review of the data for the three cases studied indicated that for the "good" performers (Point Beach Unit 1 and Prairie Island Unit 1), there was nearly total agreement by the inspectors on the adequacy and completeness of LER reporting; for Zion Unit 1, a "poorer" performer, the inspectors agreed with the reporting of LERs 88 percent of the time. This information leads us to believe that the LER data is a reasonable reflection of what is actually happening in the facility for both "good" and "poor" performers.

Finally, in order to evaluate the impact of differences in technical specifications on reporting, we identified those LERs due to violations of technical specifications and calculated the proportion of these to total LERs. Table 2 presents this information for the three case studies performed thus far. We did not include LERs that report violation of environmental technical specification limits for two reasons:

Table 2

LERs Due to Violation of Technical Specifications

	Point Beach Unit 1	Prairie Island Unit 1	Zion Unit 1
Total LERs(1)	26	63	128
Total LERs due to violation of technical specifications(1)	4	7	19
Percent of LERs due to technical specification violations	15%	11%	15%

Note

(1) Not including LERs due to violation of environmental technical specifications.

- Violations of environmental technical specifications were due in part to seasonal variations in weather and to fish migrating patterns. These fact...s cannot be totally controlled by management and personnel action, short of shutting down the facility.

- Violations of environmental technical specifications generally are less related to plant operating safety than are violations of technical specifications applicable to major facility safety and balance-of-plant systems.

Table 2 shows that the percentage of LERs due to violation of technical specifications for the case studies is relatively constant for both "good" and "poor" performers and for both "stringent" technical specifications and "looser" ones.

Technical specifications represent the limiting conditions in the proper performance of existing procedures. The existence of the technical specifications may influence the character of the procedure and may even require <u>more</u> procedures. However, it appears with few exceptions that the differences in stringency of technical specifications do not provide an obstacle to meaningful comparison of the performance of licensees. In fact, our work to date suggests that these differences are far less important than how well different licensees actually implement procedures necessary to meet specification requirements. Effective implementation appears to be less influenced by technical specification stringency than by management's motivation.

<u>Area of Concern: Licensee Attitudes Toward LER Reporting</u>

A factor of which both NRC and licensees are aware is the differences in licensee attitude toward LER reporting. Conversations with licensees leave no doubt that some follow a policy of "if in doubt, file an LER," while others report only events that clearly must be reported. There appear to be three "areas" of events - clearly reportable, clearly unreportable, and a middle "grey area." It is this "grey area" that reflects attitude differences among licensees. Those with a good corporate attitude, who are cooperative toward the NRC, and who have a systematic approach to detecting and identifying reportable occurrences, probably do file more LERs. But those same conversations with licensees lead us to believe that essentially all licensees report to the "baseline" of clearly reportable events; this category of events appears sufficient to form a solid base on which one licensee can be compared with another. As seen above, inspectors agree highly with licensees' reporting of events. Further, the content of LERs in the "grey area" often shows that immediate steps are taken to correct a problem, or that a number of the events are unrelated. In short, the content of LERs can reveal good management response; numbers of LERs are not a major factor.

Effect of the Resident Inspector Program on LERs

The presence of a resident inspector in a plant may affect the "grey are" in filing LERs, by providing the plant with immediate NRC feedback on whether an event is reportable or not. This may be bad rather than good for the purpose of evaluating licensee performance, because the LERs will begin to reflect the differences in inspector interpretation of events, rather than the licensee's interpretation. A fruitful source of information on the licensee's decision-making processes may be removed. On the positive side, LER reporting may become more "uniform," but only if a high degree of uniformity in interpreting event significance exists among the resident inspectors.

3.3.2.2 Use of Licensee Event Report Data in the FPM Model

For each case study, Teknekron reviewed the NRC Licensee Event Report file (the LER file) from the perspective of the FPM model described in Section 3.2. Using the FPM model places two essential requirements on collection and analysis of LER data:

- The FPM model yields patterns of performance over time, so the temporal relationship among events is important. Therefore each LER file event was identified, reviewed, and considered in the light of previous events. Our review of each event produced a data set that contained the event cause code and event date. As explained in Section 3.2, we did not categorize events by severity, because the analysis of each event focused on the action of the licensee rather than on the potential consequences of the event.

- The FPM model explicitly defines how performance responsibility is to be assigned to Facility, Personnel, or Management. It can also relate these elements to each other through the content of the FPM "arrows." The "Proximate Cause Code" definitions used in the LER file are not clear or detailed enough to match the cause codes with the content of the FPM "arrows," but we were able to establish a parallelism between the major FPM model elements and the existing LER file "Proximate Cause Code" definitions.

These requirements, together with guidance implicit in the FPM "arrows," provided the basis for our review of the LER file for each case study. Our use of the LER data involved two processes: first, an organization and translation process to bring the LER data into the FPM data domain, and second, the analysis of that FPM data domain to reveal patterns of performance.

Creating the FPM Data Domain

As stated earlier, the relationship of events in time can provide insight into the nature and quality of licensee performance. Thus, one critical element is the date of each event, and our initial step was to review each event in chronological order.

The FPM model also allows the primary cause of a performance defect or deficiency to be assigned to one of the FPM elements. When a licensee reports an event, he assigns a "Proximate Cause Code" in accordance with NUREG-0161. To use LER data in the FPM methodology, we developed a set of event cause codes directly related to the definitions associated with the FPM model elements (management, personnel, and facility) and then identified their parallels with the Proximate Cause Codes. We have called our codes "Event Responsibility Codes" (ERCs); their definitions, together with the parallel Proximate Cause Codes, are shown in Table 3. The ERC code for each event was derived by converting the LER Proximate Cause Code on the basis of the parallelisms shown in Table 3.

Because the LERs represent real events, the recorded ERCs are linked to particular, real situations. In order to gain a comprehensive and insightful view of the licensee's response to situations and to determine patterns in this response, events must be reviewed in the light of their common point of origin. The common point of origin of events within a licensed facility is at the facility system level, and event report data are coded into the LER file by system, subsystem, and component. Our third step was therefore to organize the Teknekron Event File by system.

This rationale is at the heart of the methodology for organizing the LER file data. In summary, all events in the NRC LER file are reorganized and reclassified by:

TABLE 3

LER PROXIMATE CAUSE CODES AND TEKNEKRON EVENT RESPONSIBILITY CODES

Proximate Cause Code	Definition	Definition	Event Responsibility Code
A	Defective Procedures	All actions falling within the purview of management responsibility, excluding "hands on" operation of the facility.	M
B	Personnel Error	All actions and responsibilities accruing to those with responsibility for "hands on" operation of the facility.	P
C	Component Failure	The failure of a component or system within the facility, not caused by personnel error in the maintenance or operation of the facility.	F
D	External Cause/Other	All events which are not related to a failure of management, personnel, or the random failure of a component. These events are unimportant to the Teknekron analysis and are grouped and designated as such.	O

- System: This establishes the common point of event origin within the facility and provides a sensitive parameter for the isolation of performance patterns.

- Chronological order of event occurrence: This permits the sequence of events over time to be examined. Such examination may show specific relationships among events (causal linkages).

- Teknekron "Event Responsibility Code": This allows a deficiency in performance to be assigned to one of the FPM model elements.

One the data are in this format, they have been transferred into the FPM data domain and are in a form that allows meaningful analysis and identification of performance patterns.

Analysis of the FPM Data Domain

To use existing LER data with the FPM model, the "Proximate Cause Code" assigned by the licensee to each event is subjected to a two-step transformation:

1) "Straight-across" conversion into an ERC, using Table 3 as previously discussed, and

2) In some cases, changing the initially assigned ERC (for example, from F to P), if the events are found to be causally linked after analyzing their relationship within the facility system.

We stated earlier that events were analyzed by system and in chronological order of occurrence. To identify event relationships, we compared each event in a system with previous events in that system, searching for these cues:

- the similarity of involved components
- the similarity of and relationship to subsystems, and
- the similarity of human response and involvement

If any of these cues was common to two events in a system, these events were considered to be causally linked. When we identified a second event as being causally linked to a prior event we always changed the Event Responsibility Code (ERC) for the subsequent event. In general, we changed the code of the second event in any causally linked group of events to ERC-M (management responsibility), on the basis that the repetition of an initially random event was due to a failure by management to identify and rectify the fundamental event cause and apply generically the lessons and information learned from the event.

We feel that use of the second event to establish the onset of causal linkage is justified, because it provides:

- convervatism--it is the earliest possible point for establishing systematic management deficiencies, and
- maximum sensitivity to detect the character of "good" and "poor" licensee performance--since an abrupt end to a series of causally linked events establishes positive licensee management performance.

It is possible that a licensee may react to a first event by recognizing that a design change or technical specification modification is required and by taking appropriate action. Under these conditions it would be inappropriate to assign further events to ERC-M. In performing the three case studies (keeping in mind they were mature plants), we found that events for which either a design change or technical specification change was required to prevent recurrence were quite rare. We also found that when a design change was required, the licensee noted this information in the event report; the event report describing the need for the design change usually either marked the end or was the close to the end of a causally linked group of events.

Time is a dependent variable in our analysis, since the licensee's deficient performance determines the frequency of occurrence of the causally linked events, as well as the number of causally linked event groups that exist in any time period.

After analyzing the data for a particular system, using the cues of similar components, similar or related subsystems, and similar human involvement to search for linked events, those data may yield a pattern significantly different from the pattern formed when they were translated "straight across" into the FPM data domain. These differences are evident as:

- A marked increase of events coded ERC-M, which were initially classified as ERC-F.
- A marked increase of events coded ERC-M, which were initially classified as ERC-P.
- The identification of causally linked groups of events within systems in which codes ERC-F were changed to ERC-M.

The patterns that emerge from the analysis of the data permit inferential judgments of licensee performance. Conversely, the absence of these patterns is also an indicator of performance. The fact that the patterns of performance are manifested on a system basis is due to the structure of the analytical technique; these patterns should not be presumed to be absent from other areas of facility operation or licensee performance, and may also hold across systems, as well as within them.

Changing Codes and Identifying Causally Linked Groups of Events: Examples

To demonstrate analysis of the FPM data domain and how we used the previously mentioned cues to find causally linked events, we have provided excerpts from the case studies in Appendix A. The first set of causally linked events occurred in the "Containment Isolation System" of Zion Unit 1. When reviewing this excerpt, note the following:

1) The similarity of involved components--solenoid valves
2) The similarity of and relationship to subsystems--the failure of each valve is linked to the instrument air supply

3) The similarity of human response and involvement--the licensee identified the first event as being due to a valve stuck open by "crud and rust." The second and subsequent events were due to "impurities in the instrument air system" and "varnish buildup." One year after the first event--and six events later--the licensee stated that new equipment was being installed; however, it is not clear what the new equipment was, since there were two subsequent events.

The date of the causally linked events, together with the cause assigned by the licensee and Teknekron's ERC Code, are:

<u>Date (Licensee Code/ERC)</u>

4-07-76(F)
8-11-76(F/M) - 2 events
9-30-76(F/M)
1-23-77(F/M)
4-25-77(F/M)
7-23-77(F/M) - 2 events

The licensee identified the cause of the 4-07-76 event as a valve (inlet unloader valve) stuck open by "crud and rust." The valve was located in the system that provides compressed air to pressurize penetrations. On 8-11-76 two events occurred in which two identical components (solenoid valves) failed. For one event, the licensee stated the cause as "...probably due to impurities in the instrument air system." The other event, involving an identical component, was listed as due to "varnish buildup." On 9-30-76, an identical event (solenoid valve failure) occurred with the same stated cause as the 8-11-76 event ("varnish buildup"). The 1-23-77 event (solenoid valve failure) identified the same component failure as the 8-11-76 event; the stated cause was impurities in the instrument air supply. The 4-25-77 event was identical to the 1-23-77 event in all respects, but the licensee stated that new equipment was being installed. On 7-23-77 two separate events occurred, each identical to the previous 4-25-77 event. In this case, the licensee stated that monthly tests would be performed and the air line blown clean.

All but the first event was upgraded from ERC-F to ERC-M, because the analysis of the first event (crud and rust causing the valve to stick) did not indicate that the licensee sought the broader implications of the specific event (possible contamination of the instrument air system) or considered generic remedies (cleaning impurities out of the instrument air system). These actions are the responsibility of management, as defined in the ERCs. Management's failure to thoroughly analyze the information gained in the first event and to conclude that an inspection should be performed to determine

1) if the cause of the crud and rust on a valve in the closed system was due to instrument air system impurities, and
2) the potential impact and implications of this event on other components in contact with the instrument air supply

probably contributed to the occurrence of subsequent events in the system, or at the very least did nothing to prevent them.

The preceeding example and discussion illustrate the use of cues in making code changes as well as establishing causal linkage among events. They also provide a first hand view of "poor" performance.

A second example will further illustrate code changes. The following set of events occurred in the "Reactor Containment System" of Prairie Island Unit 1 during 1976 and 1977. While several of the events are causally linked, the type of code changes are distinctly different from the previous example.

Date (Licensee Code/ERC)

5-04-76(P)
8-25-76(P/M)
10-23-76(P)
3-16-77(F)
9-29-77(P/M)
12-09-77(F)

On the basis of our review, events on 5-04-76 and 10-23-76 clearly are the result of isolated personnel error. But the events of 8-25-76(P/M) and 9-29-77(P/M) appear to be causally linked through apparent management failure to develop and implement administrative controls for the auxiliary building special ventilation zone. In the report of the 8-25-76 event, the licensee identified lack of administrative control as being partly responsible for the event. The 9-29-77 event seems to have resulted from a less-than-complete implementation of the administrative controls.

The change of cause code from P to M in the case of the 8-25-76 event was made because the doors should have been under administrative control. The event reflected a defect in existing procedures for which management and not personnel are responsible, according to the ERC definitions. The 9-29-77 event was similar to the 8-25-76 event in that both involved a breach of ventilation zone integrity by personnel; however, the 9-29-77 event resulted from incomplete administrative control as stated by the licensee. This event group demonstrated that the facility management was aware of the need for generic event cause identification and remedy application. It is also a demonstration of how the facility management performs its role in responding to events.

Our last example demonstrates a case of licensee management response to an event in which the potential cues for causal linkage to subsequent events are nonexistent. The event occurred in the "Hangers, Supports, Shock

Suppressors" system* for Point Beach Unit 1. During refueling outage surveillance testing of safety-related shock suppressors (snubbers), a snubber failed to lock up when the specified load rate was applied. The licensee found that the control valve on the snubber was improperly set, and attributed the event to personnel (ERC-P). The licensee stated in the event report that similar snubber control valves were checked. There were not other events in this system category during the study period. This event demonstrates:

1) the licensee's awareness of the similarity of the components involved in this event to others in the facility
2) the licensee's determination to identify the generic event cause--in this case a highly specific personnel error
3) the licensee's response to the generic event cause, concern for the potential impact of the generic event cause on other plant systems, and willingness to apply a generic remedy to a potential cause of additional events.

Performance Profiles

The patterns of a licensee's performance can be graphically presented as profiles either showing events in a single system or all events attributable to human causes or to component failure. A profile of all events for the Containment Isolation System at Zion Unit 1 is shown in Figure 3 and a profile for the Reactor Containment System at Prairie Island Unit 1 in Figure 4. Time forms the x-axis; the Event Responsibility Codes are arranged on the y-axis so that ERC-M has the greatest ordinal value and ERC-O the least. Each event is recorded as a bar located on the x-axis at the time it occurred; the height of the bar corresponds to its final

*This is not a system code in the LER file, but the component subcode makes these events readily identifiable.

PROFILE OF TOTAL REPORTED EVENTS IN

CONTAINMENT ISOLATION SYSTEM

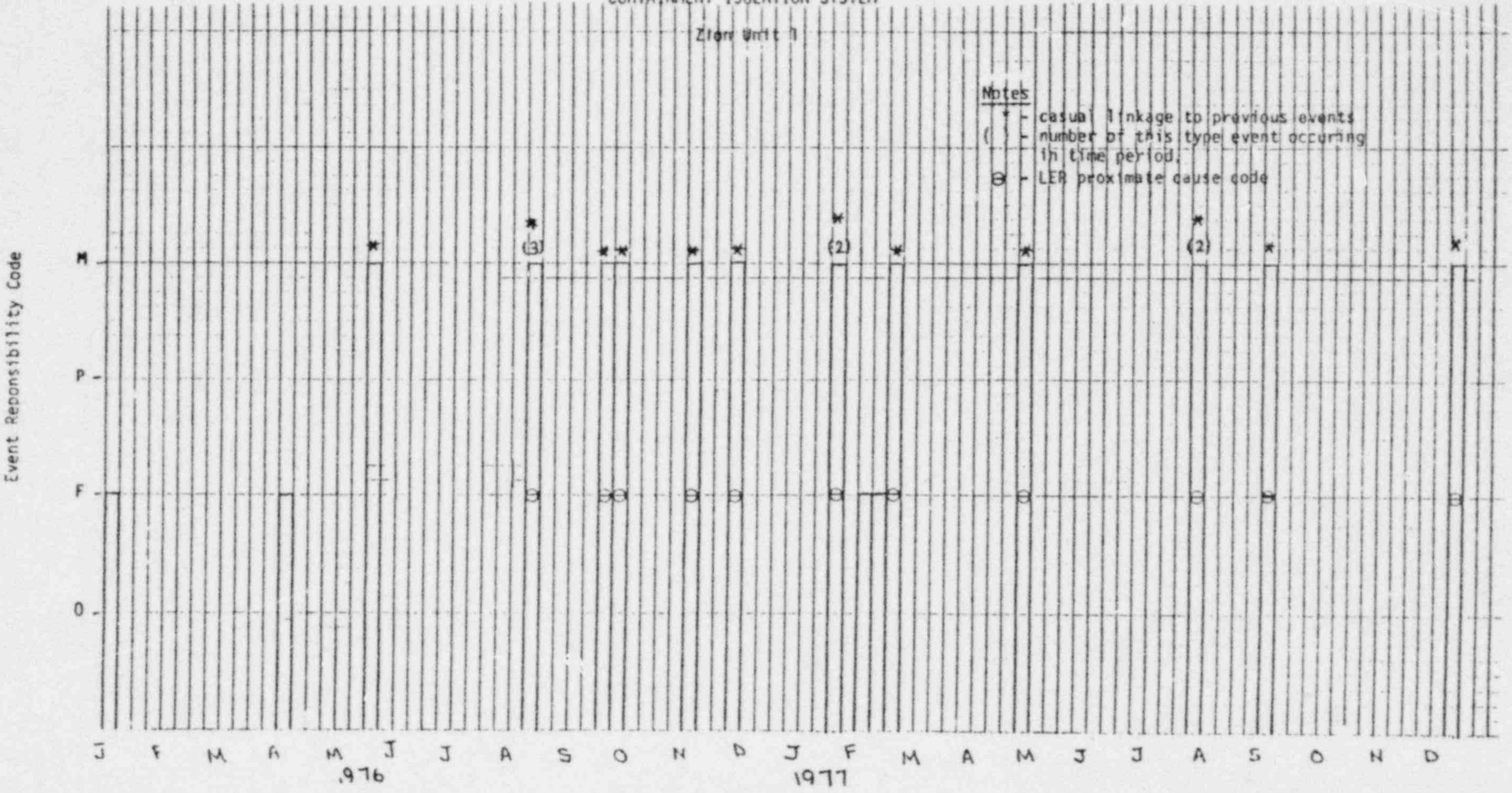


FIGURE 3

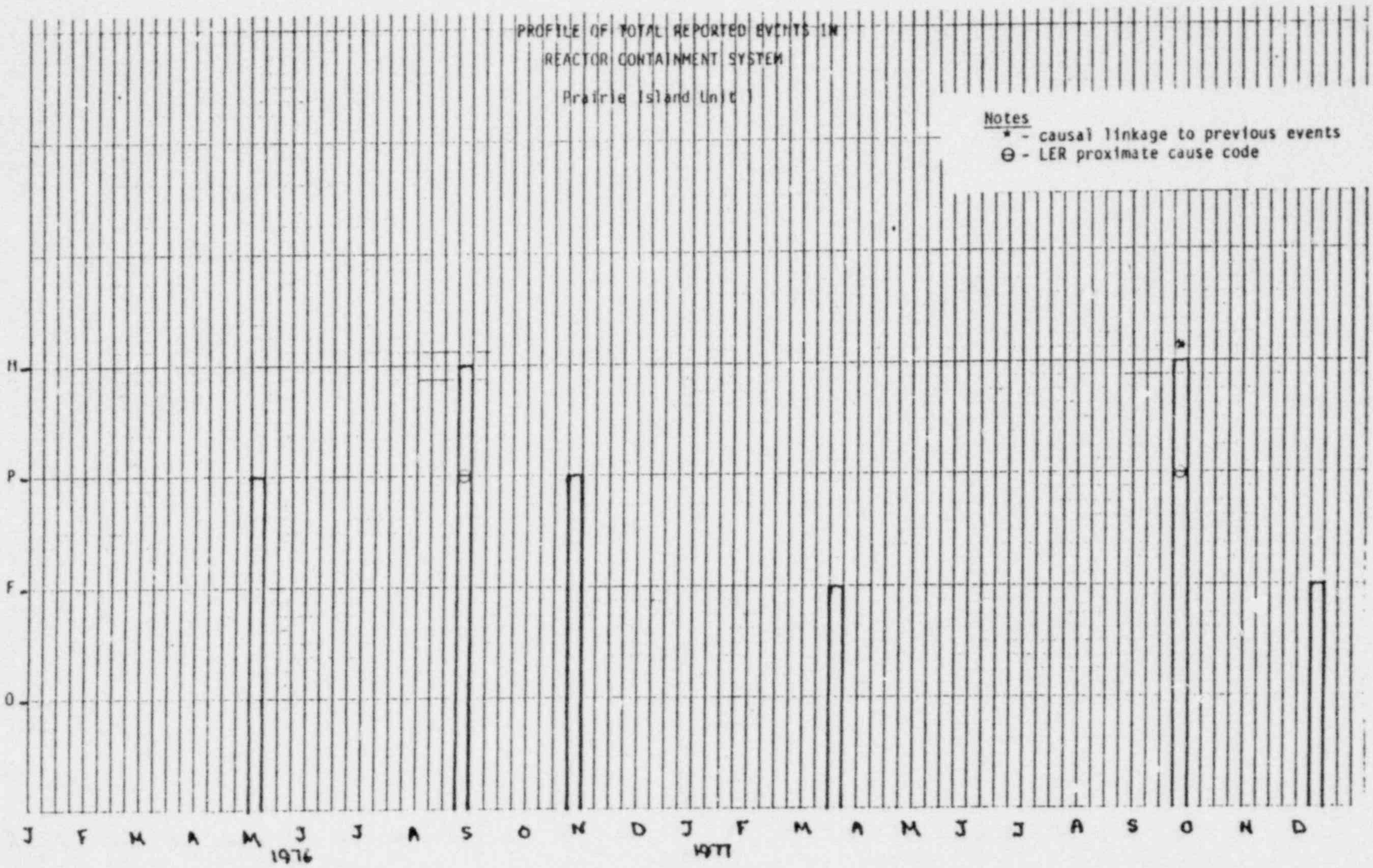


FIGURE 4

Teknekron, Inc.

ERC. If our analysis required a change in ERC, the licensee's reported ERC for the event is noted by a "⊖" on the bar. The asterisk notation "*" above a bar indicates that the event is causally linked to a previous event. Note that any system profile may include several groups of causally linked events. We have not identified different groups of causally linked events on the graphic profiles, though this could be done; the total number of causally linked events occurring within a specific time period is a sensitive indicator of licensee performance because it appears to indicate a systematic breakdown in management control.

System profiles (except those that involve environmental technical specifications such as the Circulating Water System and in some cases the Ultimate Heat Sink System) can be combined to produce a profile of all the reported events that were attributable to human causes. Profiles of this type are shown in Figure 5 for Prairie Island Unit 1, Zion Unit 1 and Point Beach Unit 1. Time agin forms the x-axis, but in this case the y-axis represents numbers of events. For each point on the x-axis, the events in all systems with codes ERC-P and ERC-M are added; the total number determines the height of the bar. The ERC for each event in this aggregate presentation is the final or "upgraded" categorization of the event, not necessarily that reported by the licensee.

An aggregate profile of events attributable to component failure can be produced by summing all events ultimately classified as ERC-F for all facility systems. These profiles are shown in Figure 6 for Prairie Island Unit 1, Zion Unit 1, and Point Beach Unit 1. The information contained in these component future profiles appears to provide a less direct indication of licensee performance than profiles of events attributable to human causes, since the three profiles in Figure 6 bear far more similarity to each other than those in Figure 5, the "Profiles of Total Reported Events Attributable to Human Causes." We believe this indicates that genuine component failures are in large part random, since the major portion of those

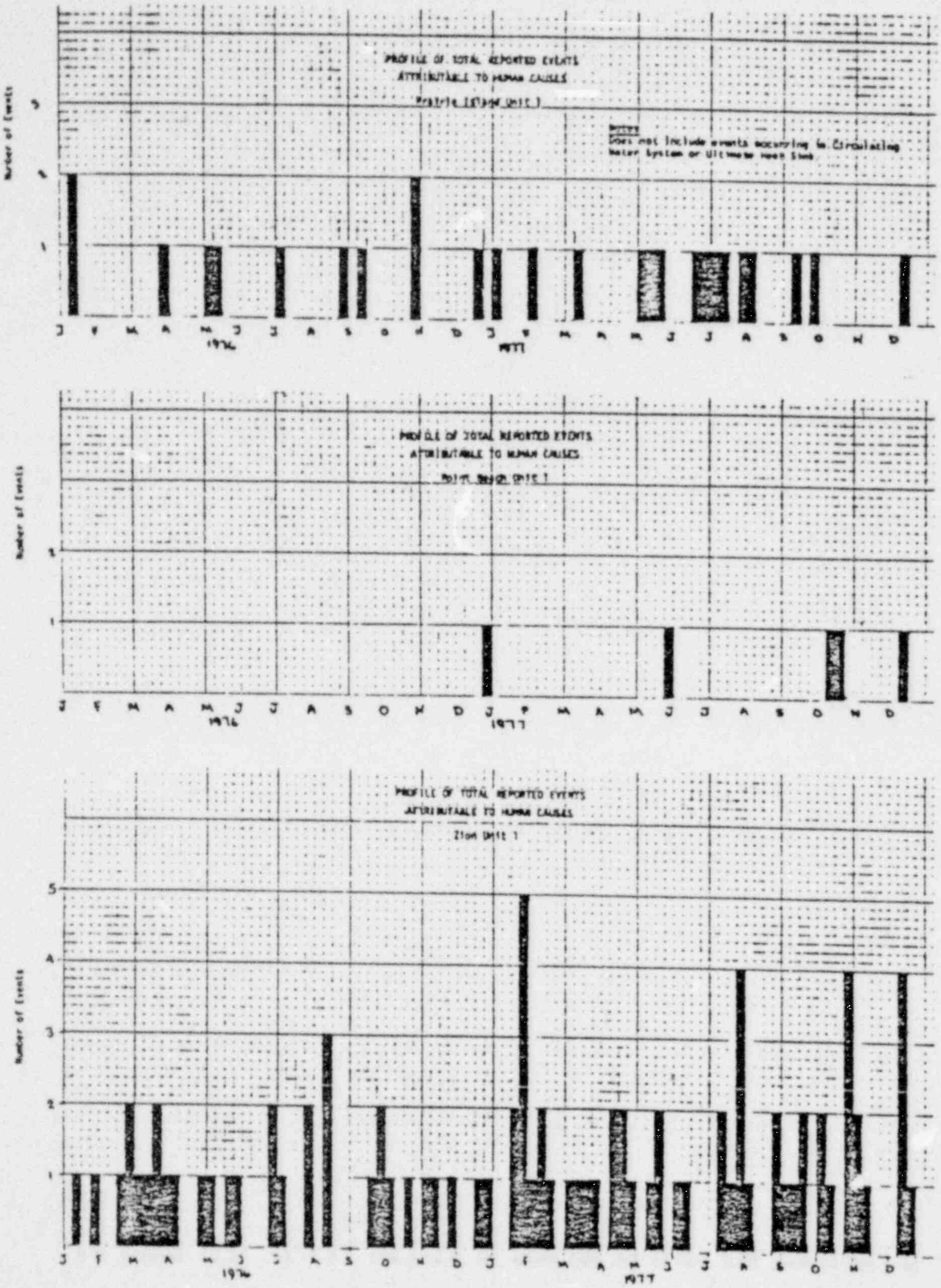


FIGURE 5

Comparison of LER Profiles

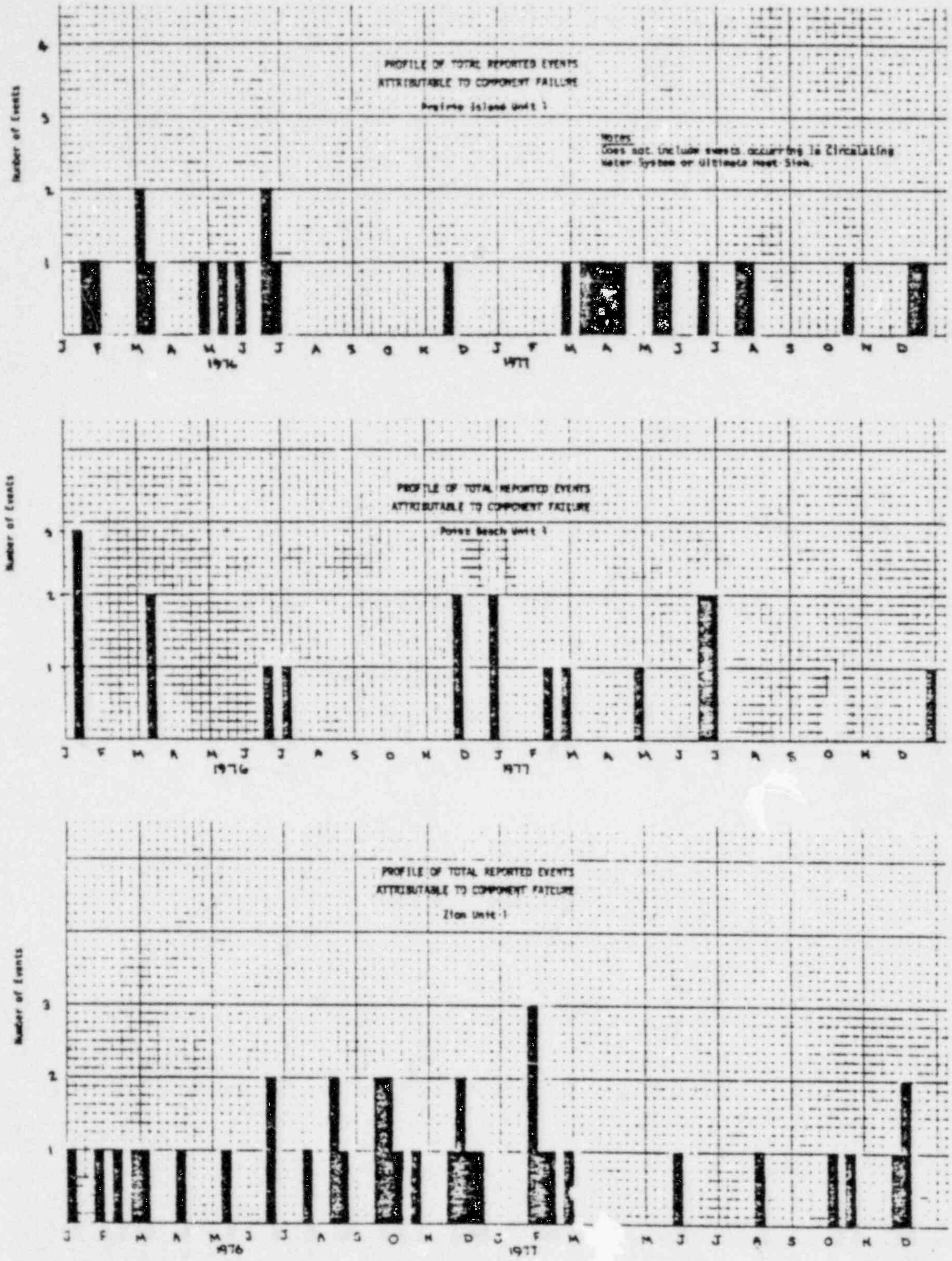


FIGURE 6

Comparison of Component Failure Profiles component failures that on analysis are due to systematic failures in human performance have been reclassified as ERC-P or ERC-M through identification of causal linkages.

We recommend that the reader review the detailed case studies presented in Appendix A for a fuller appreciation of differences in licensee performance as revealed in the LER data.

Available Licensee Performance Indicators

The previous discussion described how we analyzed the LER data, constructed profiles of events by system and, for total reported events attributable to human causes, identified patterns of deficient and adequate performance.

In one of the case studies, events occurred due to human failure that were serious from the regulatory point of view. This licensee also exhibited substantial numbers of causally linked events in several systems. It may be possible, after further case studies are complete, to say that patterns of poor performance precede the occurrence of events that NRC determines are serious enough to warrant citation.

Because these profiles are based on licensee response to actual events, we believe that these profiles are insightful and sensitive indicators of licensee performance. The performance evaluation for each licensee should include at least:

- A profile of total reported events attributable to human causes
- Profiles of those systems in which causally linked events are identified. For some licensees, a substantial number of systems may contain causally linked events, and it may be possible to construct profiles for only those systems NRC feels are most relevant to safety or that have substantial numbers of recent events.

3.3.3 Noncompliance Data

3.3.3.1 Type and Extent of Noncompliance Data

The NRC's modularized inspection program produces vast quantities of information. The 766 system is a computerized data file used to capture, maintain, and report statistical and planning data on inspection, investigation, inquiry and enforcement actions conducted by I&E.* The system provides input to the Rainbow Books, which summarize the operation of licensees and the actions taken by I&E. The 766 file accommodates inspection data on all classes of licensees, but as with the LER file, most data exist on the operating power reactors. For the calendar years 1976 and 1977, the file contains data from 1,997 inspection reports for the roughly 90 reactors under construction; 247 reports are included for 93 test and research reactors. In the same period, there are 995 inspection reports covering 38 fuel facilities, and 4,737 reports are shown for the roughly 9,600 materials licensees.

The 766 system is really a dual system. The 766 form, both sides of which are shown in Figure 7, records the management information needed to track the status of the inspection and enforcement program as applied to a particular licensee. The information contained on form 766-S, shown in Figure 8, is more valuable for licensee performance analysis. The information on the 766-S form is entered into a part of the system known as the "enforcement text file," a title that accurately reflects the major data field on the form. The computerized 766 file has existed in its present form since July of 1975. Instructions for completing the forms from which data are entered into the computer were revised in February of 1978, to account for the fees that are now being charged by NRC for routine inspections.

---

*As of October, 1977, licensee-identified noncompliances are no longer entered in the 766 file. Such noncompliances have been included in the case studies because the study period included 1976 and 1977. But note that these self-reported noncompliances were largely treated as deviations, and seldom were assigned cause codes.

MC 0535

FORM NRC 766
OCTOBER 1977
(MC 0535)

UNITED STATES NUCLEAR REGULATORY COMMISSION

INSPECTION & ENFORCEMENT – STATISTICAL DATA

FACILITY NAME ______ INSPECTOR(S) ______ PRINCIPAL INSPECTOR ______

LICENSEE/VENDOR ______ REVIEWER ______

AE
- TRANSACTION TYPE 1 (CHECK ONE) ☐ Delete ☐ Insert ☐ Modify
- DOCKET NUMBER (A) 2–9; OR LICENSE NO. (BY PRODUCT) 2–14
- REPORT NO (B) 15–18
- DATES INQ/INVEST/INSP FROM (C) 19–24 M M D D Y Y; TO (D) 25–30 M M D D Y Y
- REGION CONDUCTING ACTIVITY (E) 31

F INSPECTION PERFORMED BY: 32 — 1 ☐ REGIONAL OFFICE STAFF 2 ☐ RESIDENT INSPECTOR 3 ☐ PERFORMANCE APPRAISAL TEAM

G TYPE OF ACTIVITY CONDUCTED (CHECK ONE BOX ONLY) 33-34

INSPECTION: 01 ☐ ROUTINE (FEE) 02 ☐ ROUTINE (NO FEE) 03 ☐ INCIDENT 04 ☐ ENFORCEMENT 05 ☐ MANAGEMENT AUDIT 06 ☐ MANAGEMENT VISIT 07 ☐ SPECIAL 08 ☐ VENDOR 09 ☐ MATL ACCT. 10 ☐ PLANT SEC. 11 ☐ INVENT. VERIF. 12 ☐ SHIPMENT/EXPORT 13 ☐ IMPORT

OTHER: 14 ☐ INQUIRY 15 ☐ INVESTIGATION (IF INVEST. ALSO CHECK BLOCK R)

H INSPECTION OR INVESTIGATION WARNING: 35 — 1 ☐ ANNOUNCED 2 ☐ UNANNOUNCED

I PRIMARY INSPECTION SHIFT 36 — 1 ☐ DAY 8AM-4PM 2 ☐ EVENING 4PM-12 MIDNIGHT 3 ☐ NIGHT 12 MID-8 AM

J INSPECTION/INVESTIGATION NOTIFICATION (CHECK ONE BOX ONLY) 37 — 1 ☐ 591 2 ☐ REGIONAL OFFICE LETTER 3 ☐ REFERRED TO HQS FOR ACTION 4 ☐ REGION LETTER & HQS FOR ACTION

K INSPECTION/INVESTIGATION FINDINGS (CHECK ONE BOX ONLY) 38 — 1 ☐ CLEAR 2 ☐ NONCOMPLIANCE 3 ☐ DEVIATION 4 ☐ NONCOMPLIANCE & DEVIATION

L ENFORCEMENT CONFERENCE HELD: 1 ☐ 39

M NUMBER OF NONCOMPLIANCE ITEMS IN LETTER TO LICENSEE: 40 41

N NUMBER OF DEVIATION ITEMS IN LETTER TO LICENSEE: 42 43

NOTE: CHANGE MUST BE SUBMITTED ON 766 WHENEVER PREVIOUSLY CITED ITEM OF NONCOMPLIANCE IS OFFICIALLY DELETED FROM THE RECORD

O NUMBER OF LICENSEE IDENTIFIED ITEMS 44 45

P NUMBER OF LICENSEE EVENTS 46 47

Q REGIONAL OFFICE LETTER OR REPORT TRANSMITTAL DATE FOR INSPECTION OR INVESTIGATION
- 591 OR LETTER ISSUED TO LICENSEE 48–53 M M D D Y Y
- REPORT SENT TO HQS FOR ACTION 54–59 M M D D Y Y
- IMMEDIATE ACTION LETTER 60 DATE 65 M M D D Y Y

R SUBJECT OF INVESTIGATION (CHECK ONE BOX ONLY) 66 67

TYPE A 10 CFR 20.403: 01 ☐ INTERNAL OVEREXPOSURE 02 ☐ EXTERNAL OVEREXPOSURE 03 ☐ RELEASE TO UNREST. AREA 04 ☐ LOSS OF FACILITY 05 ☐ PROPERTY DAMAGE

TYPE B: 06 ☐ 07 ☐ 08 ☐ 09 ☐ 10 ☐

10 CFR 20.405: 11 ☐ INT. OVEREXPOSURE 12 ☐ EXT. OVEREXPOSURE 13 ☐ EXCESS RAD. LEVELS 14 ☐ EXCESS CONC. LEVELS

MISC.: 15 ☐ CRITICALITY 16 ☐ LOSS/THEFT 17 ☐ MUF 18 ☐ TRANSPORTATION 19 ☐ CONTAM/LEAKING SOURCE 20 ☐ ENVIRONMENTAL EVENT 21 ☐ EQUIP. FAILURE 22 ☐ ALLEGATION/COMPLAINT 23 ☐ PUBLIC INTEREST 24 ☐ SABOTAGE 25 ☐ ABNORMAL OCCUR 26 ☐ OTHER

HEADQUARTERS ENTRIES

S HQS ACTION ON INSP/INVEST REFERRED BY REGION: (See Reference List for Code) 68 69

T DATE HQS ENFORCEMENT LETTER, NOTICE, ORDER ISSUED: 70–75 M M D D Y Y

U CIVIL PENALTY ISSUED: 1 ☐ 76

V DATE 766 ENTERED INTO COMPUTER FILE (MO/YR): 77–80 M M Y Y

NOTE: BLOCKS K TO O MUST BE VERIFIED BY IE:HQS WHENEVER ENTRIES ARE MADE IN BLOCKS S, T, AND U

AITS REFERENCE:

FIGURE 7

766 Form - Front Side

FORM NRC 766 — SIDE 2

USNRC — INSPECTION & ENFORCEMENT STATISTICAL DATA

(NOTE: % COMPLETE AND STATUS; LEAVE BLANK FOR MC 92, 93 & 94 PROCEDURES AND 30-702, 30-703 & 30-800)

(NOTE: STATUS CODING:
BLANK - TO REMAIN OPEN
C- CLOSED
L- REOPEN & LEAVE OPEN
P- REOPEN THIS TRANSACTION ONLY)

(NOTE: MODULE REQUIRING FOLLOWUP:
USE ONLY WHEN MODULE INSPECTED IS 92-701B.)

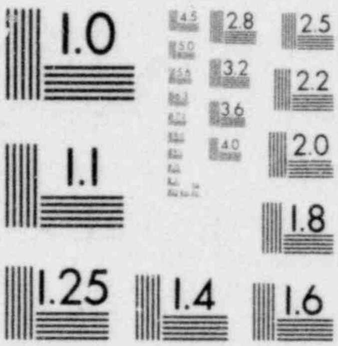
			MODULE TRACKING INFORMATION															NONCOMPLIANCE										
			MODULE NO. INSP.						DIRECT INSP EFFORT MAN HRS EXPENDED THIS INSP.	% COMPLETE TO DATE	STATUS	MODULE REQ. FOLLOWUP						N/C		N/C		N/C		N/C		N/C		
			PHASE	MANUAL	CHAP.	PROC.	NO.	LEVEL				PHASE	MANUAL	CHAP.	PROC.	NO.	LEVEL	CODE	SEV	CODE	SEV	CODE	SEV	CODE	SEV	CODE	SEV	
B		0																										
B		1																										
B		2																										
B		3																										
B		4																										
B		5																										
B		6																										
B		7																										
B		8																										
B		9																										
1	2	3	4					10	11 13	14 16	17	18					24	25	28	29	32	33	36	37	40	41	44	45

LINE COLUMN NUMBER SHOWN IN BOTTOM LINE

☐ CONTINUES ON SUBSEQUENT SHEET

FIGURE 7 (continued)
766 Form - Back Side

IMAGE EVALUATION
TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART

FORM NRC-766-S
OCTOBER 1977
(MC 0535)

UNITED STATES NUCLEAR REGULATORY COMMISSION
INSPECTION & ENFORCEMENT–STATISTICAL DATA SUPPLEMENT

GENERAL INSTRUCTIONS: BLOCK A MUST BE CHECKED TO SHOW TYPE OF INFORMATION CONTAINED ON THE FORM. ONE FORM IS TO BE COMPLETED FOR EACH:
(A) ITEM OF NONCOMPLIANCE CITED.
(B) ITEM OF NONCOMPLIANCE NOT CITED, AND,
(C) DEVIATION CITED IN ENFORCEMENT CORRESPONDENCE

DOCKET NUMBER ______ OR LICENSE NUMBER ______ REPORT NUMBER ______

S
1

A–TYPE OF FINDINGS
(CHECK ONE BOX ONLY)

2
A ☐ NONCOMPLIANCE
B ☐ LICENSEE IDENTIFIED ITEMS
C ☐ DEVIATION

B–SPECIFIC N/C CODE OR DEVIATION (3–7) | CAUSE (8) | PROCEDURE (9) | SEVERITY (10) | FUNCTIONAL AREA (11 12)

C–MODULE NUMBER (13–19)

D–HOW ITEM IDENTIFIED
(CHECK ONE BOX ONLY)

20
L ☐ LICENSEE
I ☐ INSPECTOR
O ☐ OTHER

E–CONSEQUENCES
(CHECK ONE BOX ONLY)

21
A ☐ CAUSE OR CONSTITUTED ACTUAL OCCURRENCE
P ☐ HAD POTENTIAL TO RESULT IN ACTUAL OCCURRENCE
N ☐ DID NOT HAVE POTENTIAL TO RESULT IN ACTUAL OCCURRENCE

F–EXEMPT INFORMATION

22 S ☐ CHECK BOX IF EXEMPT INFORMATION IS INCLUDED IN TEXT BELOW, PER 10 CFR 2.790

G–ADDITIONAL UNITS (ENTER "M" IN FIRST BOX FOLLOWED BY OTHER UNIT NUMBERS) (23–26)

H–NONCOMPLIANCE REPETITIVE OCCURRENCE (IF FIRST OCCURRENCE ENTER "1") (27)

I–TEXT (ENTER UP TO 2400 CHARACTERS FOR EACH ITEM. IF THE TEXT OF THE NONCOMPLIANCE OR DEVIATION EXCEEDS THIS NUMBER, IT WILL BE NECESSARY TO PARAPHRASE)

1 ... 50
(1)
(2)
(3)
(4)
(5)
(6)
(7)
(8)
(9)
(10)
(11)
(12)
(13)
(14)
(↓)
(48)

&	E	N	D	&

1 ... 5

NOTE: DATA ENTRY CLERK–THE LAST LINE ENTERED MUST CONTAIN THIS INFORMATION.

FIGURE 8

766S Form

Since this change does not affect the information now in the file, the description here is based on the forms and instructions effective November 1977.

The November 1977 revision made four changes in the 766 form. It identified which shift conducted the inspection (block I), whether an enforcement conference was held (block L), the date an immediate action letter was sent, if any (block Q), and who performed the inspection (the resident inspector, performance appraisal team, or regional office inspectors (block F). Changes in the 766-S form added the module number associated with the noncompliance or deviation (block C) and a block to record whether the noncompliance had occurred before (block H). This item is potentially very useful in analyzing performance since repetitions, particularly if closely linked in time or type, may have a common cause that the licensee has not adequately addressed. But its current usefulness is hampered by the lack of definition of "repeated noncompliance" and by the differences in individual inspector's knowledge of the compliance history of a particular plant.

766 Data Elements Used in Licensee Performance Analysis

We used three main types of 766 file data in licensee performance analysis. First, information that identifies a licensee is essential, and this role is played by the docket number that appears in block A on form 766. The license number that also appears in block A is potentially useful in extracting data on a group of licensees for aggregate rather than individual evaluation. Second, the date the inspection concluded (block D) places any noncompliance items in time. Last we used the Primary Cause of Violation code (block B on form 766-S). There are 18 noncompliance cause codes, covering various types of management, personnel, and equipment failure, and a few categories that cover situations that the licensee cannot control. On first reading, the codes seem fairly specific in attributing cause to certain types of breakdown in behavior--"inadequate plans or procedures" (code G); "s.fety devices not maintained" (code M)--but discussions with NRC personnel and staff revealed that most inspectors have "favorite" codes, and that each inspector does not use the codes in the same way. To make the current detail in the codes more useful, we need to know how inspectors actually use the codes and whether there are definable regional differences in their interpretation. This information can be obtained through a survey of inspectors, using sample "noncompliances" designed to test their responses.

The text in the 766-S file, while it provides a brief description of the noncompliances, often is so brief that it reveals little about the circumstances surrounding the noncompliance, making it difficult to analyze the licensee's behavior.

Area of Concern: Quality of 766 Data

Data quality has two parts, and the 766 system does well on one of those. The first part of data quality is "mechanical"--how accurately the data is entered into the computer and how well the file data matches the written inspection reports. Accurate entry is ensured by mechanized "edit checks" that have three parts. The simplest and first check is for the simple presence of the correct type of data: is there an entry in all required places and is it of the correct form (alpha or numeric). Next a check is made to see if the data entered are internally consistent (if item A is present, then item B must be present). The third check matches the new entry against the previous file entry to assure that the new entry is consistent with the other data in the file (for example, the date of the newest entry must be later than the date of the previous entry). After all three checks are complete, the file is actually updated to include the new entry.

An audit determines how well the data in the file matches the written inspection reports. Table 10 in the Quarterly Report for the quarter ending September 1977 displays the results of an audit conducted in that year. Our statistical consultant has reviewed the methodology on which we were told the table is based, and he states that the methodology is statistically correct. But the number of reports selected for sampling was apparently not determined by the methodology, though the chosen reports were selected randomly. Because the methodology was not completely followed, it is not clear that Table 10 reflects the quality of the data in the 766 file.

In the course of reviewing two calendar years of 766 file data for our three case studies, we performed our own "mini-audit" by reading every inspection report that was associated with a noncompliance and some reports that document inspector followup of LERs. Our limited check left us with two impressions. First, we feel confident that the superficial data match between the 766 file and the inspection reports is quite good, but the apparent match between elements of the 766 data themselves is much poorer.* The root of this problem appears to lie in the noncompliance cause codes or their use by the inspectors, and the fact that the text is often too brief. If the 766 file cause code parameters are unable to fully describe the situation in any case, then the potential usefulness of the file data is diminished. To be assured that the data on file accurately reflect the inspection reports, we feel a new audit is necessary, based on accepted and sound statistical methods. Coupled with the survey of inspectors mentioned earlier, this would provide a more solid basis for use of the 766 file data.

Our second impression is that the 766 file, especially the 766-S text, is often a pale reflection of the information in the inspection reports. The use of the cause codes sometimes depends on their interpretation by individual inspectors; the 766-S text is often too brief to provide an adequate

*Our "mini-audits" are recorded in the matrices included in the case studies for each licensee. The results of analyzing the matrices are discussed in Section 3.3.3.2.

representation of the circumstances surrounding a noncompliance. These factors make it difficult to analyze the content of the 766 file data to the same degree possible with the LER file. To be sure, the inspection reports themselves could be used in analyzing performance, but to read every inspection report for at least two years for every licensee is a formidable task. Computerized data must be used whenever possible, and the usefulness of the 766 file for licensee performance analysis could be considerably enhanced by expansion of the text and better definition and use of the cause codes.

The second aspect of data quality concerns timing, and this difficulty cannot be alleviated by improving the data quality of the 766 file. While inspection reports were generally filed within a month of the inspection, the noncompliances cited in those reports often were related to events that occurred some time past. For example, assume that a new calibration procedure was issued several months ago. The licensee calibrated his instrumentation using the new procedure, except in one area. Thus, his failure may have occurred much earlier than its detection by the inspection program. This point is discussed more fully in the next section, but in general, we feel that the usefulness of inspector-generated data is limited by the lack of a close time relation between a real action and its report through the inspection process.

3.3.3.2 Use of the NRC 766 System Data and Related Inspection Reports in the FPM Model

For each case study, we reviewed the NRC 766 system data and related inspection reports from the perspective of the FPM model. The FPM model places two essential requirements on the analysis of the 766 system data:

- The FPM model yields patterns of performance over time, so the temporal relationship among events is important. We therefore considered the factors in the inspection program that control the pattern of the noncompliance items identified by an inspector as a function of time. We also considered the temporal relationship of the performance of citable actions by the licensee to the "real time" of their detection. As explained in Section 3.2, we did not categorize noncompliance items by severity.

- The FPM model explicitly defines how performance responsibility is to be assigned to Facility, Personnel, or Management. It can also relate these elements to each other through the content of the FPM "arrows." While the "Primary Cause of Violation" codes (noncompliance cause codes) are reasonably detailed, they are not defined or used precisely enough to match the cause codes with the content of the FPM arrows. But, we were able to establish a parallelism between the major FPM model elements and the noncompliance cause code definitions.

These requirements, together with guidance implicit in the FPM "arrows," provided the basis for our review of the 766 file for each case study. Our approach is initially parallel to our review of the LER data, but it ultimately diverges. The reason for that divergence lies in the structure of the inspection program.

Specific Considerations in the Development of the 766 File Review Methodology

The relationship of events in time can provide insight into the nature and quality of licensee performance. Thus, one critical element is the date of each noncompliance or citable occurrence, and our initial step was to review each noncompliance in chronological order.

Noncompliances are either random lapses from good performance (random human error) or systematic lapses due to a performance defect or deficiency assignable to one of the FPM circles or arrows. When an inspector reports a noncompliance, he assigns a "Primary Cause of Violation Code." To use noncompliance data in the FPM methodology, we identified the parallels between the Event Responsibility Codes (ERCs) we developed for use with the LERs and the Primary Cause of Violation Codes. The relationships between ERCs and noncompliance cause codes are shown in Table 4.

TABLE 4

766 FILE CAUSE CODES AND EQUIVALENT TEKNEKRON EVENT RESPONSIBILITY CODES

NRC 766 FILE		TEKNEKRON EVENT FILE	
Primary Cause of Violation Code	Definition	Definition	Event Responsibility Code
C	Improper or Inadequate Design	All actions falling within the purview of management responsibility, exluding "hands on" operation of the facility.	M
D	Improper or Inadequate Construction		
E	Improper or Inadequate Maintenance		
G	Inadequate Plans or Procedures		
H	Inadequate Management		
J	Poor Housekeeping or Arrangement		
L	Safety Devices Not Provided		
R	Personnel -- Poor Selection or Improper Training for the Job		
T	Personnel -- Insufficient Supervision		
F	Improper or Inadequate Calibration	All actions and responsibilities accruing to those with responsibility for "hands on" operation of the facility.	P
M	Safety Devices Not Maintained		
N	Operator Error		
P	Failure to Follow Procedures		
S	Personnel -- Carelessness		
K	Equipment Failure or Faulty Equipment	The failure of a component or system within the facility not caused by personnel or error in the maintenance or operation of the facility.	F
A	Unavoidable -- Inherent Risk of Job which Could Not Have Been Reasonably Foreseen or Prevented	All events which are not related to a failure of either facility management, personnel, or the random failure of a component are unimportant to the Teknekron analysis and are grouped and designated as such.	O
B	Unavoidable -- Circumstances beyond Control; e.g., Natural Causes		
W	Causal Factor Not Determined		

As with LER data, these noncompliance ERCs are linked to real situations: in this case, citable occurrences. Citable occurrences, however, originate in two ways. First, they may stem from events occurring at the facility system level and reported by licensees in accordance with NRC's criteria for LER reporting (NUREG-0161). In some cases, inspector followup of these events results in items of noncompliance. For one of the licensees we studied, a significant number of LERs were identified as citable occurrences. In these cases, the inspection process is reacting to actual events.

The second way in which citable occurrences may originate is through detectable violations of license conditions. Under the current inspection system, detectability is a time function of:

- when the citable occurrence took place, and
- the inspection module under review.

Evaluation of citable occurrences as a function of time and according to their points of origin within the facility would lead to the identification of performance patterns. But the detection of these patterns is subject to the characteristics of the inspection program.

The NRC's modularized inspection program has its own pattern for detection of citable occurrences. The inspection modules are typically performed on a scheduled basis and some are performed repeatedly throughout the annual inspection cycle. The scheduling of some inspection modules is necessarily determined by facility status (plant shutdown for refueling). There are also I&E procedures (see MC 2515) that permit an inspection to be performed when required, independent of any preset schedule ("W" inspection code).

For these reasons, we found that the pattern of noncompliances detected by the existing inspection program is governed by the character of that program as reflected in the time dependency of the inspection modules. The inspection module becomes the "point of origin" for the detection of licensee performance patterns, a point often well removed from the actual event that created the citable occurrence. Since most module inspections occur relatively infrequently, the data produced through any single module, when viewed over time, are usually too scant to be meaningful. These factors tend to obscure a time-sensitive identification of patterns of deficient licensee performance. We conducted our analysis of the 766 file and the associated inspection reports in the light of these considerations.

Figure 9 shows the relationship to the licensee of each major dimension of available data (LER file and 766 file) that we have used in licensee performance analysis. The LER dimension more closely reflects the reality of the licensee's operation. The noncompliance dimension is a level removed, and noncompliances are detected through the filter of the inspection program.

Licensee Performance Profiles Based on Noncompliance Data

For each licensee studied, we constructed a performance profile based on the noncompliance data. In developing these profiles, we did not include noncompliance items cited in the physical protection area, since 1976 and 1977 marked the implementation of 10 CFR Part 73.55, attended by the difficulties associated with implementing any new regulation.

As explained above, the inspection module is the "point of origin" for the identifying patterns of licensee performance through noncompliances. Since most modules are inspected relatively infrequently, the data produced from any single module, when viewed over time, are usually too scant to be meaningful. To improve the density of the data, we took those noncompliances attributable to ERC-M and ERC-P and summed them to produce a profile of total licensee human performance as perceived by the regulatory process. While this summation potentially reduces the sensitivity of the data, it clearly improves its meaningfulness. When viewed from the perspective of the FPM model, a profile constructed in this way represents the aggregate deficient human performance as perceived by the inspection process.

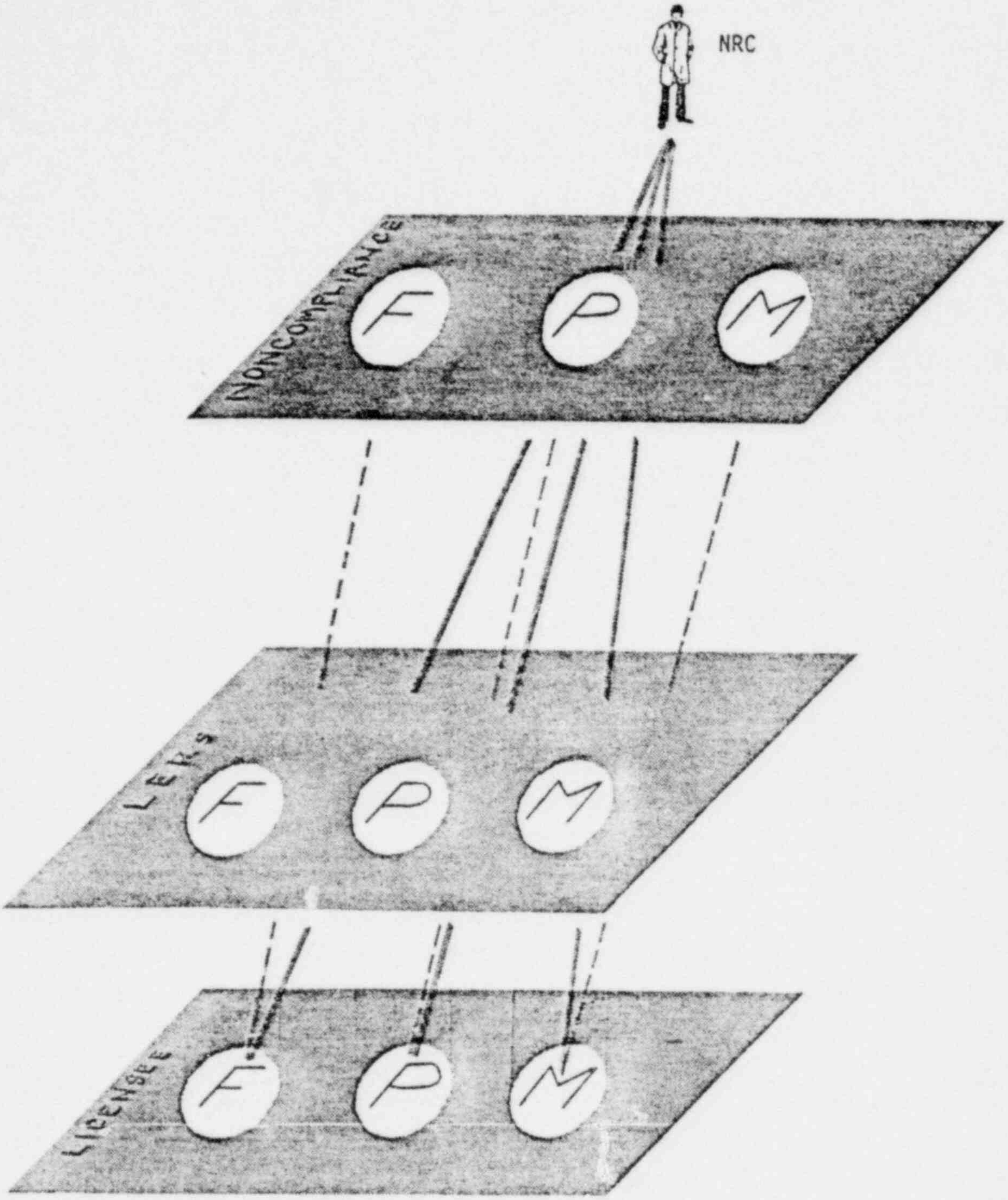


FIGURE 9

Data Dimensions for Performance Analysis

The noncompliance profiles we constructed for each of the three case study licensees are shown in Figure 10. The similarity between the profiles for Prairie Island Unit 1 and Point Beach Unit 1 is less striking than their difference from the Zion Unit 1 profile. The profiles of Point Beach Unit 1 and Prairie Island Unit 1, while unique to those licensees, are relatively similar in density, magnitude, and periodicity.

Figures 11, 12, and 13 show the noncompliance and LER profiles for each licensee we studied. Note that the vertical scale is different. Comparing each licensee's noncompliance profile to his LER profile provides an insight into the "performance" of the inspection program in handling different types of licensees. The total human noncompliance profiles are reasonably similar to the related profile of total human error in reported events for the "good" performers (Prairie Island Unit 1 and Point Beach Unit 1). However, the difference in apparent periodicity between the two profiles for Zion Unit 1 is substantial, and probably reflects the licensee's attempts to respond to regulation as well as the response of the regulatory process to the licensee. The apparent phase differences of the two profiles may be an indicator of the sensitivity of the interaction between the licensee and I&E.

In the case of the "good" performers, neither their total human noncompliance profiles nor their profiles of total events attributable to human error show sharp or sustained increases or decreases in numbers of events or noncompliances over time. The profiles exhibit a steady-state quality that can be termed the "noise" of operation. Further, the case studies in Appendix A show that Point Beach Unit 1 had very few instances of causally linked events, while Prairie Island Unit 1 experienced a somewhat larger number of causally linked events. However, both facilities appear to be reasonably free of systematic human error. But for Zion Unit 1, a "poor" performer, both the profile of total events attributable to human error and the profile of total human noncompliances show steep and sustained

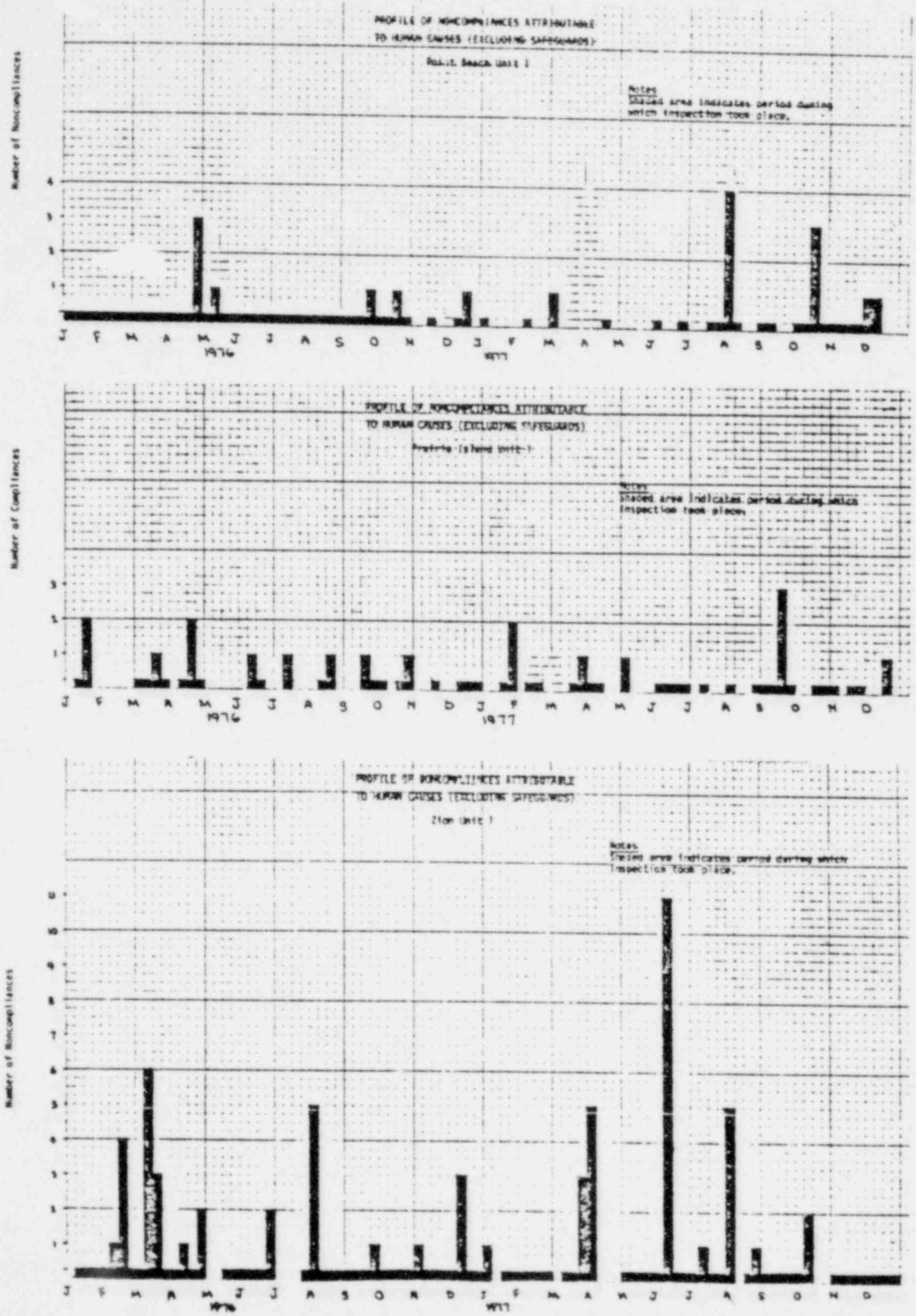
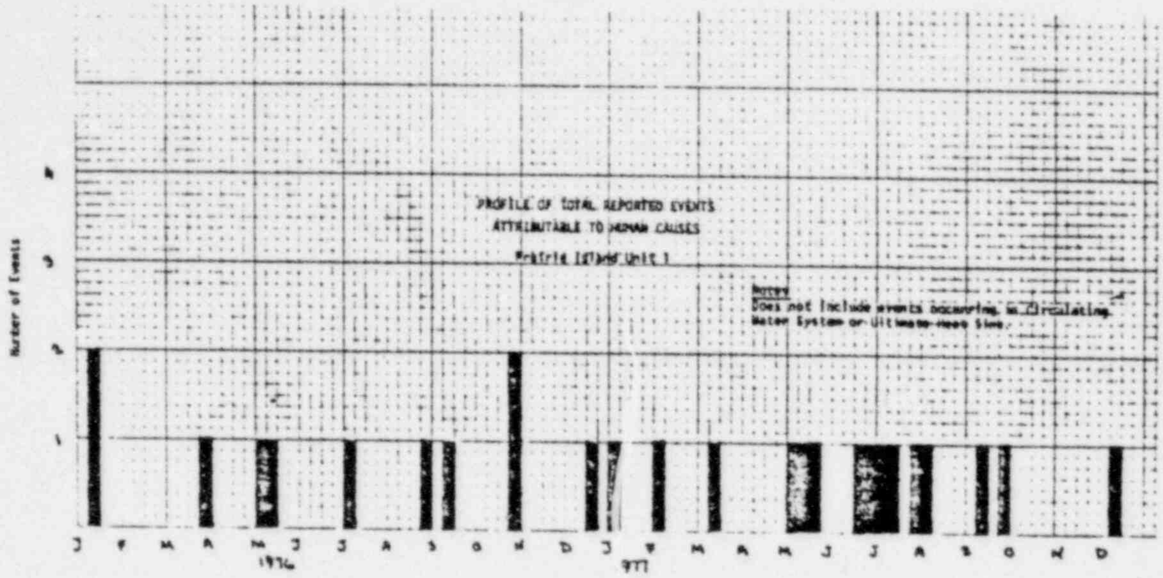


FIGURE 10

Comparison of Noncompliance Profiles



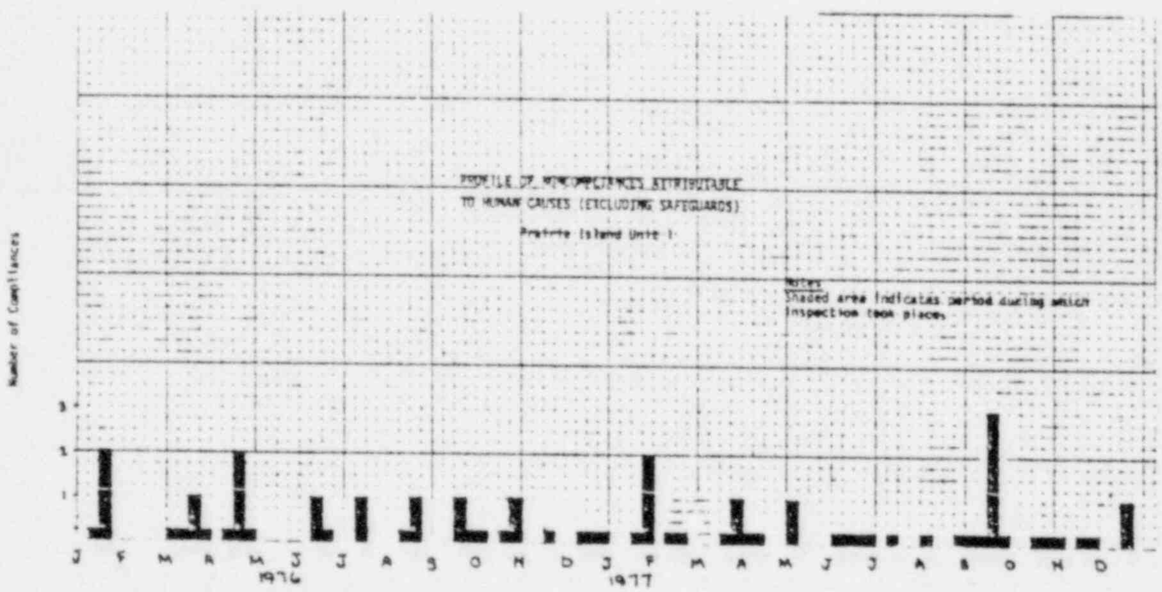


FIGURE 11

Prairie Island Unit 1 LER and Noncompliance Profiles

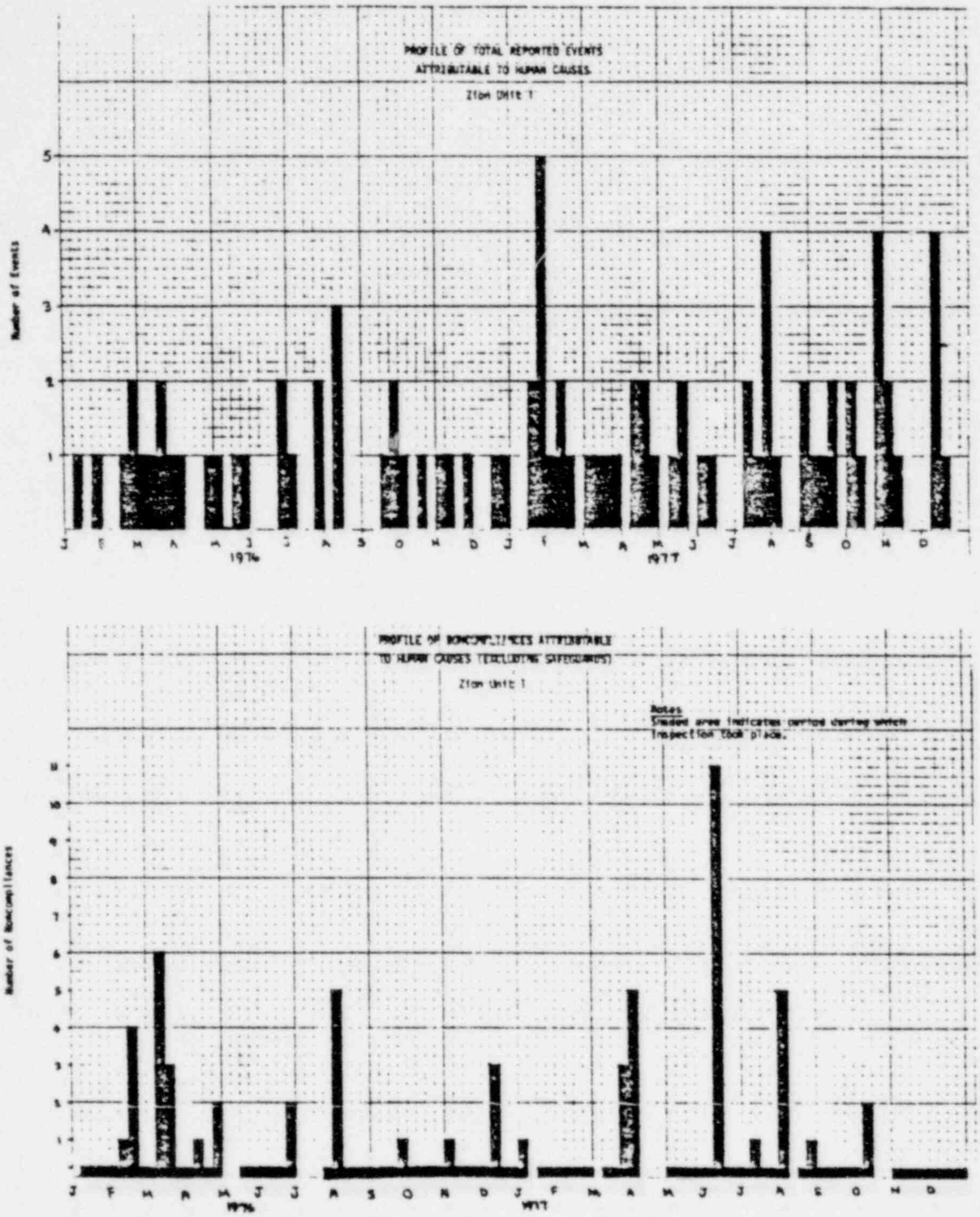
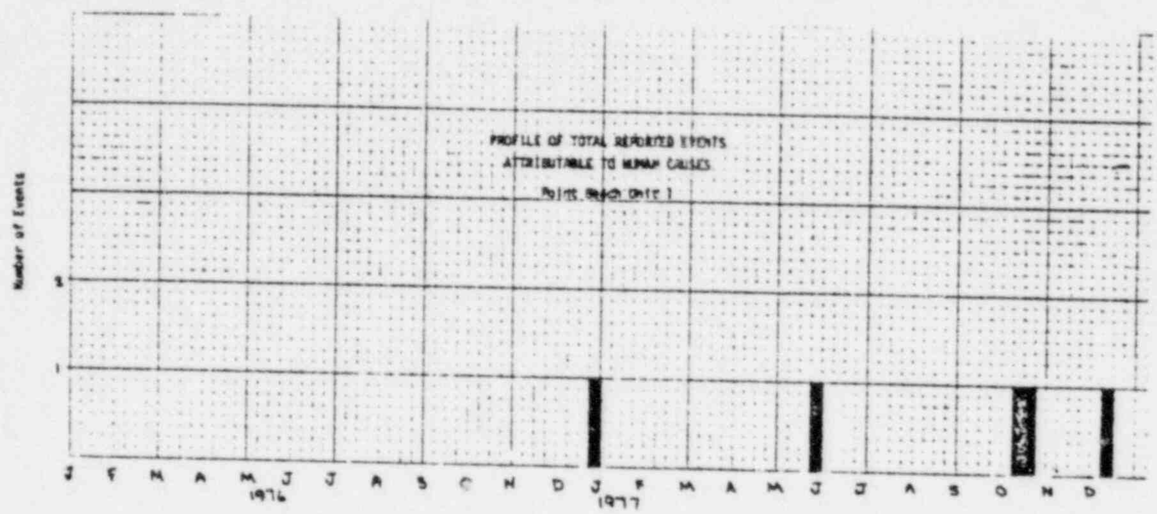


FIGURE 12

Zion Unit 1 LER and Noncompliance Profiles



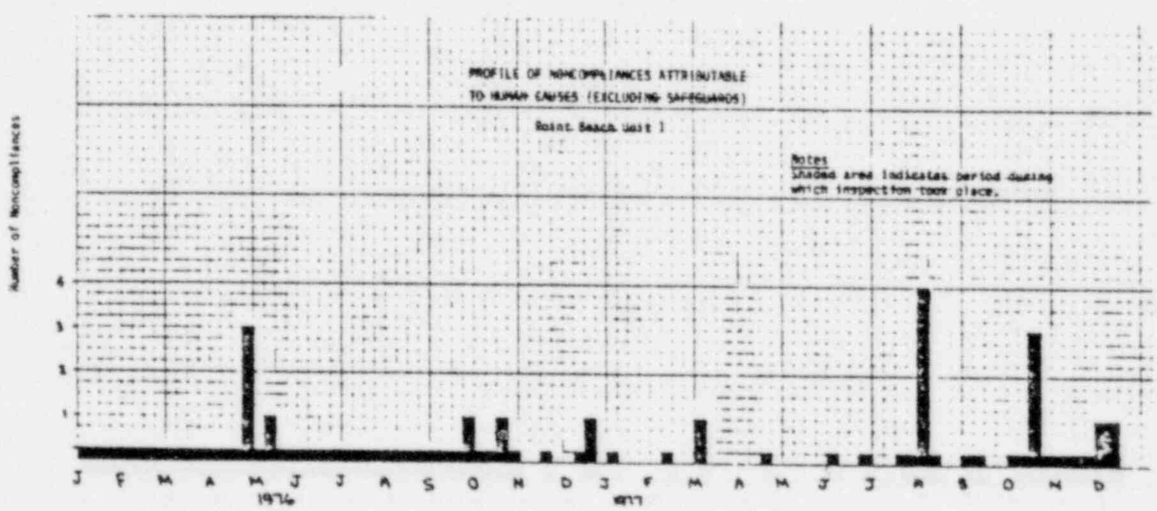


FIGURE 13

Point Be ch Unit 1 LER and Noncompliance Profiles increases or decreases in numbers of events or noncompliances over time. The case study shows that Zion Unit 1 had many causally linked events when compared to the "good" performers, which indicates that Zion's profile of total events attributable to human error is dominated by events due to systematic management deficiencies. While there is no direct basis for assuming that the profile of total human noncompliances displays causally-related defects in licensee performance (due to the modularized nature of the data), we believe that such defects were perceived by the inspection process. This is corroborated by a review of the NRC inspection reports of NRC management meetings with the licensee following incidents at the facility, as well as the inspector's perceptions of the licensee.

Analysis of 766 File and Associated Inspection Reports

As part of our review of the 766 file for each case study, we investigated the relationship of the 766 file to its written counterpart, the inspection reports. This "mini-audit" is briefly described in Section 3.3.3.1. The details of these investigations are provided in the case studies in Appendix A, in the form of summary matrices. Table 5 summarizes each case study matrix. Each matrix contains information on three specific relationships we felt were particularly important:

- The relationship of key 766 file data to the associated inspection reports: to use the 766 file data for analysis, we must know how well it agrees with the inspection reports.
- The relationship of inspector cues (LERs and licensee-identified items) to noncompliances: what guided the inspector in identifying citable occurrences?
- The relationship of the licensee to the regulatory process: his readiness to specify remedies to items of noncompliance, his action on previously identified enforcement items, and inspector agreement or disagreement with the licensee's reporting of LERs.

TABLE 5

SUMMARY OF COMPARISON OF 766 FILE DATA AND ASSOCIATED INSPECTION REPORTS FOR 1976 AND 1977

	Point Beach Unit 1	Prairie Island Unit 1	Zion Unit 1
Disagreement/ambiguity between I&E inspection report and 766 file noncompliance cause code	12%	20%	9%
Disagreement/ambiguity between 766 file noncompliance cause code and 766 file enforcement text	44%	37%	47%
Noncompliances associated with inspector cues (as percent of total noncompliances)			
LERs	0%	17%	32%
Licensee identified items	12%	11%	20%
Total	12%	28%	52%
Noncompliance remedies (as percent of total noncompliances) suggested by licensees in:			
Inspection report	36%	45%	50%
Followup letter	44%	31%	21%
Licensee action on previously identified enforcement items	Always complete	Complete (1 exception)	Deficient in one or more items, 70% of the time this was reviewed by inspect r.
Repeat noncompliances	0	0	5 (in 1976)
Serious events due to human error	0	0	3

1. Relationship of 766 File to Associated Inspection Reports

The level of disagreement between the 766 file noncompliance cause code and the associated inspection report details ranged from a low of 9% to a high of 20% (Table 5). This represents fairly good agreement and suggests that it would be possible to have inspectors gather data according to FPM model definitions. Because the FPM definitions are considerably more precise and offer less opportunity for ambiguous application than the 766 file noncompliance cause codes, we believe that the data gathered in this way would be reliable and consistent in character.

Disagreement between the 766 file noncompliance cause code and the 766 file enforcement text ranged from a low of 37% to a high of 47%. This indicates that the 766 text and associated noncompliance codes cannot provide a confident understanding of the circumstances surrounding a noncompliance. In most cases, we had to use the associated inspection report to gain insight into the cause of a noncompliance. However, we found strong agreement between the 766 file enforcement texts and the summaries of the noncompliance items in the inspection reports. Therefore, the major difficulty in understanding the actual cause of a noncompliance from the 766 file information lies in the interpretation and use of the 766 noncompliance cause codes; the enforcement text does not provide enough supporting detail. A study to determine how inspectors use these codes could help to substantially improve the codes' precision and make the 766 file data more useful in the future.

Data are coded on the 766 file input forms in the regions, and the inspection report is prepared simultaneously. A "stratified" statistical sampling program on a regional basis is required to determine the precise level of agreement that actually exists between primary 766 file data elements and associated inspection reports. This program would permit NRC headquarters to identify error-input sources into the 766 file and, at the same time, would indicate differences in regional attitudes toward the data base by illuminating the way in which the information is handled.

2. Relationship of Noncompliance Items to Inspector Cues

We next examined the relationship of noncompliance items to cues (LERs and licensee-identified items) to the inspector. These cues are an obvious source for identifying citable occurrences, but their use varied considerably from one licensee to another. For example, 32 percent of noncompliances at Zion Unit 1 were related to inspector followup of LERs; in contrast, this percentage was zero for Point Beach Unit 1 and 17 for Prairie Island Unit 1. The second source of inspector cues (licensee-identified items such as procedure changes) produced 20 percent of the total noncompliance count at Zion Unit 1, but only 12 percent at Point Beach Unit 1 and 11 percent at Prairie Island Unit 1.

As part of our analysis, we determined the overall inspection results (noncompliance items/100 module hours) by year for each licensee studied. These results are shown in Table 6. As stated earlier, we did not include inspection hours or noncompliances related to physical protection. Table 6 also shows the results from that part of the inspection process that detects noncompliances <u>without</u> using cues provided by the licensee. (These results show the ability of the unaided inspection process to detect noncompliances.) Note that for 1977, Point Beach Unit 1 shows a somewhat higher result (2.1) in non-cued yield than Zion Unit 1 (1.8). This is strikingly inconsistent with the overall performance patterns and case studies for these plants, which show that Point Beach Unit 1 is the better-managed facility of the two.

We believe that results of the kind shown in Table 6 may say more about the inspection process than about the licensees toward whom the process is directed. On the basis of these three case studies, it appears that the inspection process in its current form can make gross distinctions between licensees in terms of "good" and "poor" performance, if cues are utilized by the inspectors. But in the case of Zion Unit 1, the <u>perception</u> of "poor"

TABLE 6

INSPECTION RESULTS

Total Module Hrs.(1)	Point Beach Unit 1	Prairie Island Unit 1	Zion Unit 1
1976	490	464	904
1977	378	595	1032
Total N/C's(1)			
1976	6	10	37
1977	10	8	31
Overall Results: N/C's per 100 Module Hrs.			
1976	1.2	2.2	4.1
1977	2.6	1.3	3.0
Total N/C's not due to inspector cues(1)			
1976	5	7	12
1977	8	4	19(2)
N/C's not due to inspector cues per 100 module hrs(3)			
1976	1.0	1.5	1.3
1977	2.1	0.7	1.8

NOTES.

(1) Does not include time or noncompliances related to physical protection.

(2) Inclu s six noncompliances for which reports were not available. These noncompliances were not related to physical protection; including them as uncued findings gives maximum weight to the inspection process.

(3) Module hours spent on followup of lice. see-provided cues were not removed, since noncompliances resulting from cues were spread rather uniformly throughout the inspections and time spent specifically on these items was seldom separately shown.

performance through the inspection process, even using cues, appears to have lagged the timely performance shown in the LERs. The inspection process also appeared to have no particularly sensitive licensee performance indicator similar to the causally linked events of the LER data analysis. This apparent lack, together with the apparent usefulness of licensee-provided cues, tends to support the view that the inspection process in its current form may lack the sensitivity or direction needed to foster licensee performance analyses that are both accurate in terms of quality and at least approximately correct in terms of magnitude.

3. Relationship of the Licensee to the Regulatory Process

Ideally, the licensee/regulatory relationship is interactive. On one hand, NRC must monitor the level to which licensees adhere to required operating and other functional states and conditions. It is also NRC's obligation to cite departures from license conditions and to impose sanctions if these are considered necessary and appropriate. Some may argue that in the interest of public welfare, the agency should provide help and guidance to the licensee if required, even though this function clearly lies outside of the literally interpreted regulatory domain.

On the other hand, the licensee's relationship to the regulatory agency obligates him to:

1) operate his facility in such a way that he violates the original license conditions to the least possible extent; and

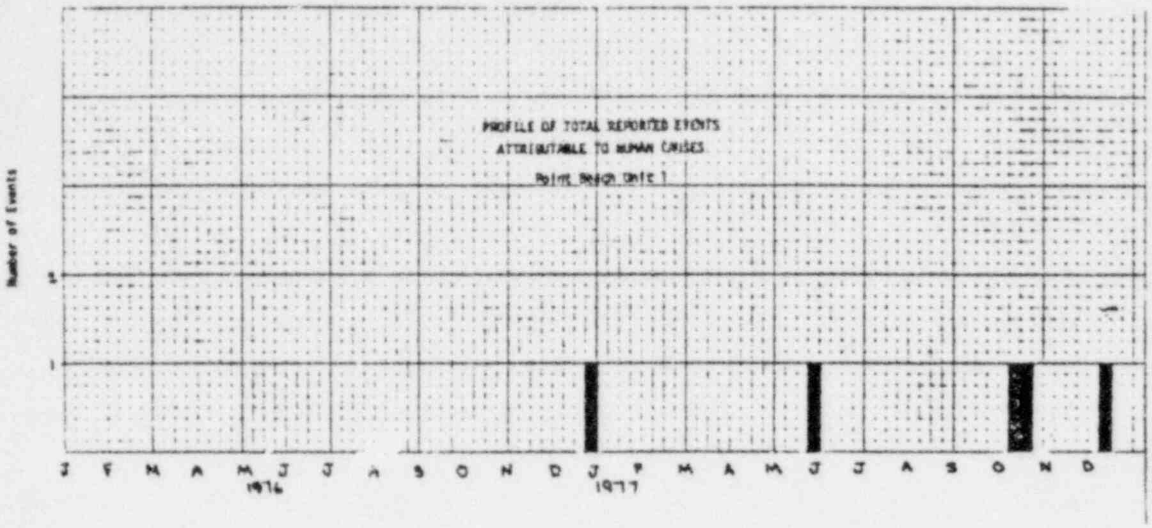
2) institute adequate remedial measures in the least possible time period if such violations occur.

As the licensee fulfills these obligations, it is wholly immaterial whether a violation is initially identified by the licensee or by the NRC inspector. The key factor is the licensee's willingness and ability to respond effectively to the identified situation.

The concepts set forth above are expressed in extremely general terms. In the following discussion we will show how these concepts can be specifically applied to making accurate distinctions, on the basis of currently available data, between licensees who may be considered "good performers" and "poor performers."

Although the licensee is obligated to minimize the frequency of his departure from operating license conditions (in the case of the "perfect performer" this departure would be zero), it is inevitable that "good performers" as well as "poor performers" will experience events, noncompliances, and other lapses. We may expect such lapses, whether identified by the licensee or by NRC I&E, to occur with greater frequency in the case of "poor" as opposed to "good" performers. But it does not necessarily follow that <u>numbers</u> of lapses provide reliable absolute indicators of overall licensee performance levels. We cannot assume that, because Facility A has twice as many lapses as Facility B over a similar time period, that Facility A is only half as safe as Facility B. From both LER and NRC inspection data, this study shows that lapse <u>recurrence</u> is a far more sensitive indicator of licensee performance (particularly managerial performance) than lapse frequency as such. The data presented in Table 6 show that lapse frequency cannot stand alone as a performance indicator. The overall inspection result in 1977 was 3.0 for Zion Unit 1; for Point Beach Unit 1 it was 2.6. These two numbers are quite similar. But the performance profiles based on the LER data shown in Figure 14 make it immediately apparent that the performance difference bewteen these two licensees is substantial, a difference that is obliterated by the overall inspection result indexes.

While Table 6 shows that frequency alone is a poor performance indicator, Table 5 shows that recurrence is far more sensitive. Table 5 shows that all three licensees are similar in their readiness to suggest remedies to noncompliance items. But when we examine 766 file data on the recurrence of identical noncompliance items, we see that five such instances occurred



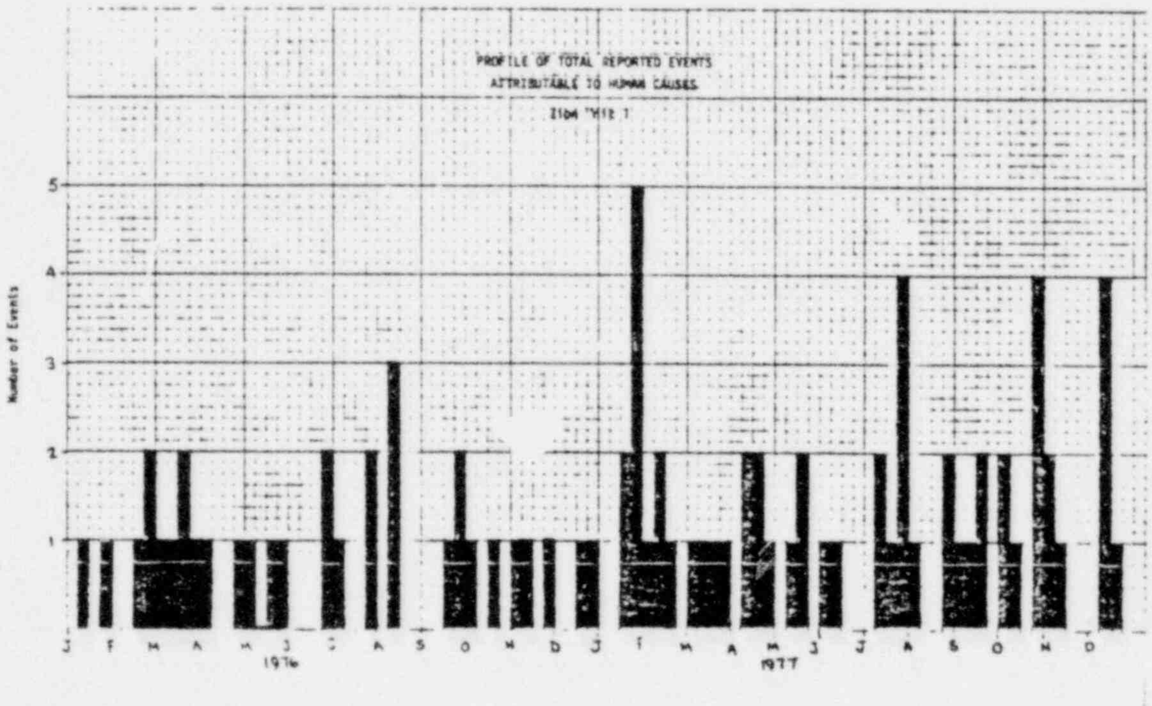


FIGURE 14

LER Profiles for Point Beach Unit 1 and Zion Unit 1 in 1976 at Zion Unit 1; Point Beach Unit 1 and Prairie Island Unit 1 had no repeat noncompliances.

The record of licensee action on identified items in Table 5 supports this. Point Beach Unit 1 actions were always complete, and with one exception, Prairie Island Unit 1 actions were also complete. But at Zion Unit 1 at least one action deficiency was noted during 70 percent of the inspections in which licensee followup was performed. Recurrence is an inverse measure of a licensee's ability and willingness to respond effectively. These findings on the relative sensitivity of frequency and occurrence as performance indicators are consistent with the licensee performance patterns developed from the LER data.

Available Licensee Performance Indicators

Based on the previous discussion, these indices provide a context for licensee performance:

- Percentage of noncompliance items identified due to inspector cues (followup of LERs and licensee-identified items).
- Percentage of noncompliance items for which the licensee has proposed remedies. This is a measure of stated licensee responsiveness to the inspection process.
- A "stratified" and regionalized sample to determine the error that actually exists between the 766 file data and associated inspection reports. This will indicate differences in regional attitudes toward the data base as well as demonstrate the quality of the data being used for performance evaluation.

Once this contextual information is available, useful indicators of licensee performance are:

- Licensee action on identified noncompliance items. This indicates actual licensee willingness and ability to comply once the problem is identified.
- Repeat noncompliance items. These reflect licensee ability to implement changes to and maintain the program.

- A profile of licensee performance based on the noncompliances attributable to human causes. This measures perceived aggregate deficient licensee performance. However, when this profile is compared with the associated LER profile of total reported events attributable to human causes, it provides insights into the licensee's relationship to regulatory process, the licensee's response to the process, and perhaps the applicability of the process to the licensee.

3.4 SUMMARY OF THE THREE CASE STUDIES

We have illustrated our use of the LER and noncompliance data with examples from the three case studies. From the outset of this project it was clear that case studies were necessary to empirically test the validity of the chosen approach. Since our approach was to develop a comprehensive model and procedure applicable to all classes of licensees, we chose to perform case studies of operating power reactors to test the FPM model, methodology, and performance indiactors against the most complex of NRC's licensees. Further, the data available for operating power reactors are the most complete.

The full case studies are presented in Appendix A. The rationale for choosing which licensees to study and a summary of the results of those studies are presented here so that the main body of this report can stand alone.

Selecting the Case Studies

To eliminate any possible regional effects that could diminish the meaningful comparison of one case study with another, we performed all the case studies in one NRC region. To prevent the possible bias of cross-NSSS vendor comparison, we searched for facilities using the same equipment. Third, based on discussions with NRC personnel, we felt that any facility must have been operating for more than two years, to prevent a "learning curve" effect from destroying meaningful comparison and possibly obscuring the patterns or indicators that might otherwise be evident in a mature facility.

Finally, we decided to study at least two facilities, one perceived by NRC as a "weak" performer and the other as a "good" performer. This provided us with the opportunity to empirically identify patterns and indicators related to each performance category ("poor" and "good"). It also offered the chance to gain insight into underlying causal factors associated with the dichotomy of performance.

For these reasons, we selected Zion Unit 1 and Point Beach Unit 1. Both are in Region 3, both are Westinghouse plants, and last, both had more than two years of operating experience by the beginning of 1976. When we discussed our choices with Region 3 management, it was mentioned that the differences in technical specifications and reporting requirements between Zion Unit 1 and Point Beach Unit 1 were considerable. Region 3 felt that we should consider studying a third performer with reporting requirements and technical specifications similar to Zion Unit 1, and suggested Prairie Island Unit 1. Consequently, we studied three licensees--Zion Unit 1, Point Beach Unit 1, and Prairie Island Unit 1. This gave us the additional opportunity to begin to examine the impact of differences in reporting requirements and technical specifications on the FPM model and methodology.

Performing the Case Studies

We performed the case studies in accordance with the FPM model and methodology discussed in Section 3.2, and we analyzed the LER file data and the 766 file data as described in Section 3.3.2.2 and 3.3.3.2 of this report. The study period covered calendar years 1976 and 1977, in order to produce profiles extending over a sufficient length of time to allow potential changes in performance to be seen and assessed. In any ongoing performance analysis, the study period should obviously be current, and each of these three case studies can be readily updated.

Presenting the Case Studies

Each case study is presented in two separate parts that reflect the two different data dimensions--LER data and 766 data--used in the study. This allows the reader to gain an appreciation of the types of insights each data source provides as well as an appreciation of the sensitivity of each source to specific aspects of licensee performance. Performance profiles and supporting data sheets help the reader gain insight into the foundations of the case study effort as well as an appreciation of the study details.

Summary of Case Study Conclusions

The FPM model and methodology, using existing LER and 766 file data, appear to have both the capacity and sensitivity to differentiate "poor" from "good" performers. Figure 15 presents the profiles of total reported events attributable to human causes for the three licensees; the profiles for Prairie Island Unit 1 and Point Beach Unit 1, the "good" performers are clearly different from that for Zion Unit 1. Figure 16 shows the profiles of noncompliances (excluding safeguards) attributable to human causes, and again the differences are clear.

We found the LER file data essential to gaining insight into why the licensees perform as they do. As discussed in Sections 3.3.2.1 and 3.3.2.2, LERs promptly report real events occurring within facility systems. This close link to the "plant operating reality" offers the insight into management and personnel response to actual situations. The 766 file data was a less meaningful and sensitive performance indicator than we had anticipated at the start of our work. The cause codes in the data file are not precise and their use sometimes reflects inspectors' interpretations; the enforcement text is often too brief to establish the actual content of a noncompliance. Also, the discovery of noncompliances through the inspection program is often widely separated in time from their actual occurrence, due to the

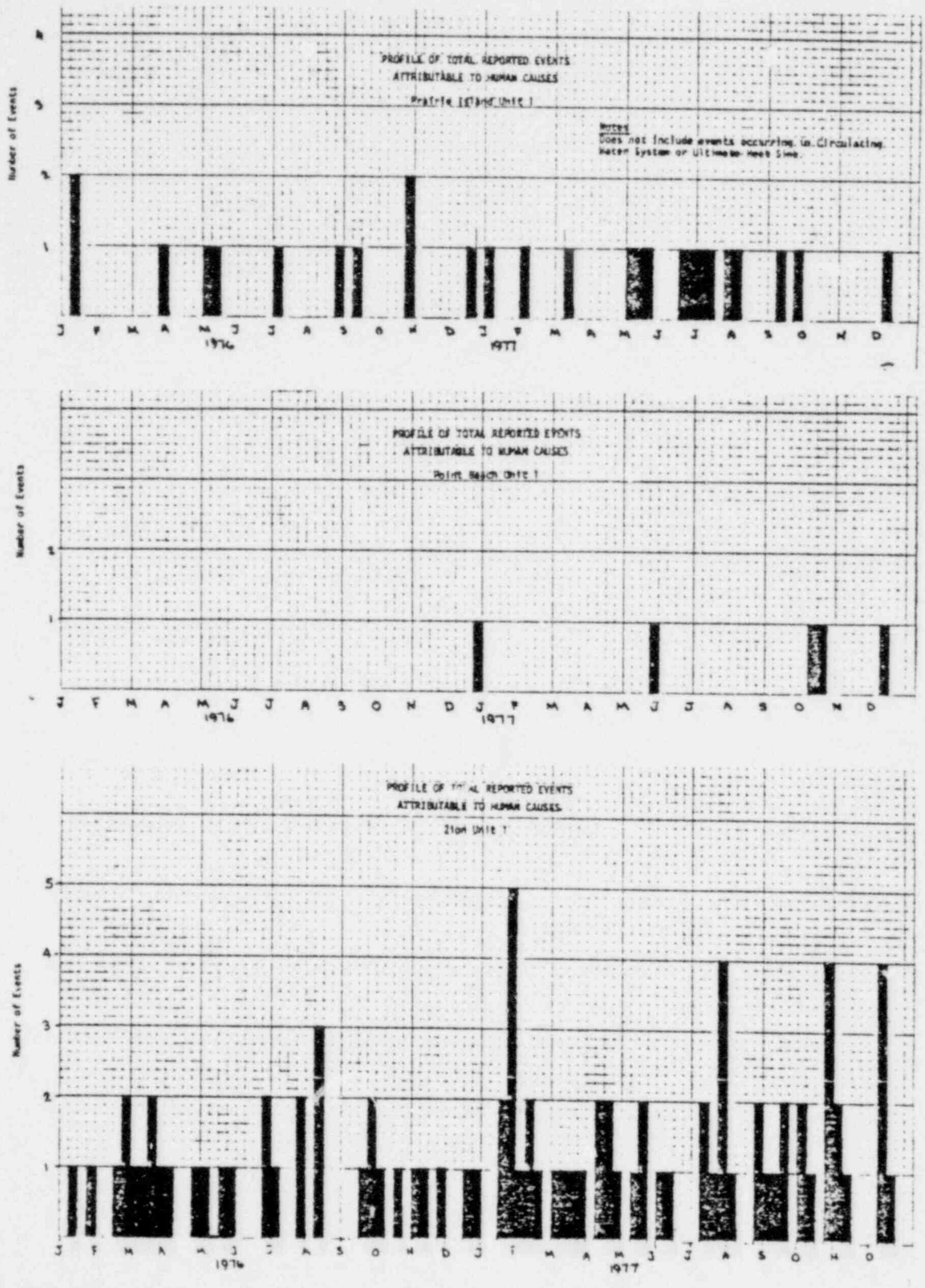


FIGURE 15

Comparison of LER Profiles

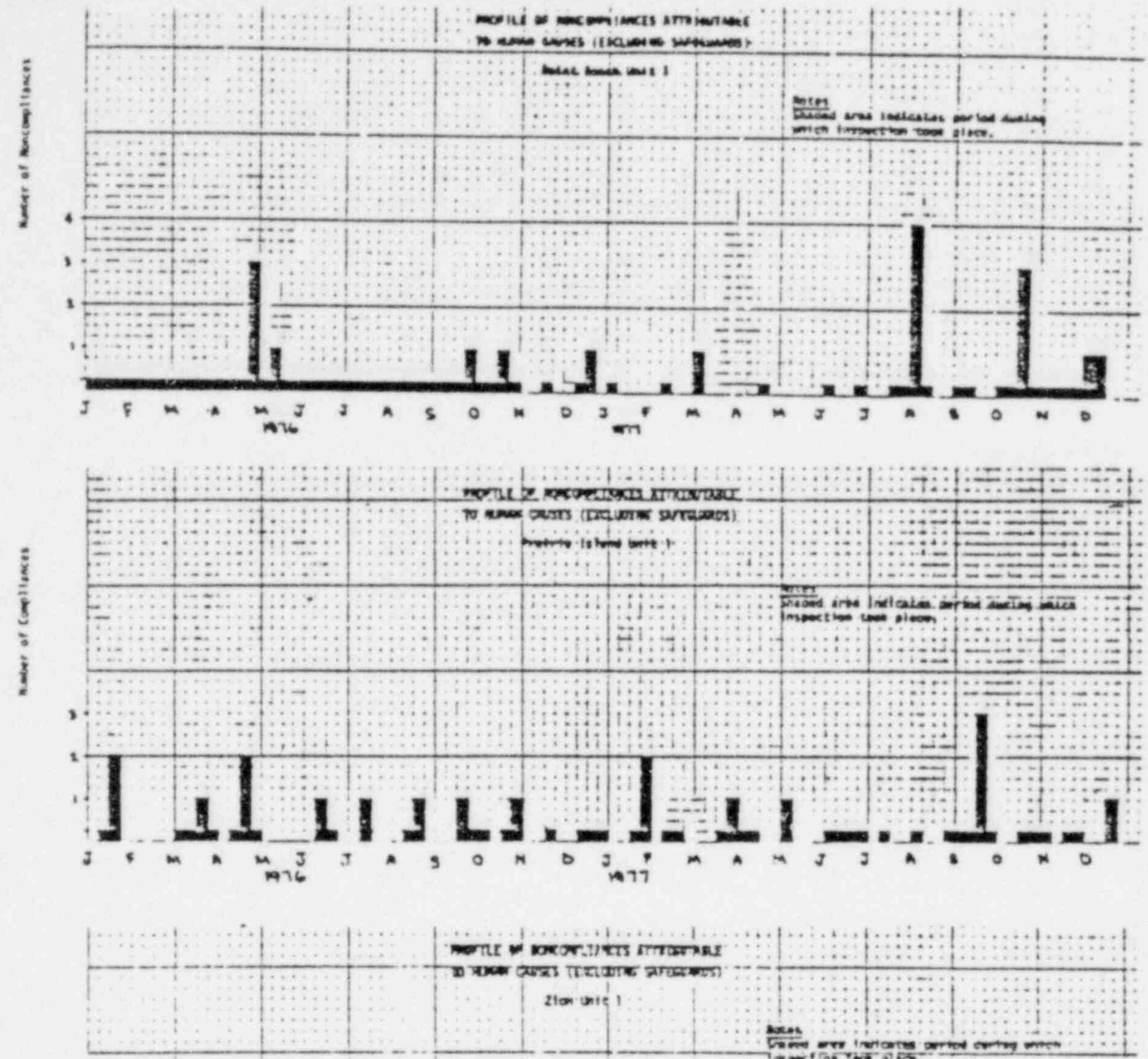


FIGURE 16

Comparison of Noncompliance Profiles

structuring of the program into time-dependent modules. These findings are discussed fully in Sections 3.3.3.1 and 3.3.3.2.

Differences in reporting requirements and technical specifications appeared to have little impact on the performance analysis results. We had expected little impact, since the FPM model is not inherently influenced by differences in technical specifications. But the empirical proof was in the performance profiles, as shown in Figure 17. The LER performance profiles for Point Beach Unit 1 and Prairie Island Unit 1, with different technical specifications, were relatively similar to each other. Zion Unit 1 technical specifications are similar to those for Prairie Island Unit 1, but Zion's LER profile is substantially different from both Prairie Island's and Point Beach's. Table 2, on page 38, establishes that technical specifications had little effect, at least for these three licensees. Further case studies will provide more indication of the sensitivity of the model to reporting and technical specification differences. We also expect that case studies of BWRs will permit comparisons that have until now been difficult.

Finally, we found that comparing the LER profile and noncompliance profile for a licensee provides insight into the capability and effectiveness of the regulatory process in managing the licensee's performance. This regulatory/licensee relationship may vary from region to region. Figure 18 shows these profiles for Zion Unit 1: the differences in phasing and frequency between the LER and noncompliance profiles are apparent, and the LER profile continues to show high levels of human error. Figures 19 and 20 show the profiles for Point Beach Unit 1 and Prairie Island Unit 1, where phasing and frequency are more similar.

3.5 LICENSEE PERFORMANCE ANALYSIS AND THE PERFORMANCE APPRAISAL TEAM PROGRAM

While the outlines and goals of the Performance Appraisal Team (PAT) Program are reasonably firm, the actual activities PAT will perform to meet those

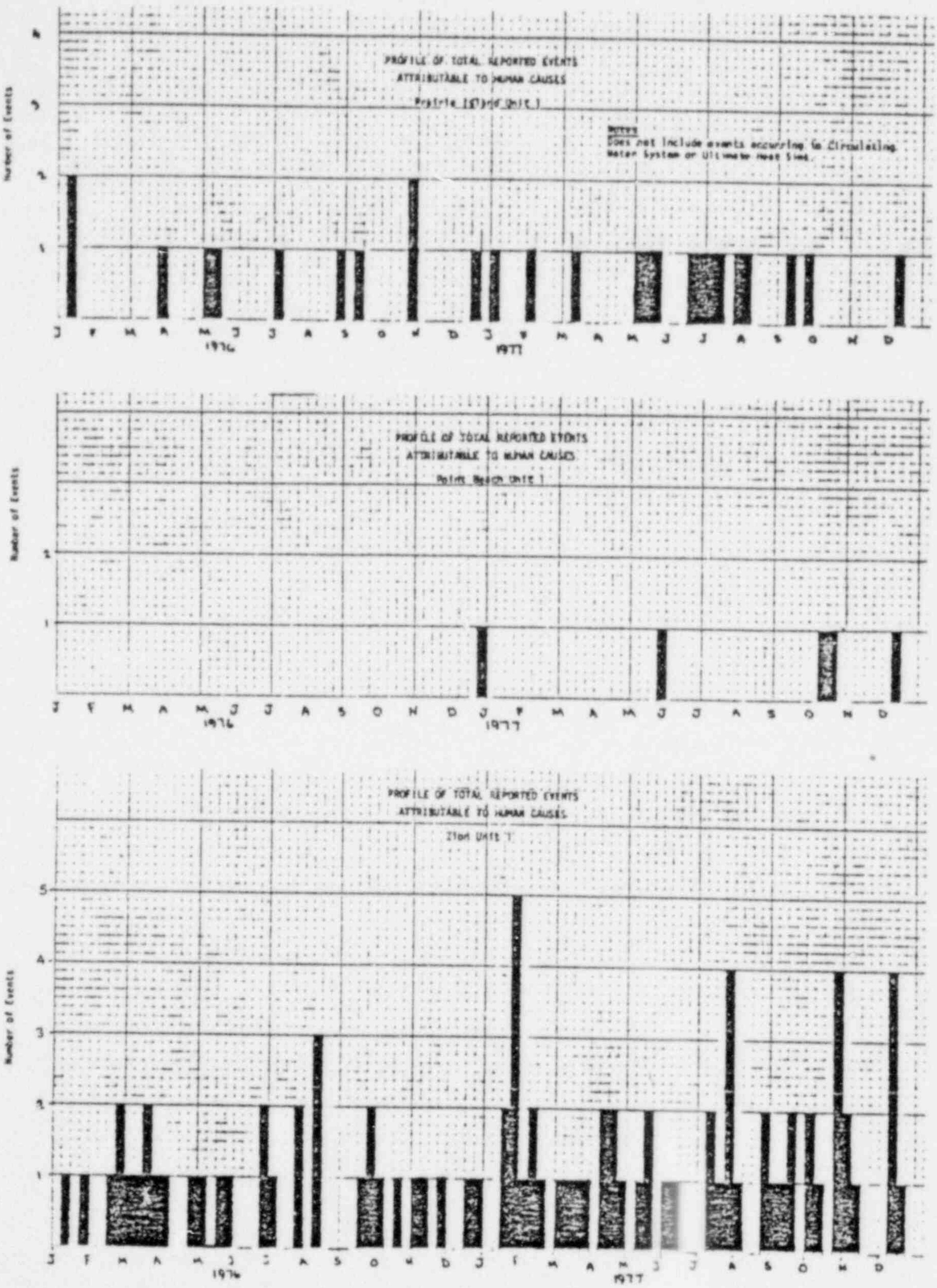


FIGURE 17

Comparison of LER Profiles

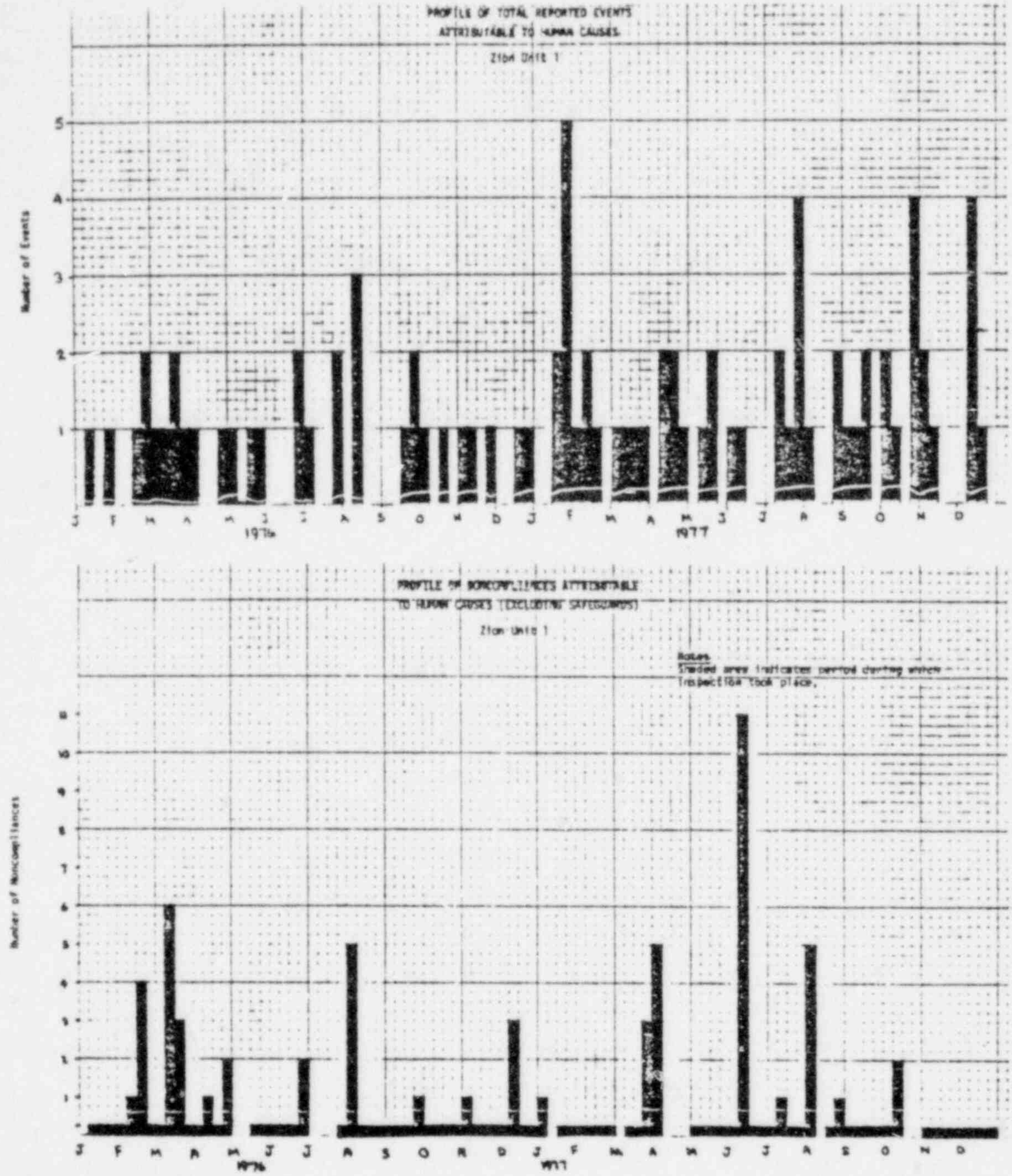


FIGURE 18

Zion Unit 1 LER and Noncompliance Profiles

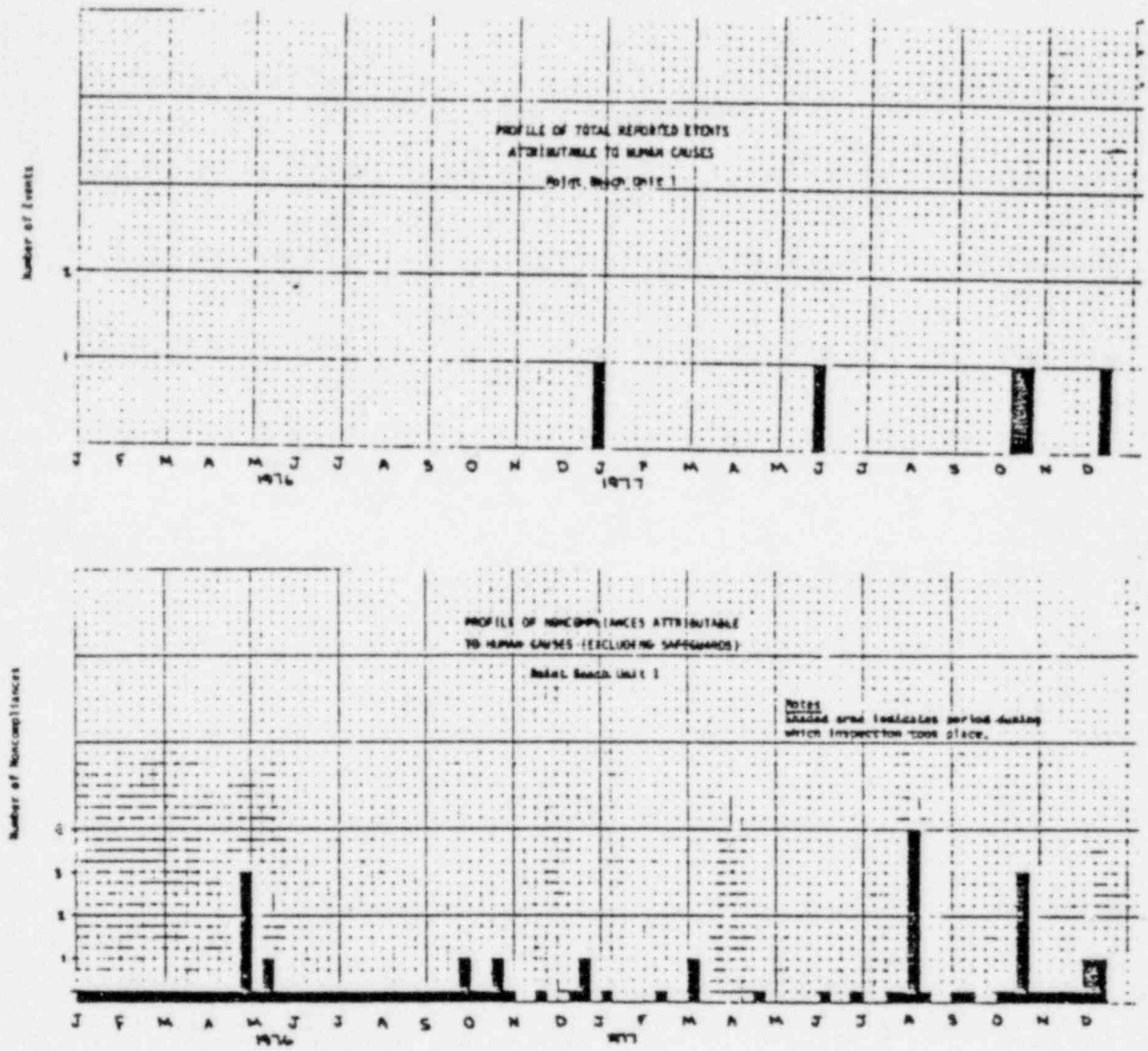
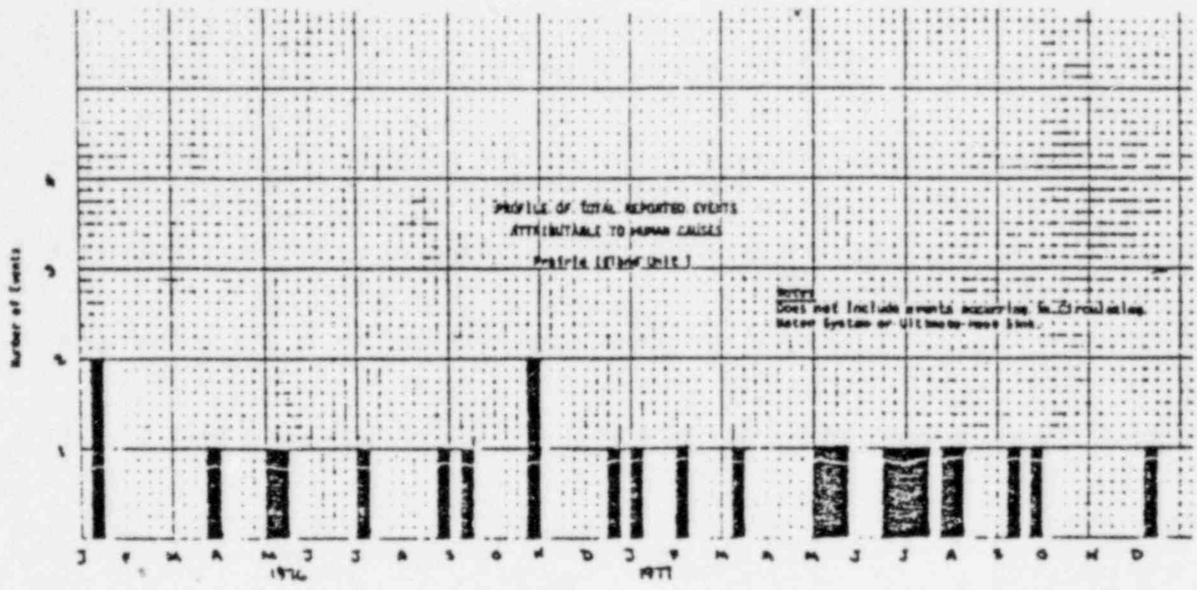


FIGURE 19

Point Beach Unit 1 LER and Noncompliance Profiles



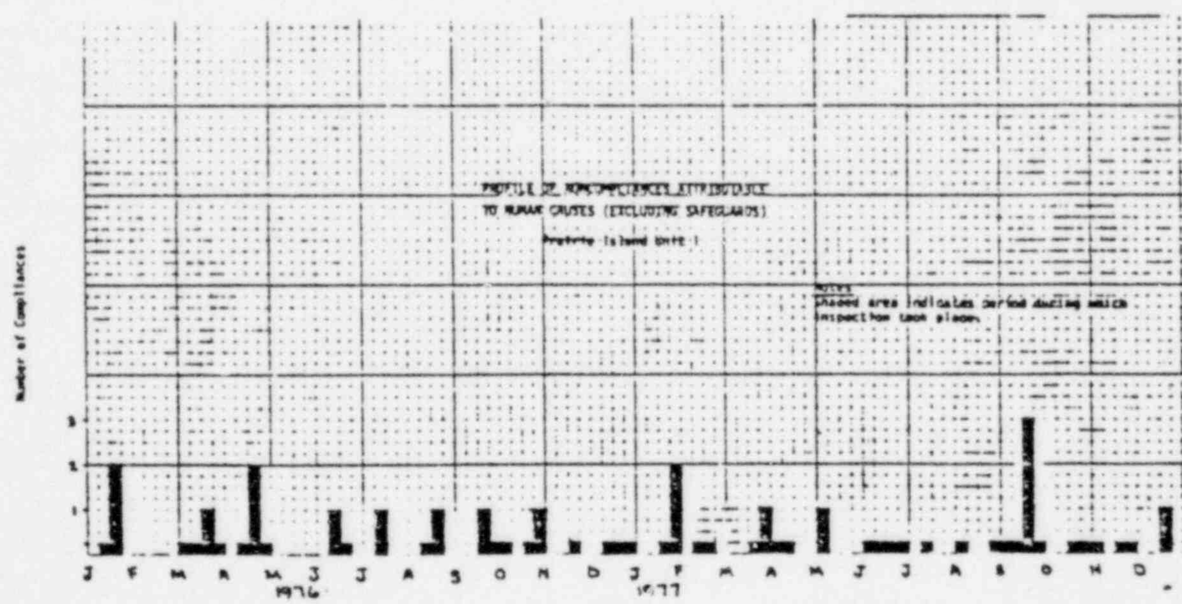


FIGURE 20

Prairie Island Unit 1 LER and Noncompliance Profiles

goals are not. Licensee Performance Analysis seems to have substantial links to PAT.

PAT has a dual purpose--to provide national perspective in analyzing licensee performance and to assess the effectiveness of NRC's own inspection program. At this time, PAT personnel have begun to devise the methodology they will use in Phase I of their program. Phase I is to include a subjective evaluation of plants, probably attempting to place them in "high, medium, and low" categories. This subjective evaluation will be based on the results of management inspections, routine inspections, and the resident inspection program. Each inspector will complete an evaluation sheet estimating the performance quality of each power reactor they visit. PAT will use these evaluations and other factors, such as the number of noncompliances, to arrive at a subjective evaluation of each plant.

Using the FPM model and the licensee performance analysis methodology can augment or replace the subjective evaluation of licensees. At a minimum, it should serve as an input to the PAT program. Performance analysis can also serve as a tool for evaluating the effectiveness of the inspection program in improving the performance of individual licensees. It is clear that the NRC regions differ in their management styles; these differences are reflected in varying results (number of noncompliances generated per 100 hours of inspection) and in varying methods of allocating inspector manpower. Performance analysis through the FPM model can help determine whether the inspection program is effective by comparing the profile of licensee response to events and the profile of NRC noncompliances to see the relationship between them. Ideally, action taken by NRC should improve the licensee's response: this is practically a definition of an effective program.

Presentation of License Performance Analysis

Continuously-updated and accessible licensee performance analyses could be highly valuable in directing the attention of regional personnel and the PAT teams to those licensees whose performance requires improvement.

The most obvious possibility is to place an interactive computer terminal in each region and at PAT headquarters, where personnel could immediately see the current performance profile for any licensee. The data base would be continuously and automatically updated through links with the LER and 766 files.

The "Rainbow Book" format is a second possibility. Figures 21a-c offer one possible format.

3.6 APPLYING THE MODEL TO EACH CLASS OF LICENSEE

3.6.1 Tailoring the Model

In our proposal, we stated that we would first develop a comprehensive assessment methodology designed to handle the most complex class of licensees--the operating power reactors. We also indicated that by deleting or combining elements, the same methodology could be applied to less complex licensees (materials licensees). The FPM model represents the "general" licensee to ensure that consideration of possible performance indicators would be both systematic and comprehensive.

Applying the FPM model to operating power reactors, we found that the model offered insight into the reasons for performance and was sensitive to actual differences in licensee performance. The model is equally applicable to less complex classes of licensees, since the general model elements ("F", "P", and "M") have clear parallels in each licensee category. [illegible]ing the medical materials licensee group as an example, "F" is the radioactive source and the supporting physical facility. "P" is the technicians and doctors using and calibrating the device, and "M" is the hospital or clinic management responsible for operations other than "hands on."

LICENSEE PERFORMANCE ANALYSIS - 1976 and 1977

Point Beach Unit 1

INDEXES

Per cent noncompliances due to inspector cues: 12%

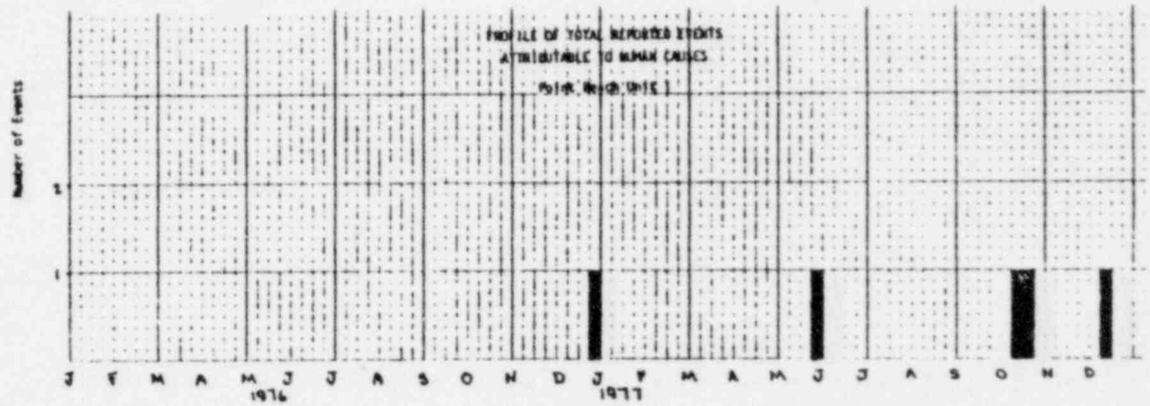
Licensee remedies proposed in inspection report: 36%

Per cent regional error in 766 system input: not available

INDICATORS

Licensee action on identified noncompliance items: always complete

Repeat noncompliances in two-year period: 0



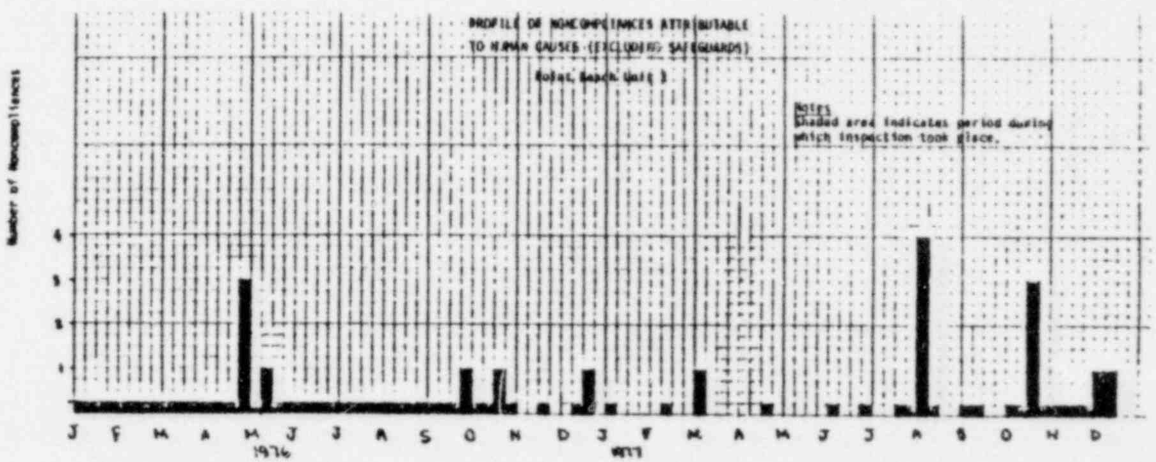
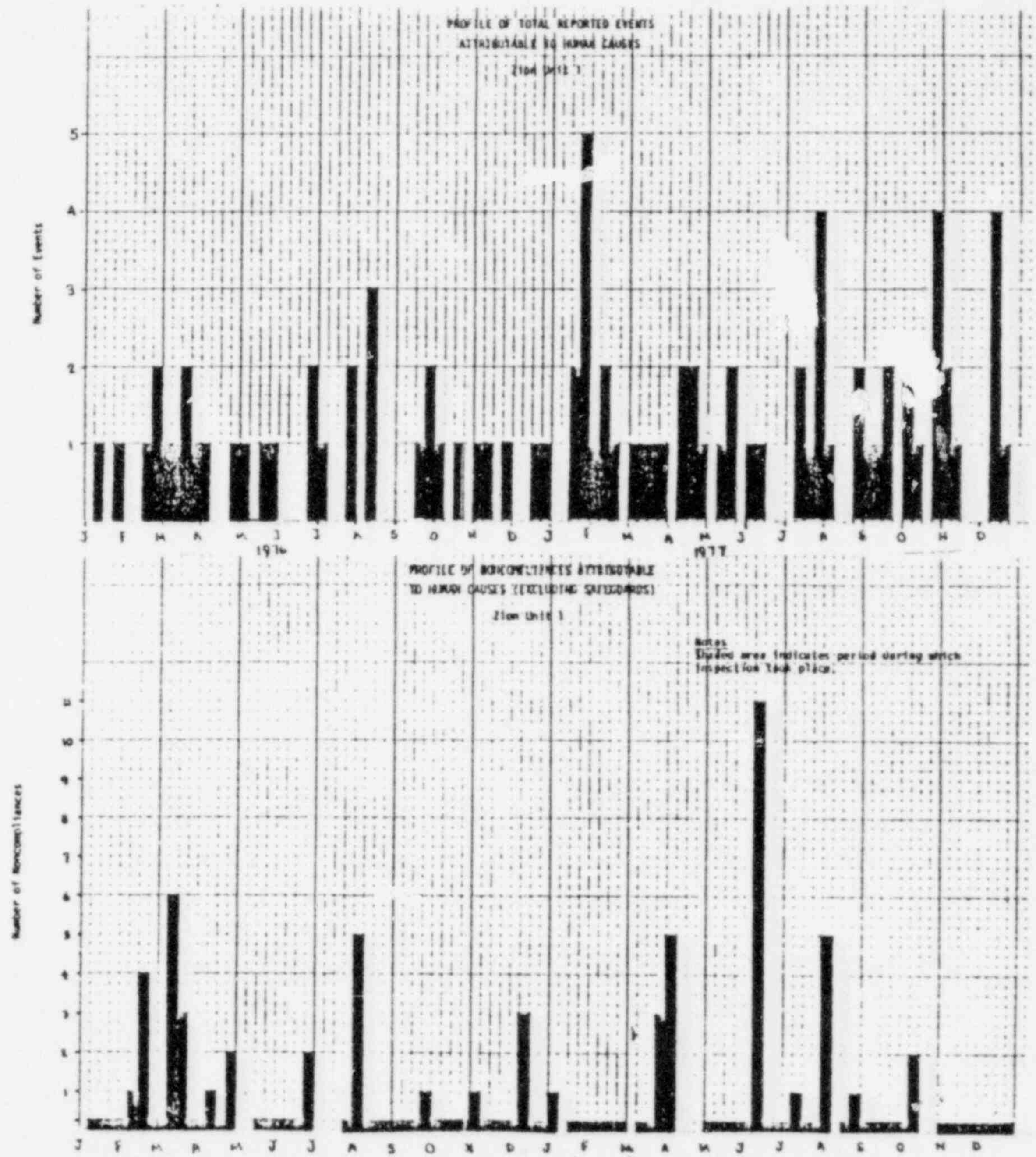


FIGURE 21a

LICENSEE PERFORMANCE ANALYSIS - 1976 and 1977

Zion Unit 1



INDEXES

Per cent noncompliances due to inspector cues: 52%

Licensee remedies proposed in inspection report: 50%

Per cent regional error in 766 system input not available

INDICATORS

Licensee action on identified noncompliance items: 70% deficient

Repeat noncompliances in two-year period: 5

FIGURE 21b

LICENSEE PERFORMANCE ANALYSIS - 1976 and 1977

Prairie Island Unit 1

<u>INDEXES</u>

Per cent noncompliances due to inspector cues: <u>28%</u>

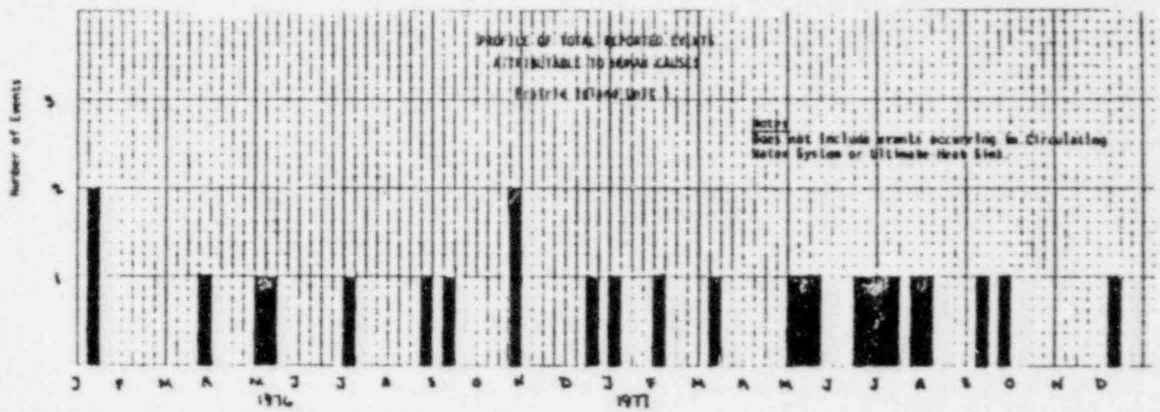
Licensee remedies proposed in inspection report: <u>45%</u>

Per cent regional error in 766 system input: <u>not available</u>

<u>INDICATORS</u>

Licensee action on identified noncompliance items: <u>complete (1 exception)</u>

Repeat noncompliances in two-year period: <u>0</u>



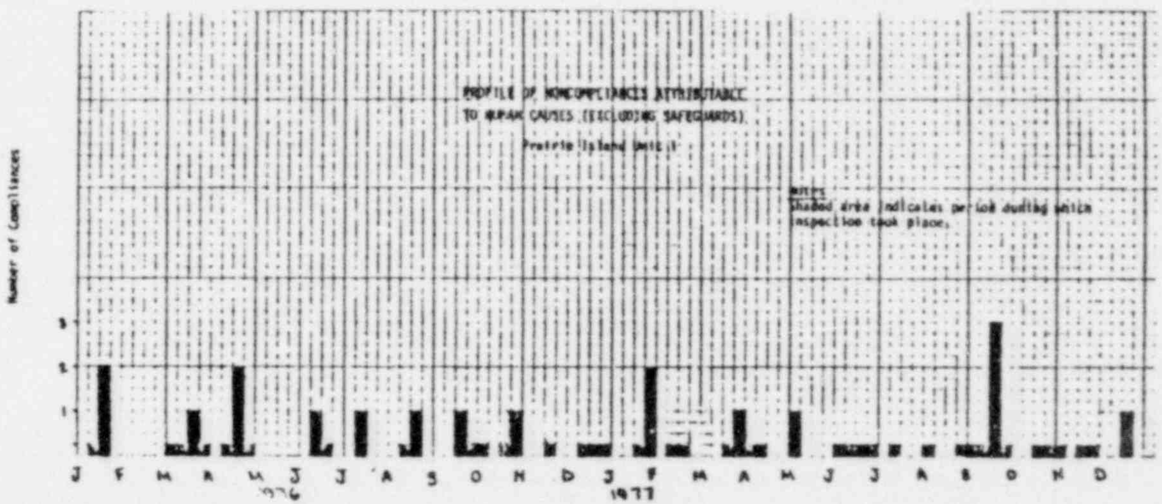


FIGURE 21c

3.6.2 Performing the Assessment

There is no question that the model is fundamentally applicable to each individual licensee. But whether it is possible to use the model to analyze performance of any class of licensee depends on:

- the availability of necessary data, and
- the availability of manpower resources to perform the analysis.

Availability of Data

Availability of data is briefly discussed in Sections 3.3.2.1 and 3.3.3.1. Here, we summarize whether sufficient data are available to make performance analysis possible for each class of licensee. The stress here is on computerized data in the LER and 766 files, since use of non-automated data, while possible, is less practical.

Operating Power Reactors

There are sufficient data in the 766 file and the LER file to analyze the performance of each licensee in this class.

Reactors Under Construction

For the 51 sites on which 93 reactors are under construction, there are only 78 construction deficiency reports for 28 sites in the LER file for 1976 and 1977. The rest of the construction deficiency report exist in written form (as 50.55e reports), primarily in the regions. Without resorting to the regional reports, the LER data are too scant to be used in performance analysis.

The 766 system contains data from 1,997 inspection reports in reactors under construction per year. This data density is probably adequate for performance analysis, keeping in mind the *caveats* of Section 3.3.3.2.

Test and Research Reactors, Fuel Facilities, and Materials Licensees

For the 93 test and research reactors, there are data from 247 inspection reports in the 766 file for 1976 and 1977, an average of 1.3 reports for each reactor per year. This data density is probably not adequate for performance analysis.

There were data from 995 inspection reports in the 766 file for the 38 fuel cycle facilities in 1976 and 1977, an average of 13 reports per facility per year. This data density is adequate for performance analysis. There were 4,737 inspection reports prepared for the more than 9,600 materials licensees in 1976 and 1977, an average of less than .25 reports per licensee per year. This data density is clearly inadequate for performance analysis.

LER data for these three licensee classes are practically nonexistent. In 1976 and 1977, a total of 137 events were entered into the LER file for all these classes combined. Most of these LERs are for the 93 test and research reactors and the 38 fuel facilities, producing an average of .5 report per year for those 131 licensees. The data density is inadequate for performance evaluation for these classes of licensees.

To summarize, we believe it is possible to perform meaningful two-dimensional performance analyses (using LER file and 766 file data) only for operating power reactors at this time. Only the single dimension of 766 data is adequate to analyze performance for reactors under construction and for fuel facilities. However, due to data limitations as discussed in Section 3.3.3.2, this one-dimensional analysis will not provide a comprehensive evaluation of licensee performance nor the necessary insights into the reasons for that performance.

Availability of Program Resources for Performance Evaluation

The FPM model and methodology permits the performance of licensees to be analyzed individually. For certain classes of licensees, particularly the operating power reactors, sufficient data makes individual analysis possible. However, there are classes of licensees - materials licensees, for example - for which the existing data are scant and manpower and time to gather more data may not be available. But we believe that the performance of these classes can be analyzed in the aggregate through inspection of a "stratified" statistical sample of the class. As used in this context, statistical sampling is similar in principle but differs from previous uses NRC has made of this technique. NRC has in the past considered statistical sampling to determine the number of items to be inspected for each licensee, as in the Statistical Sampling Inspection Program discussed in Section 2.5.1 of this report. We propose to use statistical sampling techniques to determine the total number of licensees upon which inspection resources would temporarily be focused.

A performance profile can be established for each licensee in the statistically selected sample group and licensees with similar profiles within sample group can be identified. The result will be a statistically selected sample of licensees that can be grouped on the basis of similarity in performance profiles. This method will permit NRC to make statistically valid statements that characterize:

- The performance of a licensee class in terms of what percentage is represented by each profile--the establishment of "class performance groups."
- The risk presented by a class of licensees on the basis of the "class performance groups."

This type of analysis will permit the NRC to focus its resources on those sub-classes of licensees that require further attention. It will also permit the NRC to evaluate the type and amount of additional regulatory attention it should devote to a particular class of licensee.

4.0 RESPONSE TO REQUIREMENTS OF THE NRC REQUEST FOR PROPOSAL

4.1 SUPPORT FOR NRC'S MISSION AND GOALS

NRC must continually ask whether its actions effectively support its mission to protect the public health and safety, to safeguard nuclear materials, and to maintain environmental quality. This question is especially important in a program that may be somewhat controversial. We believe that licensee performance analysis fully supports NRC's mission and goals for several reasons:

- Licensee performance analysis can be used as a tool for effectively allocating inspection resources. If increased attention to a licensee can help him improve his operational safety, then that improvement directly supports NRC's mission.
- Our study to date indicates that licensees whose performance patterns display sequences of causally linked events either at the system level or in aggregate are more likely to experience future significant events than those whose patterns suggest more effective managerial control. This inference could prove helpful to NRC through alerting the agency to the need for appropriate action.
- NRC must have an effective enforcement program, and the performance profiles can be used to establish a context for determining the severity of sanctions when noncompliance occurs.
- A properly structured performance analysis tool can improve relationships between NRC and the licensees by more clearly defining a level of acceptable performance. A poor relationship between NRC and the licensees affects the ability of both parties to protect public health and safety in an efficient and effective way.

We are also convinced that licensee performance analysis offers insight into the safety differences among licensees. Mechanical safety of a plant is the result of the licensing process, and to the extent that the licensing process does its job, all plants should meet minimum safety requirements when an operating license is issued. After a plant begins operating, safety is much more a function of the management and personnel than of equipment. Licensee performance analysis is capable of revealing management and personnel response to the "signals" provided by the plant through checking for chains of causally linked events. A small number of related events show that management and personnel can accurately pinpoint problems and solve them; many related events indicate the failure to react adequately.

The question of publishing the results of licensee performance analysis should not assume a major weight, but it must be considered. There is little doubt that the results of the analyses will be published in some form, simply because NRC has an obligation to report to the public. The existence of the Freedom of Information Act guarantees that the obligation will be met. The real issue is the form in which the analyses will be released; the potential public use or misuse may influence that form. The information released should be factual rather than inferential; one possible format is an annual "rainbow book" presenting profiles for each licensee together with other information such as inspection hours and numbers of noncompliances.

4.2 MEETING THE NRC'S "EVALUATION CONSIDERATIONS"

In its Request for Proposal, NRC identified several "Evaluation Considerations" against which the developed evaluation methodology was to be tested. Each of these considerations is presented below together with responses based on the FPM model and methodology.

- The relationship between the evaluation criteria and safety. Each measure of licensee performance selected, including compliance with NRC requirements, must be strongly related to NRC's mission of insuring safety.

  Response: LERs are indicators of "out-of-bounds" operation; thus analysis of their content can provide insight into potential safety problems.

- NRC's regulatory authority. Those evaluation methods proposed for near-term application must be consistent with NRC's existing regulatory authority. For example, it may not be appropriate to evaluate licensees on the basis of commercial productivity factors, unless it can be demonstrated that those factors relate to NRC requirements or to the safety of a licensed operation. Areas where NRC's regulatory authority should be expanded will be identified.

Response: The FPM methodology can be fully applied within the current scope of NRC's regulatory authority.

- Analytical depth. For any class of licensees, the appropriate level of analytical depth permits identification of actual differences in licensee performance. While these insights may derive from a relatively simple, aggregated analysis of summary data, it may be judged necessary to evaluate performance on the basis of in-depth examinations of specific events, incidents or occurrences.

  Response: To effectively use the model, the content of data must be analyzed in appreciable depth. For example, licensee performance patterns based on LERs are derived from the contents of these reports, which must be carefully analyzed and evaluated if the end results are to be meaningful and useful.

- Quantitative versus qualitative evaluation. Both types of measures must be considered. Quantitative evaluations are based upon measurable indicators such as numbers of items of noncompliance. Qualitative judgments involve subjective ratings by Regional Directors or other similar measures.

  Response: The FPM methodology is not quantitative in the sense implied above because its purpose is to achieve both a temporal assessment of the licensee's performance patterns and an insight into the reasons for the shapes of these patterns.

- Data considerations. In quantitative evaluations, the lack of suitable data may limit the ability to evaluate licensees. Evaluation methods must be based on data currently available or upon data that is obtainable with reasonable effort. The contractor will identify data that should be made available and suggest appropriate methods for its collection.

  Response: The case studies included in this report were all based on currently available computerized data. The subject of what data would be most useful for application in the methodology described, including the question of appropriate collection means, is complex and is discussed in Section 6.0 of this report.

• Licensee control over rating factors. To be fair, licensees must be evaluated on the basis of factors that they can directly influence.

Response: This is a valid consideration, and the FPM methodology ignores factors not within the control of the licensee.

• Uniform application. The population of NRC licensees will be partitioned into homogeneous groups for the purpose of evaluating their performance. Evaluation methods will not discriminate against particular licensees in any given group.

Response: This methodology is applied uniformly to the members of a given licensee class and in a form appropriate to that class.

• Categories of evaluations. Two distinct aspects of licensee performance must be captured in the evaluation methodology--overall performance and performance in specific areas of responsibility.

Response: In the case studies, we developed performance patterns for overall licensee performance and for performance as reflected by event histories of specific facility systems. The FPM methodology is inherently suitable for evaluating performance in various areas of responsibility, provided that appropriate data can be made available.

• Relative versus absolute performance. The evaluations will consider a licensee's performance both in comparison to that of other similar licensees and as measured against reasonable absolute standards of acceptability.

Response: The FPM methodology permits evaluations of both types. It is not, however, designed for the ranking of licensees on a numerical scale.

• Weighting. If licensee performance evaluations are to be based upon several independent factors, the relative importance of these factors must be reflected in the weights assigned to each. Also, the sensitivity of evaluation results to various choices of weights will be investigated.

Response: This methodology does not combine diverse performance indicators within a single end measure, but instead portrays licensee performance as a pattern over time. The question of factor weighting is not relevant to the FPM methodology.

5.0 PLAN OF ACTION FOR PHASE II

This section presents our proposed plan of work for Phase II of this study. Originally, Phase I was to be a feasibility study, and a methodology was to be developed in Task II and applied to Task III, both components of Phase II. Our Phase I work meets all the requirements of Phase I and Task II. We developed the FPM model and found that it applies to all classes of licensees in principle (see Section 3.4.1) but that currently available data are insufficient to permit meaningful performance analysis of licensees other than operating power reactors (see Section 3.4.2). Potential solutions to this problem are discussed in Section 6.1 as "Work Area 3." We have also begun to meet Task III, by applying the analysis methodology to three operating power reactors to test its worth and sensitivity to performance differences; it appears capable of producing performance patterns that not only distinguish "good" from "poor" performers but that illustrate the reasons for those distinctions.

5.1 PHASE II WORK PLAN

In Phase II, we plan to continue to test and refine the FPM model by conducting licensee performance analyses of seventeen additional power reactors. I&E management has already identified seven licensees in this group:

1) Trojan
2) San Onofre 1
3) H.B. Robinson
4) Indian Point 2
5) Oconee 1
6) Browns Ferry 1
7) Arkansas 1

In the original RFP statement of work, NRC proposed analyzing "twenty reportable events that had potential safety significance...." The Phase I work has demonstrated that a far more complete and searching study of licensee performance is essential if an event or combination of events is to be viewed in a meaningful perspective. It is for this reason that the remaining case study effort will be expanded considerably beyond the scope that NRC had originally envisaged.

The seventeen case studies to be performed, together with those originally conducted, will include operating power reactor licensees selected from all five NRC regions. The Phase II analyses will follow the methodology described in this report and, to provide consistent data, will cover the same two-year period of 1976 and 1977. Reviewing and interpreting the analyses from this larger population should expand the insight into the causal mechanisms explaining licensee behavior, and will help determine the effect (if any) of different reporting requirements and technical specifications. Comparing and analyzing a large number of licensee performance profiles may reveal indicators of the probability of future event occurrences.

The complete description of work performed in Phase II, together with analyses and interpretations of the case study findings, will be provided in the Phase II report. This report will deal primarily with specific licensee analyses rather than general methodological considerations.

5.2 PHASE II REVISED ESTIMATE OF EFFORT

Even though, as explained above, the Phase II work effort to be performed exceeds that originally envisioned, we believe that the work can probably be accomplished within the remaining contract resources. However, we have found that the resources required per case study vary considerably. As an example, we analyzed roughly five times as many LERs for Zion Unit 1 as we did for Point Beach Unit 1. Obviously, these factors make it difficult to predict the aggregate Phase II level of effort with precision.

If the magnitude of the total available data for the remaining seventeen cases should prove to be quite large, we would seriously consider its reduction and analysis by computer, as was discussed in our original technical proposal. It is expected that a judgment will be made early during the Phase II work period regarding the benefits and costs of this approach. The Project Officer will be immediately informed of this judgment and its implications for project resources to allow him to come to a prompt decision.

6.0 RECOMMENDATIONS FOR CONSIDERATION BY THE NRC

This section sets out several work areas identified during the Phase I effort which, although outside of the current scope of work, should be given consideration by the NRC. These areas relate either directly to licensee performance analysis or to NRC's inspection program. In virtually all cases these areas could not have been precisely defined prior to the performance of the work described in this report. Largely as a consequence of studies to date and to some degree as a result of discussion of our preliminary results with I&E staff personnel, it is clear that the recommendations summarized below address agency needs that are coming into sharper focus:

The recommended study areas fall into two categories:

- Direct extensions of the current effort: These work areas address necessary refinements and expansions of the licensee performance analysis methodology already developed. They also include applying this methodology to earlier phases of power reactor operating history than have been considered to date.
- Supplements to the current effort. In a strict sense, these topical areas fall outside of licensee performance analysis as a methodology, since they relate to the formal structure and the practical implementation of the NRC inspection process.

6.1 DIRECT EXTENSIONS OF THE CURRENT EFFORT

Work Area 1. Data Quality Improvement for Licensee Performance Analysis

In Sections 3.3.2 and 3.3.3, we identified the inadequacies in the currently existing computerized data for operating power reactors and discussed how we adapted those data for use in the FPM model. To use the FPM model to its fullest capacity, it is essential that these data be made available to the licensee performance analyst in a form that permits the analyst to draw complete and accurate inferences about the information within the FPM model arrows and the actions within the FPM circles. It is equally important that data accuracy and completeness be well standarized among the NRC regions. Current criteria for LER reporting must be carefully reviewed to make these criteria more specific, particularly with respect to the "grey area." At present, it is not uncommon for a licensee to seek guidance from I&E as to whether an LER is required, especially when the severity of the event in question is marginal.

We recommend that both the LER and 766 reporting formats and requirements be modified to provide more directly useful information for licensee performance analysis. Appropriate codification of this restructured data will permit licensee performance patterns to be generated by computer.

Work Area 2. Automation of Licensee Performance Analysis for Operating Power Reactors

In Section 3.5, we discussed the relationship of the FPM methodology to the PAT program and suggested the possibility of applying that methodology through interactive computer capability. This would permit "real time" performance profiles to be continuously and automatically produced and updated through links with the 766 and LER data systems.

Work area 1 was concerned with achieving a data form compatible with the FPM model and amenable to automated processing. Once these necessary steps have been taken, appropriate software for licensee performance pattern generation and interpretation can be developed.

Automation of licensee performance analysis will serve I&E interests in two key respects:

- It will relieve scarce personnel resources of the burden of generating and interpreting performance profiles by "hand and eye."
- Uniformity of pattern interpretation will be enhanced by excluding variable human judgment.

This work area cannot be implemented until or near the conclusion of the previous work area effort.

Work Area 3. The Data Availability Problem (Licensees other than Operating Reactors)

As explained in Section 3.6.1, the FPM model is general in concept and is inherently applicable to all classes of NRC licensees. The difficulty in applying the model to classes of licensees other than operating power reactors lies in the paucity of reliable data. For example (see Section 3.6.2), in the two-year period of 1976 and 1977, there are only 137 LERs in the NRC computer file for the 93 test and research reactors, 38 fuel cycle facilities, and the more than 9,600 materials licensees. For this same period there are 247 inspection reports for 93 test and research reactors, 995 inspection reports for the 38 fuel cycle facilities and 4,737 inspection reports for the more than 9,600 materials licensees. While the density of 766 data is acceptable for most licensees (except for materials licensees and test and research reactors), the LER data for licensees other than operating reactors is not adequate to permit meaningful performance analysis.

We believe that the density of LERs could be increased if reporting requirement were specifically tailored to reflect the performance-sensitive characteristics of each licensee class. At present, it does not appear likely that the density of inspection reports for those licensee classes in which it is now low can be materially increased because of I&E personnel resource limitations. It is possible, however, that applying appropriate population sampling techniques can appreciably augment the inspection information density for the sample. This will permit valid statistical inferences about the different licensee classes.

Work Area 4. Performance Profiles of Immature Operating Power Reactor Licensees

It is a matter of common knowledge that personnel performance tends to improve as a new task is gradually assimilated and mastered. The rate of improvement in new task performance with time can be graphically shown as a "learning curve." By analogy, in the case of operating power reactors, the significant event occurrence rate usually decreases as the operating history lengthens. This consideration was taken into account in the case studies presented in Appendix A, since each plant had been operating more than two years prior to the period of analysis. To the degree that it is reasonable to assume that the level of risk presented by newer plants (particularly during the startup phase) is greater than the level associated with more mature facilities, there could be a real advantage in developing and analyzing licensee "learning curves" based on data from the first three years of operation. It is quite possible that analysis may reveal performance patterns that are characteristic of early but not late periods of facility operation. These patterns can be extremely valuable to I&E in its effort to reduce the risk associated with facility operation during its immature period. We recommend that the early performance of about 10 plants be studied. These plants should have commenced operation not earlier than 1973, because the LER system was activated in that year.

6.2 SUPPLEMENTS TO THE CURRENT EFFORT

Work Area 5. Realignment of the Inspection Process

In Section 3.3.3.2, we discussed why the modularized inspection program does not lend itself to revealing the reasons for performance. The module under inspection is the "point of origin" of noncompliances, a point often well-removed from the actual event occurrence. The scheduling of the modules makes the noncompliance data in the 766 file reflect a time-dependence that is not inherent in the events, but in the program. Testing the 766 data against the model pointed out how certain aspects of the current inspection methods could be modified in a manner most beneficial to licensee performance analysis.

At this point, we believe it is important to distinguish between an inspection process in principle and the particular form in which that principle is implemented. For this reason, we propose to consider elments of program form that govern the output of the inspection process. At least five of these elements are: 1) time-dependence; 2) program area; 3) inspection unit (system or module); 4) use of inspector cues (reactivity); and 5) inspectable performance indicators other than noncompliances. These elements are highly interdependent. Therefore, we propose to develop experimental inspection programs with various "mixes" of the formal elements and to test the output of these experimental programs against the FPM model. The key objective of this redirection would be to enhance the informational value of the inspection process in the context of performance analysis; this redirection is in no way intended or designed to impair the critical role of the inspection process as a basis of safety assurance.

Work Area 6. I&E Regional Performance Analysis

NRC headquarters personnel, regional personnel, and the licensees all state that the regions vary in their management of inspection resources and in their general management approach. Whether the regional inspection program operations reflect the relative qualities of "good" and "poor" performing licensees is unknown. Regional program variations of this type can in part be observed through examining the temporal phase differences seen between the LER- and noncompliance-derived performance profiles. We believe that an inter-regional analysis whose objectives include the identification of correlations between phase lag magnitudes and noncompliance inspection yields will provide a useful tool for understanding differences among the regional operating philosophies (the relative preferences of regional directors for high non-compliance yields vs. short lags).

We do not suggest that the regional performance indicators mentioned above include all those most appropriate for identifying and assessing inter-regional differences. As in the case of licensee performance analysis, it will be necessary to construct an insightful model of I&E regional structure and operation (RSO model). Once this has been accomplished, the model will directly guide us to those parameters that are both meaningful and sensitive.

APPENDIX A
CASE STUDIES

INTRODUCTION

From the outset of this project it was clear that case studies were necessary to empirically test the validity of the chosen approach. Since our approach was to develop a comprehensive model and procedure applicable to all classes of licensees, we chose to perform case studies of operating power reactors to test the FPM model, methodology, and performance indicators against the most complex of NRC's licensees. Further, the data available for operating power reactors are the most complete.

Selecting the Case Studies

To eliminate any possible regional effects that could diminish the meaningful comparison of one case study with another, we performed all the case studies in one NRC region. To prevent the possible bias of cross-NSSS vendor comparison, we searched for facilities using the same equipment. Third, based on discussions with NRC personnel, we felt that any facility must have been operating for more than two years, to prevent a "learning curve" effect from destroying meaningful comparison and possibly obscuring the patterns or indicators that might otherwise be evident in a mature facility.

Finally, we decided to study at least two facilities, one perceived by NRC as a "weak" performer and the other as a "good" performer. This provided us with the opportunity to empirically identify patterns and indicators related to each performance category ("poor" and "good"). It also offered the chance to gain insight into underlying causal factors associated with the dichotomy of performance.

For these reasons, we selected Zion Unit 1 and Point Beach Unit 1. Both are in Region 3, both are Westinghouse plants, and last, both had more than two years of operating experience by the beginning of 1976. When we discussed our choices with Region 3 management, it was mentioned that the differences in technical specifications and reporting requirements between Zion Unit 1 and Point Beach Unit 1 were considerable. Region 3 felt that we should consider studying a third performer with reporting requirements and technical specifications similar to Zion Unit 1, and suggested Prairie Island Unit 1. Consequently, we studied three licensees--Zion Unit 1, Point Beach Unit 1, and Prairie Island Unit 1. This gave us the additional opportunity to begin to examine the impact of differences in reporting requirements and technical specifications on the FPM model and methodology.

Performing the Case Studies

We performed the case studies in accordance with the FPM model and methodology discussed in Section 3.2, and we analyzed the LER file data and the 766 file data as described in Section 3.3.2.2 and 3.3.3.2 of this report. The study period covered calendar years 1976 and 1977, in order to produce profiles extending over a sufficient length of time to allow potential changes in performance to be seen and assessed. In any ongoing performance analysis, the study period should obviously be current, and each of these three case studies can be readily updated.

Presenting the Case Studies

Each case study is presented in two separate parts that reflect the two different data dimensions--LER data and 766 data--used in the study. This allows the reader to gain an appreciation of the types of insights each data source provides as well as an appreciation of the sensitivity of each source to specific aspects of licensee performance. Performance profiles and supporting data sheets help the reader gain insight into the foundations of the case study effort as well as an appreciation of the study details.

PRAIRIE ISLAND UNIT 1 CASE STUDY

Review of the LER File for Prairie Island Unit 1

During 1976 and 1977, events occurred in 22 systems as shown in Table A-1 on page A-11. The Circulating Water System sustained an extraordinarily large number of events in comparison to the other 21 systems. These 21 systems averaged 3.0 events over the 24-month period. Four of these 21 systems had an average of 7.25 events per system; removing these systems from the group of 21 resulted in an average of 2.0 events in 24 months for the remaining 17 systems. A detailed review of these 17 systems revealed two systems (one with three events; the other with four events) in which causally linked events were related to failures in human performance.

Circulating Water System

In 24 months, 41 events occurred in this system. The licensee attributed three of these events to component failure and the remainder to cause code "other." We upgraded two of the events designated by the licensee as component failure to Teknekron Event Responsiblity Code M (ERC-M); we upgraded 26 of the 38 events classified as "other" to ERC-M.

For 20 months, this system was unable to meet the environmental technical specifications for tower blowdown. A large number of our reclassifications were prompted by equipment design temperature requirements that could be met only by increased blowdown rates, a factor we considered due to faulty design. Our remaining reclassifications were made on the basis of apparently high velocities in the intake structure, which result in fish impingement outside of technical specifications, which we also consider faulty design. We consider virtually all of these 26 events to be causally linked. However, the number and frequency of the events, as well as the way they were reported in the LERs, indicates that management was aware of the basic cause. By 8/04/76 plant engineers were studying alternative designs. It was also evident that a conscious decision had been made by the facility management to continue to operate the facility while redesigning the circulating water system because the system does not affect operating safety.

Ultimate Heat Sink Facilities

Eight event reports were associated with the operation of this system. The results of our review produced a reclassification of five events from a licensee-identified cause code of "other" to ERC-M. Four of these were causally linked because flow rates in excess of the environmental technical specifications were required to maintain system design temperature conditions for a period of two months. This points to system design inadequacy, in which case the plant management should have redesigned the system or changed the technical specifications. But these causally linked events occurred only for a two month period of 1976 and did not occur thereafter, probably indicating corrective management action.

Containment Heat Removal System

The profile for this system is shown in Figure A-1. This system had nine events in 24 months, and we noted two groups of causally linked events. The first group involved three events spanning a 19-month period. The date on which they occurred, together with the Event Responsibility codes assigned by the licensee and by Teknekron, are:

Date (licensee code/ERC)
1-21-76(F)*
7-01-77(F/M)
7-26-77(F/M)

During a containment inspection on 1-21-76, the dome discharge damper for the No. 14 fan coil unit was found to be improperly positioned. The licensee stated the cause and its response as "binding of the actuator shaft in its bushing. All actuators will be disassembled and inspected at the upcoming refueling

*If no change in code occurs, only the licensee cause code is given.

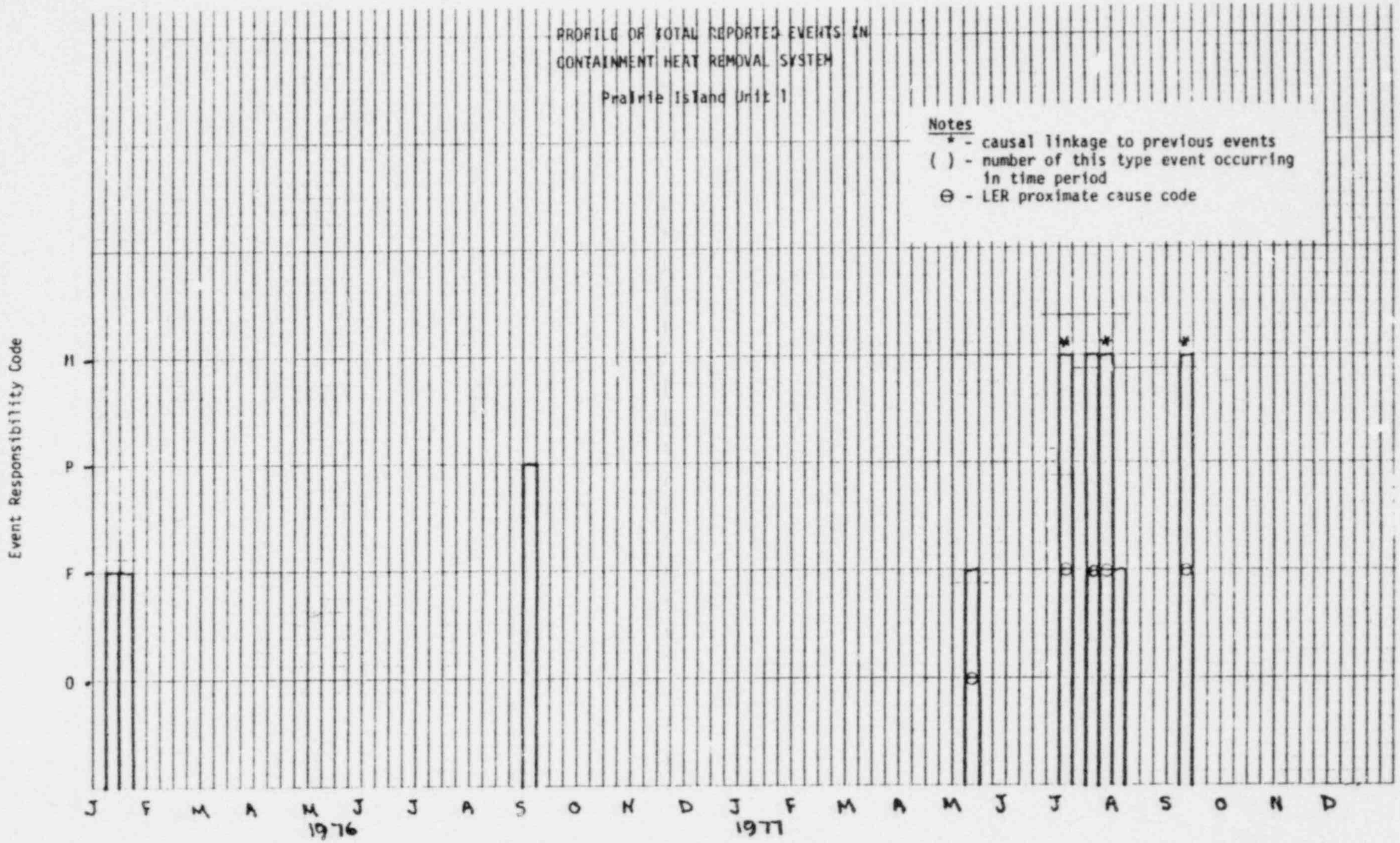

PROFILE OF TOTAL REPORTED EVENTS IN
CONTAINMENT HEAT REMOVAL SYSTEM
Prairie Island Unit 1
Notes
* - causal linkage to previous events
( ) - number of this type event occurring in time period
Θ - LER proximate cause code
Event Responsibility Code
M
P
F
O
J F M A M J J A S O N D J F M A M J J A S O N D
1976
1977

FIGURE A-1

outage. Airline lubricators will also be installed at that time." On 7-01-77 the No. 12 fan coil unit dome damper failed to operate. The licensee stated cause and response was "sheared pins in the damper-to-actuator shaft couplings. Pins were replaced. Pacific Air Products damper with Ramcon No. R-260 actuators." On 7-26-77, during containment inspection, the No. 12 fan coil unit dome damper was found partially closed. The damper was immediately clamped in full open position. The licensee stated that "actuator failure" was the cause of the event. Both actuators were replaced. The equipment involved was a Pacific Air Product damper with Ramcon No. R-260 actuators.

In summary, there appears to be a causal link existing between the 1-21-76 and the 7-26-77 events, since the two failures occurred in similar equipment in redundant systems. This may indicate an incomplete identification of the cause of the 1-21-76 event, an incomplete application of the prescribed remedies to the 1-21-76 event, or possibly just a random subsequent failure. The failure of the actuator-to-damper pins in the 7-01-77 event indicates that the identified causes and/or the remedies prescribed for the 1-21-76 event may not have been adequate. However, the lack of subsequent events in the LER file for the period of record very likely indicates that management and personnel had identified and implemented generic remedies to prevent this type of event.

The second group of causally linked events occurred on 7-27-77 (F) and 9-14-77 (F/M). These events were identical in that the cause of both events was a failure of control fuses and both events occurred in redundant systems (No. 13 and No. 14 fan coil units). The lack of subsequent events in the LER file indicates that management and personnel had probably identified and implemented generic remedies to prevent recurrence.

Reactor Containment System

As the system profile in Figure A-2 shows, events on 5-04-76 and 10-23-76(P) clearly are the result of isolated personnel error. But the event of 8-25-76 (P/M)

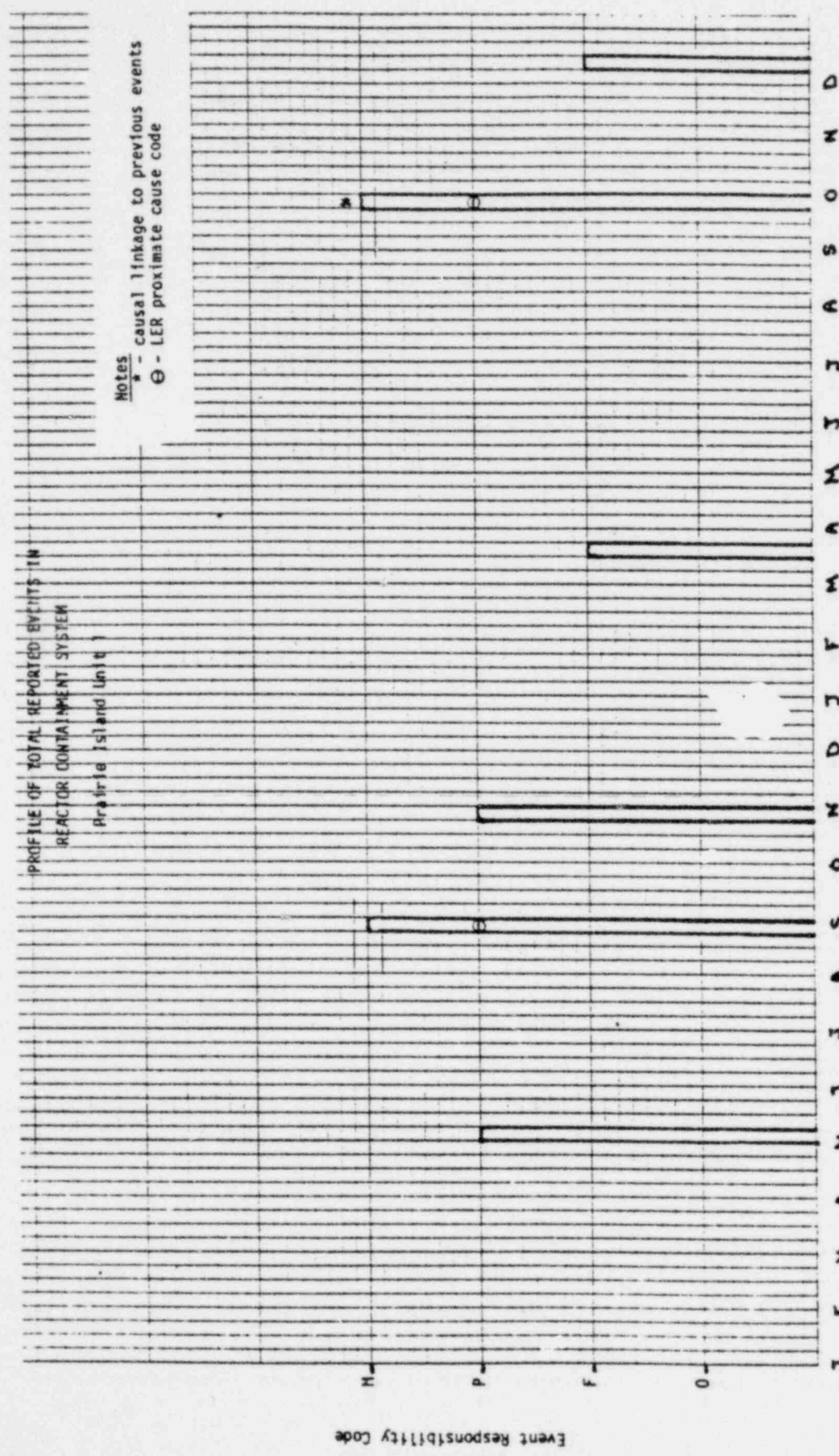

PROFILE OF TOTAL REPORTED EVENTS IN
REACTOR CONTAINMENT SYSTEM
Prairie Island Unit 1
Notes
* - causal linkage to previous events
Θ - LER proximate cause code
N
P
F
O
Event Responsibility Code
J F M A M J J A S O N D J F M A M J J A S O N D
1976
1977

FIGURE A-2

and 9-29-77 (P/M) appear to be causally linked through apparent management failure to develop and implement administrative controls for the auxiliary building special ventilation zone. In the report of the 8-25-76 event, the licensee identified lack of administrative control as being partly responsible for the event. The 9-29-77 event seems to have resulted from a less-than-complete implementation of the administrative controls. This event group demonstrated that the facility management was aware of the need for generic event cause identification and remedy application. It is also a positive demonstration of how the facility management performs its role in responding to events.

Station Service Water System

The licensee coded the event on 2-25-77 as component failure; the event of 5-20-77 was coded as sluggishness of the diesel water cooling pump governor. The 5-20-77 event also was associated with a sluggish governor. At this point, management began surveillance testing of governor response. There were no subsequent events, indicating effective management response.

On Site Power System

All three events in this system are causally linked. In the events of 6-15-76(F) and 11-21-76(F), the cause and specific system point of occurrence are identical. The cause of the event on 3-14-77 is identical to the previous two, but it occurred in a redundant system. The fact that another event with the same cause has not occurred in the period of record indicates effective management action.

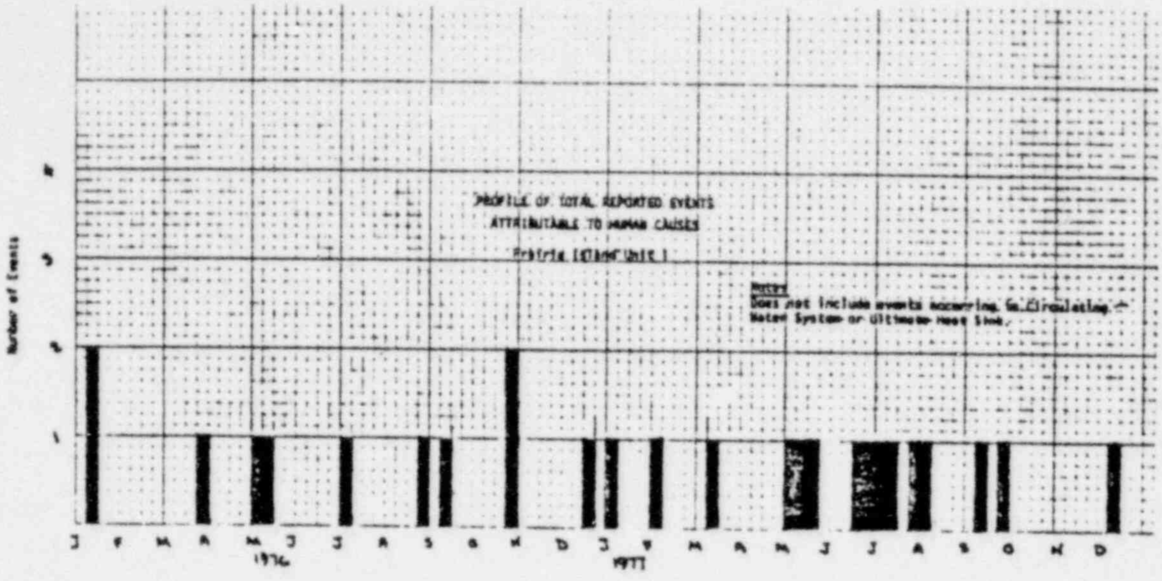
System Code Not Applicable

Point Beach Unit 1 used this "catch all" category to collect occurrences related to technical specification violations by personnel and to record management oversights and communication breakdowns among personnel. The six events in this system ranged from a licensed operator's misunderstanding of the requirements for reactor core axial offset control to a failure to perform a required test because personnel were absent.

Summary

The analysis of the LER event reports for this licensee indicated design problems in the Ultimate Heat Sink Facility and the Circulating Water System. It appears that design changes in the Ultimate Heat Sink Facility must have been made around 10-76, since there are no event reports on file for this system after this date. It is also possible but we do not think likely that the licensee ceased to report events resulting from the operation of this system after 10-76. A review of other system files of which patterns could be identified (Containment Heat Removal system, Reactor Containment Systems, Station Service Water System, On Site Power System, and System Code Not Applicable) indicated management attention to repeated component failures and personnel errors. In the systems where causal relationships did appear, the facility management's responsiveness was such that no more than three events occurred before an apparent resolution was found and event reports ceased to appear. On the basis of the LER "Event Description" and "Cause Description" provided by the licensee, the facility management approach to resolution of events was to analyze each event for its generic impact on the plant and resolve the event accordingly. This undoubtedly resulted in the low repeatability of events and demonstrates ongoing management awareness of and attention to unscheduled occurrences, particularly in those areas which can be identified as safety-related.

The two profiles in Figure A-3 show the overall facility pattern of the cause of events. The top profile shows human error (management and personnel) as a function of time. Human error for this facility appears to uniformly distributed, indicating a well-managed facility operating in the "noise" band of event data. The bottom display shows component failure as a function of



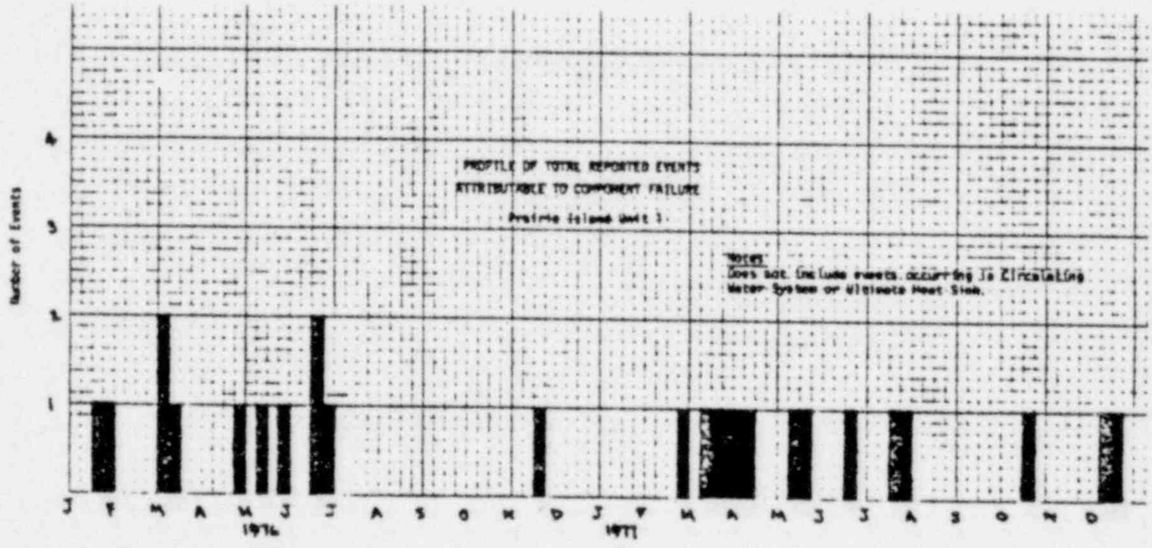


FIGURE A-3

Prairie Island Unit 1 Performance Profiles

Table A-1

LERs BY SYSTEM AT PRAIRIE ISLAND UNIT 1 - 1976 and 1977

Emergency Generating System	Containment Combustion Gas Control System	Containment Heat Removal System	Circulating Water System	Ultimate Heat Sink Facilities	Reactor Trip System	Emergency Core Cooling System
1-09-76(O)	1-13-76(P)	1-15-76(F)	1-21-76(O)	1-28-76(F)	2-08-76(O)	3-01-76(F)
12-09-77(P/M)		1-21-76(F)	2-11-76(O)	2-03-76(O)	6-17-76(F)	5-13-76(F)
		9-10-76(P)	3-04-76(F)	5-13-76(O/F)	1-07-77(M)	4-11-77(F)
		5-13-77(O/F)	6-16-76(O)	6-01-76(O/M)	10-15-77(F)	
		7-01-77(F/M) (6)	6-30-76(O)	6-06-76(O/M)		
		7-21-77(F/M)	7-07-76(O)	7-01-76(O/M)		
		7-26-77(F/M) (8)	7-14-76(O)	8-01-76(O/M)		
		7-27-77(F)	7-21-76(O/M)	10-01-76(O/M)		
		9-14-77(F/M) (10)	7-28-76(O/M)			
			8-04-76(O/M)			
			8-11-76(O/M)			
			8-18-76(O/M)			
			8-25-76(O/M)			
			9-01-76(O/M)			
			9-01-76(O/M)			
			9-08-76(O/M)			
			9-15-76(O/M)			
			9-22-76(O/M)			
			9-29-76(O/M)			
			10-06-76(O/M)			
			10-13-76(O/M)			
			10-20-76(O)			
			10-27-76(O)			
			11-03-76(O)			
			11-10-76(O)			
			11-17-76(O)			
			11-24-76(O)			
			12-01-76(O/M)			
			12-08-76(O/M)			
			12-15-76(O/M)			
			12-22-76(O/M)			
			1-03-77(O/M)			
			1-26-77(O/M)			
			2-14-77(F/M)			
			2-23-77(O/M)			
			3-08-77(F/M)			
			4-18-77(O/M)			
			5-12-77(O/M)			
			6-30-77(O/M)			
			7-31-77(O/M)			
			8-31-77(O/M)			

Table A-1 (cont.)

LERs BY SYSTEM AT PRAIRIE ISLAND UNIT 1 - 1976 and 1977

Other Engineered Safety Feature Systems Instrumentation	Engineered Safety Feature System Instrumentation	Airborne Radioactive Monitoring System Instrumentation	Reactor Containment System	Station Service Water System
3-01-76(F)	3-08-76(O/F)	4-24-76(F)	9-04-76(P)	5-11-76(P)
3-28-76(M)	4-07-77(F)	5-24-76(F)	8-25-76(P/M)	6-29-76(F)
3-08-77(F)			10-23-76(P)	2-25-77(F)
			3-16-77(F)	5-20-77(O/F)[2]
			9-29-77(P/M)[9,4,11]	
			12-09-77(F)	

Air Conditioning, Heating Cooling, Ventilation System	On Site Power System	Chemical Volume Control System (Chlorine Addition to Cir. Water System)	Spent Fuel Storage Facilities	Containment Isolation System
5-18-76(M)	6-15-76(F)	7-01-76(M)	10-24-76(P)	3-24-77(F)
	11-21-76(F)[2]	2-03-77(P/M)		
	3-14-77(F/M)[3]			

Table A-1 (cont.)

LERs BY SYSTEM AT PRAIRIE ISLAND UNIT 1 - 1976 and 1977

Feedwater System	Systems Code Not Applicable	AC Onsite Power System	Chemical, Volume Control, & Liquid Poision System	Reactor Coolant System
3-01-77(M)	1-11-76(P)[4]	6-17-77(F/P)	6-28-77(F/P)	12-20-77(O/F)
6-18-77(F)	8-05-76(O)[5]			
	12-21-76(P/M)			
	5-08-77(P/M)[4]			
	7-14-77(P/M)[7]			
	8-05-77(P/M)[9]			

NOTES: FOR TABLE A-1

1. This event was not assigned to a system in the LER. The category selected for this event by Teknekron was due to the continued necessity for high blowdown rates which identified it as the circulating water system.

2. This event is an identical repeat of the previous event in terms of equipment type and cause of failure - suggests a possible design deficiency

3. This event appears to be a repeat of the previous event in terms of equipment type and cause of failure - management should be reviewing this as a design deficiency.

4. Violation of technical specifications.

5. Vendor error in accident analysis assumptions.

6. Appears to be identical to previous event 1-21-76 which required equipment to be disassembled and lubricated - now the pins are sheared (perhaps lack of lubrication?).

7. Similar to 12-21-76 event - appears to be failure of management oversite in scheduling of personnel.

8. Similar to previous event 7-01-77 and 1-21-76.

9. Communications breakdown among personnel and management.

10. Similar to previous event on 7-27-77 in a redundant system.

11. Similar to previous event on 8-25-76.

time. This display indicates a certain periodicity with a fairly uniform distribution at periodic intervals. Since most component failures were identified during routine surveillance testing, the apparent periodicity may be associated with the surveillance test frequency and mode of facility operation.

Review of 766 System Data File and Inspection Reports for Prairie Island Unit 1

When we reviewed the 766 system data file and associated inspection reports for 1976 and 1977, we found a total of 48 inspection reports detailing the results of NRC I&E inspector findings. Sixteen of these report identify a total of 29 items of noncompliance. Eleven of these 29 items involve physical protection and are identified in three separate inspection reports.

Matrix A-1 summarizes the findings of each of the 16 inspection reports and associated 766 system data file entries that identify noncompliances. Not including those noncompliances due to physical protection, nine noncompliances were assignable to ERC-M, and nine to ERC-P.

In general, the noncompliance cause code as listed in the 766 system and the detailed discussion in the "Report Details" section of the inspection report agreed reasonably well. Less than 20 percent of the noncompliance cause codes either were ambiguous or did not agree with the associated inspection report details. There was generally strong agreement between the enforcement text provided for each item of noncompliance identified in the 766 system and the "Enforcement Actions" section of the associated inspection report. There was less agreement between the noncompliance cause code in the 766 system and the 766 enforcement text: approximately 37 percent of the items bore either an ambiguous or irrelevant relationship to each other. The ambiguity was partly due to a lack of supporting detail in the 766 enforcement text, and also reflects the nearly 20 percent ambiguity found in the relationship of the 766 system cause codes to the inspection report. This substantial ambiguity between the noncompliance cause code and the 766 enforcement text for

MATRIX A-1

Review of 766 File and Inspection Reports for Prairie Island

NAME PRAIRIE ISLAND UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
76-02	FJP3	M	YES	YES	YES	YES	NO	YES	NONE	1 EVENT/AGREE
	ASE2	M	NO	CAN'T TELL	YES	NO	NO	NO	NONE	
76-03	FPG2	P	YES	YES	YES	NO	NO	YES	NOT INSPECTED	
76-08 (Phy. Prot.)	RLC2	P	NO	CAN'T TELL	YES	NO	NO	IN SUBSEQUENT LETTER		
	RMC2	P	YES	YES	YES	NO	NO	YES		
	RLC2	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
76-09	FJL3	M	YES	YES	YES	YES	YES	IN SUBSEQUENT LETTER	NONE	2 EVENTS/AGREE
	FDB2	M	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
76-11	FJE3	M	YES	YES	YES	NO	NO	NO	YES (3 ITEMS)	9 EVENTS/AGREE
76-13	FJG2	M	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER	YES (1 ITEM)	5 EVENTS/AGREE
76-15	JAY3	P	YES	YES	NO	NO	NO	IN SUBSEQUENT LETTER	NOT INSPECTED	
76-16	FCG2	P	YES	CAN'T TELL	YES	NO	NO	YES	YES (1 ITEM)	2 EVENTS/AGREE
76-18 (Phys. Prot.)	RMB2	M	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER	YES (3 ITEMS)	
76-19	FCG2	P	YES	NO	YES	YES	NO	YES	YES (3 ITEMS)	2 EVENTS/AGREE
77-02	FCS2	P	CAN'T TELL	CAN'T TELL	YES	NO	YES	NO	YES (2 ITEMS)	

NAME PRAIRIE ISLAND UNIT 1

Insp. Rpt.	Non Comp.	Teknek-ron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
77-02	FEP3	P	YES	YES	YES	NO	NO	YES		
77-07	FPF2	P	YES	YES	YES	NO	NO	NO		
77-11	FJF2	P	YES	CAN'T TELL	YES	NO	YES	YES		3 ITEMS/AGREE
77-18	FDP3	M	YES	YES	YES	NO	NO	YES	2 ITEMS	9 EVENTS/LICENSEE FAILED TO REPORT AS REQUIRED
										3 EVENTS/AGREE
	FDP3	P	CAN'T TELL	CAN'T TELL	YES	NO	NO	IN SUBSEQUENT LETTER		
	FJP3	M	CANT'T TELL	CAN'T TELL	YES	YES	NO	IN SUBSEQUENT LETTER		
77-23 (PHY. PROT.)	NDE3	M	YES	YES	YES	NO	NO	YES	YES (2 ITEMS)	

NAME PRAIRIE ISLAND UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed. Adequacy of Response (Disagree?)
77-23 (PHYS. PROT.)	NEG2	M	YES	NO	YES	NO	NO	NO		
	NED2	M	YES	NO	YES	NO	NO	YES		
	NED2	P	YES	YES	YES	NO	NO	YES		
	NED2	P	YES	YES	YES	NO	NO	YES		
	NED2	M	YES	CAN'; TELL	YES	NO	NO	NO		
	NED2	P	YES	YES	YES	NO	NO	NO		
77-26	FJE2	M	YES	YES	YES	YES	NO	YES		6 EVENTS/AGREE

Prairie Island Unit 1 means that a review of the 766 enforcement text and the noncompliance cause code without the supporting inspection report would not provide a sufficiently comprehensive understanding of a noncompliance and the circumstances of its origin.

We also reviewed possible sources of cues that may have aided inspectors in identifying noncompliance items. In approximately 17 percent of the cases a noncompliance resulted from inspector followup of an LER. In only three cases did a noncompliance result from a licensee-identified matter. For this case study, about 28 pecent of the noncompliances resulted from possible inspector cues. While these percentages are not insignificant, the majority of noncompliances did not result from possible cues to the inspector.

For 45 percent of the noncompliance items, licensee remedies to prevent recurrence of the event were specified in the inspection report, while 31 percent of the items were addressed in a subsequent letter.

The licensee's action on previously identified enforcement items was always timely and generally complete at each inspector visit in which these items were reviewed. On one occasion, the licensee had not resolved several items; this appears to be an isolated instance. In reviewing LERs, the inspector never disagreed with the licensee's reporting of the event. However, there was one occasion on which the inspector identified a group of items that the licensee failed to report. There were no events due to human failure that were serious from the regulatory point of view.

Figure A-4 is a profile of the total noncompliances attributable to human causes, excluding safeguards.

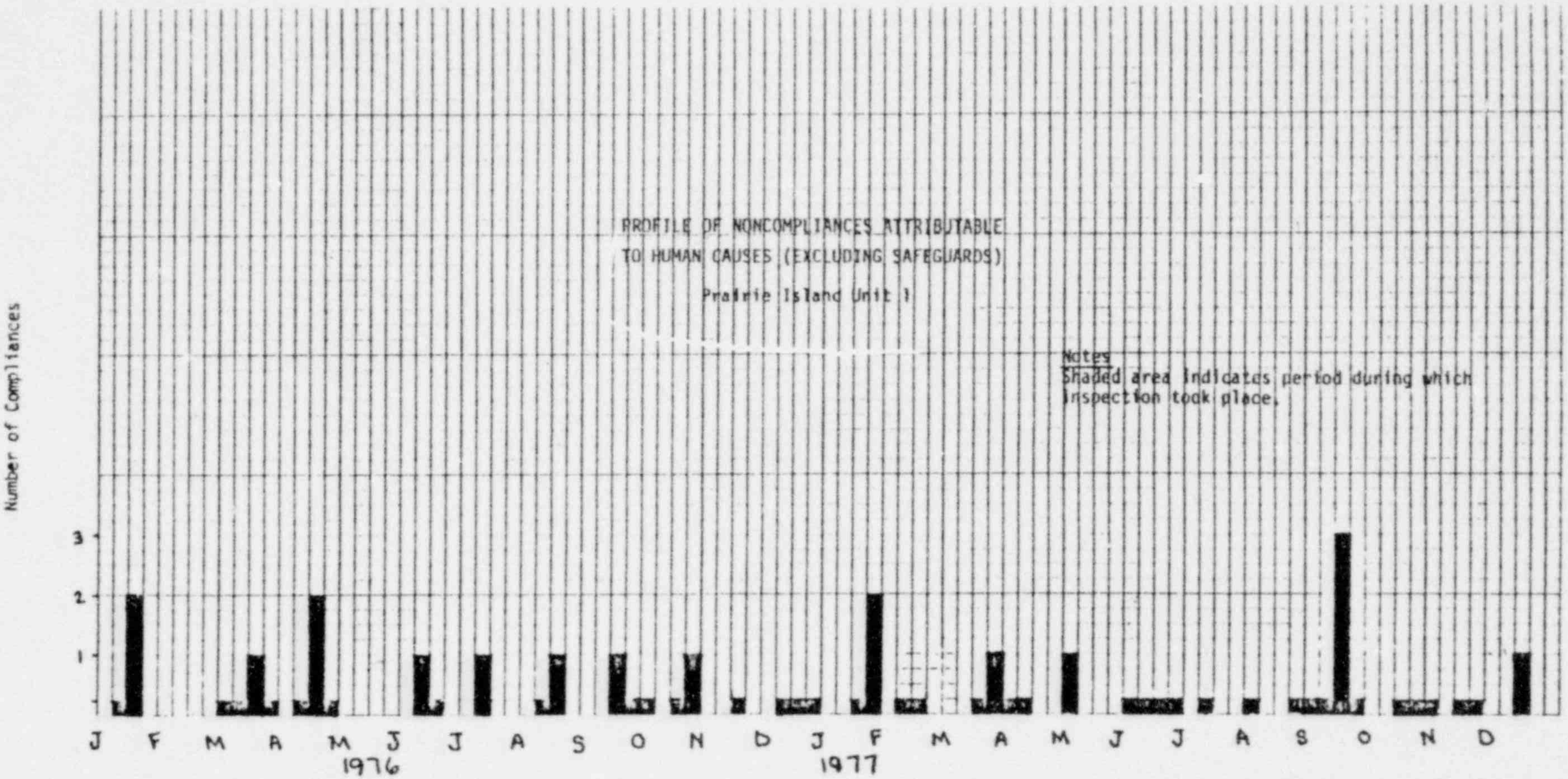

PROFILE OF NONCOMPLIANCES ATTRIBUTABLE
TO HUMAN CAUSES (EXCLUDING SAFEGUARDS)
Prairie Island Unit 1
Notes
Shaded area indicates period during which
inspection took place.
Number of Compliances
3
2
1
J F M A M J J A S O N D J F M A M J J A S O N D
1976
1977

FIGURE A-4

ZION UNIT 1 CASE STUDY

Review of the LER File for Zion Unit 1

During 1976 and 1977, events at this unit occurred in 26 systems, as shown in Table A-2 on page A-35. Six systems, the Containment Isolation System, Reactor Trip System, Airborne Radioactive Monitoring System, System Code Not Applicable, Emergency Core Cooling System, and Hangers, Supports, and Shock Suppressors* had large numbers of events - two of them extraordinarily large numbers - when compared to the other 20 systems. In addition, these six systems exhibited significant numbers of causally linked events. A number of these causally linked groups occurred repeatedly over long periods of time with only brief intervals between repetitions.

In the six systems with the most events, the Containment Isolation System had 20 events, Reactor Trip Systems had 27 events, System Code Not Applicable had nine events, the Airborne Radioactive Monitoring System had 11 events, the Emergency Core Cooling System had eight events, and the Hangers, Supports, and Shock Suppressors had eight events. The remaining 20 systems averaged 2.6 events over 24 months. Three of these 20 systems had a group average of 5.6 events per system, and removing these systems from the group of 20 resulted in an average of 2.0 events in 24 months for the remaining 17 systems. A detailed review of these 17 systems indicated six systems with casually linked events that appear related to failures in human performance (Reactor Core, three events; Feedwater Systems, four events; Area Monitoring System, four events; Containment Air Purification and Cleanup System, two events; Containment Heat Removal System, one event linked to a pre-1976 event; liquid Radioactive Waste Management System, three events).

Containment Isolation System

This system had 20 events in 24 months, as shown in Figure A-5. The licensee attributed one of these to human failure and the rest to component failure. We reclassified 15 of these 19 events as Teknekron ERC-M. and identified

*This is not a system code in the LER file, but as explained later in this section, Zion Unit 1 had a number of closely related and highly similar events involving these related components.

PROFILE OF TOTAL REPORTED EVENTS IN
CONTAINMENT ISOLATION SYSTEM

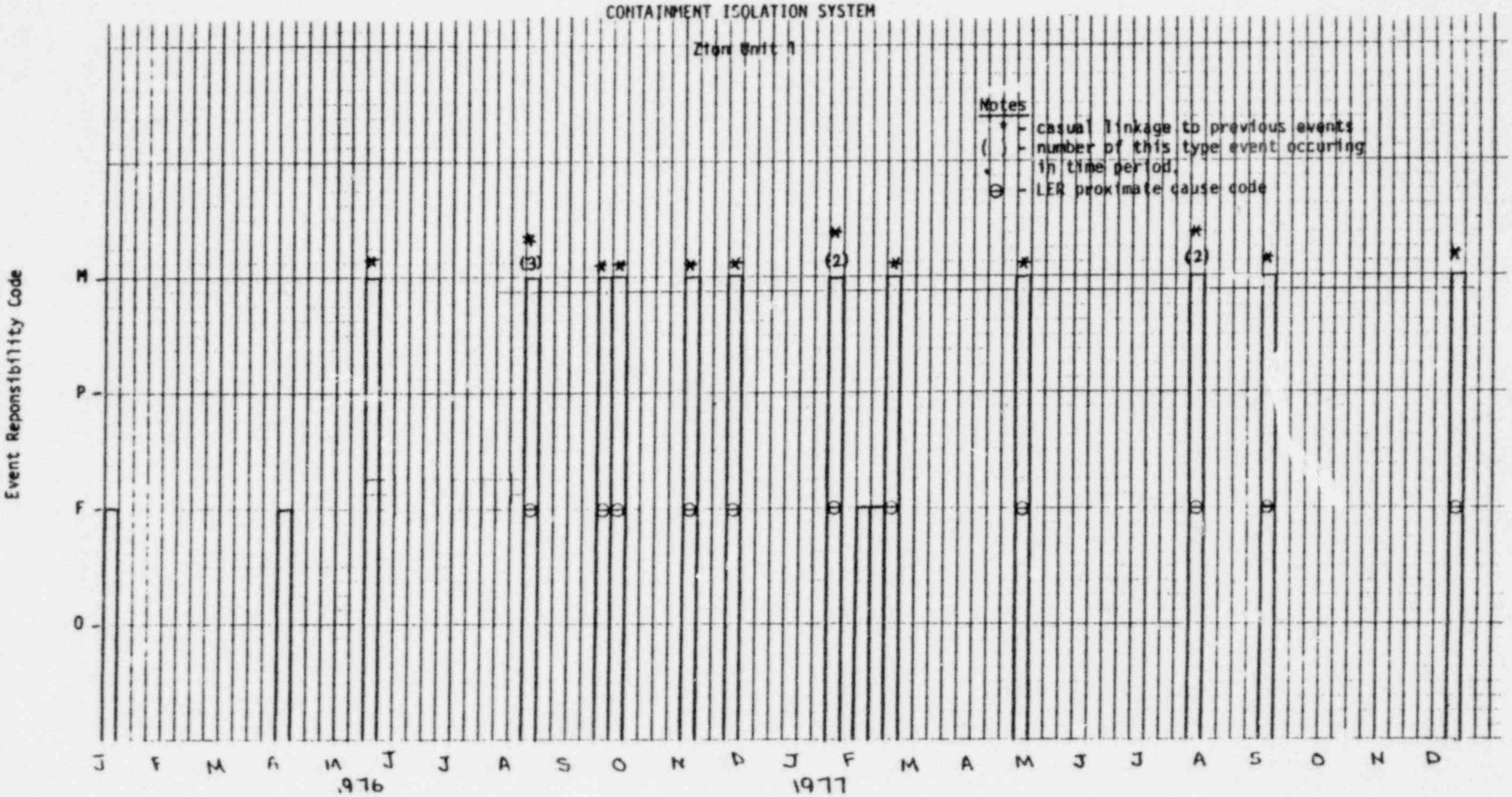


FIGURE A-5

three causally linked groups that included 15 of the 20 events. The dates of the first group of causally linked events, together with the cause assigned by the licensee and Teknekron's ERC Code, are:

<u>Date (licensee code/ERC)</u>

9-21-76(F/M)

11-04-76(F/M)

11-22-76(F/M)

1-16-77(F/M)

2-13-77(F/M)

9-01-77(F/M)

12-08-77(F/M)

The licensee stated that 9-21-76 event was similar to a previous event and identified the cause of excessive leakage of the containment purge isolation valve as a bulge on the valve's seating surface. The cause of the 11-04-76 event was identified as "cold air," so the licensee insulated and heat traced the valve and stated that no further problems were anticipated. On 11-22-76 the same event occurred; the cause was stated as overloaded circuits that cut off the heat tracing. In the 1-16-77 event, the licensee stated that the heat tracing was unable to keep the valve seats warm; they began using temporary space heaters. Extraneous material caught in the valve seats produced the 2-13-77 event. The 9-01-77 event stemmed from the valves' maladjustment. The cause of the 12-08-77 event was identified as failure to energize the heat tracing.

The second group of causally linked events is:

Date (licensee code/ERC)

4-07-76(F)

8-11-76(F/M) - 2 events

9-30-76(F/M)

1-23-77(F/M)

4-25-77(F/M)

7-23-77(F/M) - 2 events

The licensee identified the cause of the 4-07-76 event as a valve (inlet unloader valve) stuck open by "crud and rust." The valve was located in the system that provides compressed air to pressurize penetrations. On 8-11-76 two events occurred in which two identical components (solenoid valves) failed. For one event, the licensee stated the case as "...probably due to impurities in the instrument air system." The other event, involving an identical component, was listed as due to "varnish buildup." On 9-30-76, an identical event (solenoid valve failure) occurred with the same stated cause as the 8-11-76 event ("varnish buildup") The 1-23-77 event (solenoid valve failure) identified the same component failure as the 8-11-76 event; the stated cause was impurities in the instrument air supply. The 4-25-77 event was identical to the 1-23-77 event in all respects, but the licensee stated that new equipment was being installed. On 7-23-77 two separate events occurred, each identical to the previous 4-25-77 event. In this case, the licensee stated that monthly tests would be performed and the air line blown clean.

Two occurrences make up the third group of causally linked events:

Date (licensee code/ERC)

1-07-76(F)

5-18-76(M)

In the 1-07-76 event, a valve failed to close, and the stated cause of the failure was that the valve internals were galled (due to unknown reasons), causing mechanical binding. No further action was planned. The 5-18-76 event was identical, and the licensee stated that "... procedures were revised."

In summary, it appears that proper management attention to these three groups of causally linked events would have prevented their further occurrence. In the first group, events occurred about every two months over a 15-month period. The second group of events also extended over 15 months with an occurrence frequency of about two months. The third group of two events extended over four months.

Reactor Trip System

This system had 27 events in 24 months. The licensee attributed four events to human failure and all but one of the remaining 23 events to component failure. We reclassified 13 of these 23 events as ERC-M and identified four groups of causally linked events encompassing 17 of the 27 total events. The system profile is shown in Figure A-6.

The second group of causally linked events is:

Date (licensee code/ERC)

2-26-77(F)

3-19-77(F/M)

4-16-77(M)

5-12-77(F/P)

7-08-77(F/M)

7-29-77(F/M)

PROFILE OF TOTAL REPORTED EVENTS IN
REACTOR TRIP SYSTEM

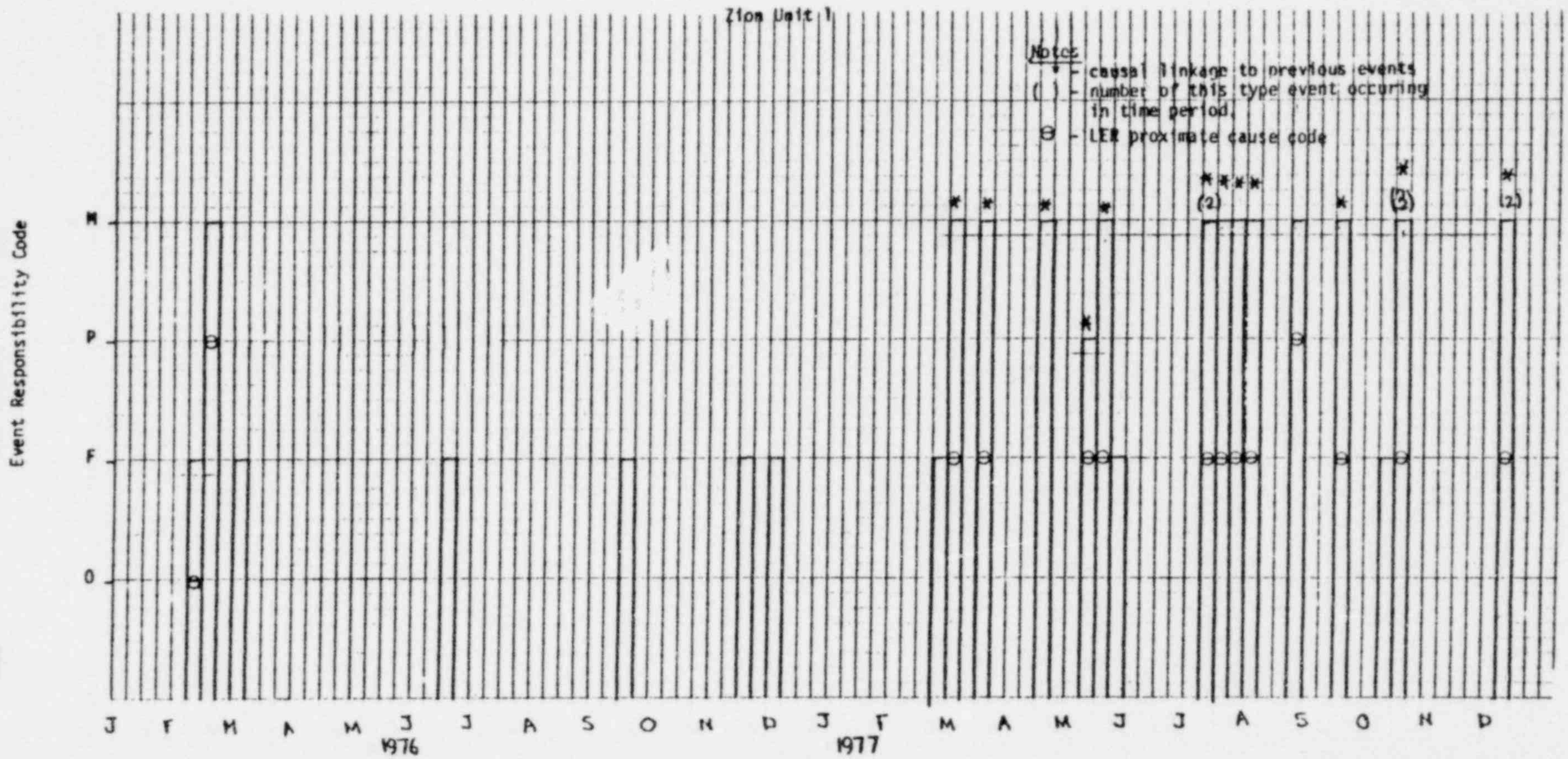


FIGURE A-6

On 2-26-77 the licensee received a low steam flow indication from steam generator 1D electrical instrumentation. The cause of the low flow indication was determined to be a defective coil in the Fischer-Porter flow transmitter. On 3-19-77 an identical failure occurred in the 1D steam generator instrumentation, with the identical cause. On 4-16-77, a similar failure occurred in the 1D steam generator, but this time the licensee identified in the cause as "loss of fluid in the DP cell for the differential pressure transmitter." The failed transmitter was replaced with a spare and returned to service. On 5-12-77, a similar event to the 4-16-77 event occurred in steam generator 1D. The licensee identified the cause as "apparently due to an intermittent connection, since the problem disappeared when the transmitter was replaced." On 7-08-77 the licensee identified a Fischer-Porter transmitter out of calibration in a situation similar to the 5-12-77 event. On 7-29-77 the licensee again reported low steam flow indication for steam generator 1D and stated the cause to be sediment plugs in the differential pressure lines on the Fischer-Porter transmitter.

The third group of causally linked events is:

Date (licensee code/ERC)
11-17-76(F)
7-19-77(F/M)
8-06-77(F/M)
9-14-77(F/M)
12-08-77(F/M)

On 11-17-76 the licensee reported a failure in the loop D instrumentation, a defective lead/lag module made by Hagan Controls. On 7-19-77 a defective Hagan Controls lead/lag module failed in the instrumentation for the pressurizer pressure channels. On 8-06-77, the set point of a Barton Model 386 pressurizer level transmitter was found to have drifted. This event is linked to the event of 7-19-77 because both involved failure in the pressurizer instrumentation. It appears that management should have examined all the pressurizer instrumentation at that time. On 9-14-77 another instrumentation failure occurred and was identified by the licensee as a "recurring problem" involving a Hagan Corporation signal summator. On 12-08-77, the licensee reported an event identical to the 8-06-77 event.

The fourth group of causally linked events is:

<u>Date (licensee code/ERC)</u>

10-21-77(F)

10-28-77(F/M)

10-31-77(F/M)

12-09-77(F/M)

On 10-21-77 the licensee reported that the setpoints of the steam generator level transmitters had drifted. The licensee rezeroed and recalibrated the Fischer-Porter transmitters. On 10-28-77 setpoint drift occurred in the reactor coolant flow transmitter. The licensee rezeroed and recalibrated the Fischer-Porter transmitter, stated an intention to study and to "trend" setpoint drift and remarked that no further action was required. On 10-31-77, during testing, the licensee found that the reactor coolant flow transmitters in loop D had experienced setpoint drift. The licensee recalibrated these Fischer-Porter transmitters. On 12-09-77 the steam flow from steam generator loop A was found to be reading low, and the cause was found to be setpoint drift of the Fischer-Porter flow transmitter.

These four causally linked groups have been established on the basis of subsystem location, equipment manufacturer, and function. Groups one and three may be crosslinked since both involve Hagan Controls equipment; Group four and group two may be crosslinked since both involve loss of indication and Fischer-Porter instrumentation (though somewhat different failure modes).

The sheer number of these apparently related events and the time period over which they occur seem to indicate an inability on the part of facility management and personnel to technically identify fundamental causes of problems and to effectively manage their resolution.

Airborne Radioactive Monitoring System

Eleven events occurred in this system in 24 months, and the system profile is shown in Figure A-7. The licensee attributed two of these to human failure, two events to other causes and the remaining seven events to component failure. We reclassified all seven component failures as Teknekron code ERC-M. We reclassified one of the two events classified by the licensee as "other" as ERC-M and one as ERC-F. Eight of the 11 events appear to fall into two causally linked groups.

Before describing the two groups of events, a single event on 4-13-77(O/M) deserves special mention due to its stated cause and resolution. On that date, the air ejector radiation monitor blower tripped out of service. The licensee stated that the blower tripped because the monitor cabinet was overheated due to poor ventilation. The licensee's solution: "The monitor cabinet was opened slightly to allow better ventilation."

The first group of causally linked events is:

Date (licensee code/ERC)
7-01-76(F/M)
11-12-76(F/M)
8-28-77(F/M)

On 7-01-76 the containment purge iodine monitor was declared inoperable due to a blower failure. The licensee stated that "the failure of the blower is directly related to its continuous operation," and that "an equipment lubrication and preventive maintenance program is in operation at this time." This statement indicated an awareness of the cause and potential generic resolution of the event. On 11-12-76 the gas decay tank monitor failed. The licensee

PROFILE OF TOTAL REPORTED EVENTS IN
AIRBORNE RADIOACTIVE MONITORING SYSTEM

Zion Unit 1

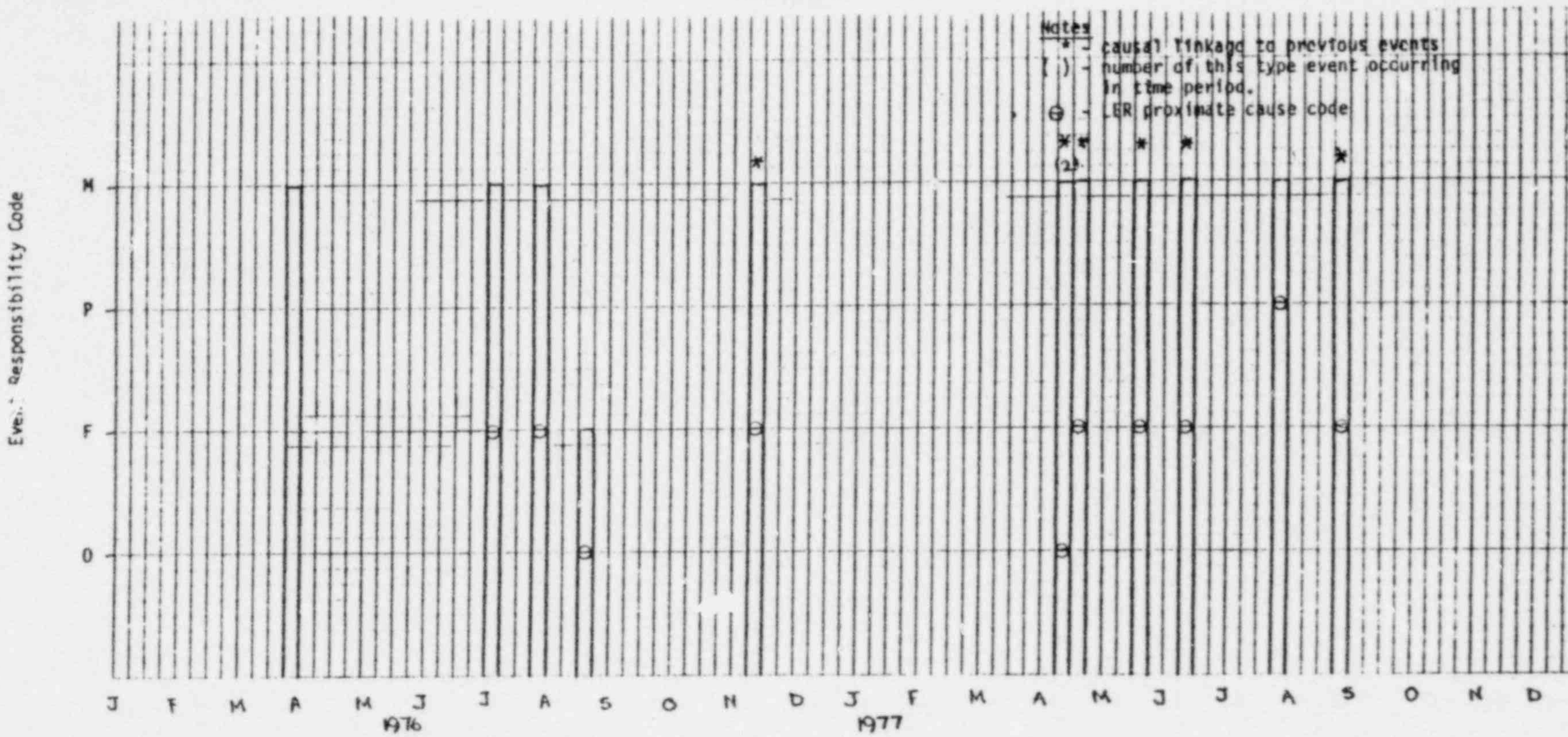


FIGURE A-7

attributed the failure to "...constant operation of the monitor." On 8-28-77 the pump for the containment particulate radiation monitor failed. The licensee stated that "...cause of pump failure was approximately 10,000 hours of continuous use." The pump was replaced.

The second group of causally linked events is:

<u>Date (licensee code/ERC)</u>

8-16-76(O/F)

4-09-77(O/M)

4-19-77(F/M)

5-21-77(F/M)

6-14-77(F/M)

On 8-16-76 the containment radiation monitors for gas and particulates were declared inoperable due to electrical problems. The licensee stated that "inoperability of the monitors was due to blown fuses in the circuits which control input to blowers and monitors. Cause for fuse failure unknown. Fuses replaced and monitors returned to service." On 4-09-77 the containment radioactive gas monitor became inoperable. The stated cause and response were "loss of contact between instrument drawer and instrument panel. Contact was cleaned and restored, with the monitor responding correctly." The event of 4-19-77 was identical to the 4-09-77 event. The licensee-stated cause was "plug connector was worn from opening and closing drawer for monitor surveillance and other related periodic checks." On 5-21-77 the containment purge radioactive iodine monitor failed. The stated cause was identical to the 4-09-77 event. On 6-14-77 the passive gas failure monitor failed. The stated cause of the event was a capacitor failure that caused the circuit board in the instrument drawer to fail.

In summary, the first event in this system, which received special mention, was singled out because it indicates 1) a lack of management awareness of the potential generic implication of events and 2) a lack of management commitment to resolve identified causes of events with a permanent fix.

The first and second groups of causally linked events indicate that when the generic implication of events is identified, the management appears unable to implement effectively a preventive program over an extended time period.

Emergency Core Cooling System

This system had eight events in 24 months. The licensee attributed three events to human failure, four events to component failure, and one event to "other." We reclassified three of the four component failures and the event classified as "other" to human error. We found two groups of causally linked events comprising five of the eight total events.

The first group of causally linked events is:

Date (licensee code/ERC)

4-01-76(F/M)

6-23-76(O/M)

10-19-76(M)

On 4-01-76 the 1C accumulator level transmitters experienced setpoint drift. The licensee stated that "the Barton Model 384 level transmitters experienced instrument drift. There is a very tight tolerance on these transmitters due to an improper application." On 6-23-76 the 1D accumulator was found to be overfilled. The licensee identified the cause as "apparently due to momentary backleakage of reactor coolant water through check valves into the accumulator." The licensee resolved this by draining the accumulator to the proper level and resuming power operation. On 10-19-76 the accumulator level transmitters for the 1A, 1B, and 1C accumulators drifted high. The licensee stated the cause as "inadequacy of presently installed transmitters Barton Model 384 for the given measuring range. Plans are being made to replace these transmitters."

The second group of causally linked events is:

Date (licensee code/ERC)
1-26-77(F)
1-28-77(F/M)

On 1-26-77 the 1A accumulator discharge valve failed to open after closing. The licensee stated that "...a long-term solution is being investigated...." and listed the cause as, "the contacts in the motor operator control center were hung up." On 1-28-77 an identical event occurred in the 1B accumulator.

To summarize, the first group of causally linked events indicates a management willingness to tolerate identified technical deficiences in equipment design and application in safety-related systems. The first and second groups of events show a lack of management willingness to explore generic causes of events and implement immediate resolution. When aware of the technical causes of events, the frequency of event occurrence appears to guide timeliness of resolution by management.

Hangers, Supports, and Shock Suppressors

This "system" is unique in that it is not classified as a system in the LER file codes but as a component. However, it is a component that is present in most, if not all, facility systems; and its absence from the system list may indicate a weakness in that data system. For the purpose of this analysis, the events identified as "Hangers, Supports, and Shock Suppressors" under various systems were collected and reviewed as we would a system.

The licensee identified a total of nine hydraulic snubber failures due to the escape of hydraulic fluid past thread seals. The first event on 2-30-77 involved the pressurizer snubbers. Not until 8-06-77 was this type of event reported again, and eight events of this type occurred in hydraulic snubbers in eight different systems from 8-06-77 to 11-09-77. The last event on 11-09-77 was similar to the 2-03-77 event since the pressurizer snubbers were involved. The licensee stated that the hydraulic snubbers in the pressurizer

Table A-2

LERs BY SYSTEM AT ZION UNIT 1 - 1976 and 1977

Containment Isolation System	Engineered Safety Features Instrumentation System	System Code Not Applicable	Reactor Containment System	Chemical Volume Control & Liquid Poison System	Reactor Trip System	Process & Effluent System
1-07-76(F)	1-08-76(O/P)	1-21-76(O)[1]	1-22-76(M)	1-28-76(F)	2-09-76(O/F)	2-20-76(F)
4-07-76(F)	9-23-76(F)	3-18-76(M)	5-04-76(O)	2-03-76(F)	2-21-76(P/M)	9-22-77(F/P)
5-18-76(M)[3]	1-27-77(F)	3-19-76(M)	5-26-76(P/M)[2]	2-27-76(F)	3-05-76(F)	9-25-77(F)
8-11-76(F/M)[7]		4-13-76(O)		1-28-77(F)	6-18-76(F)	
8-11-76(F/M)[8]		6-25-76(P/M)[5]		3-22-77(F/M)[21]	9-17-76(F)	
8-11-76(F/M)[8]		8-05-76(O)		5-30-77(P)	11-17-76(F)	
9-21-76(F/M)[5]		8-11-76(O/F)			12-01-76(F)	
9-30-76(F/M)[8]		11-30-76(F)			2-26-77(F)	
11-04-76(F/M)[11]		2-24-77(O)			3-03-77(F/M)[18]	
11-22-76(F/M)[13]					3-19-77(F/M)[19,20]	
1-16-77(F/M)[13]					4-16-77(M)[24,25]	
1-23-77(F/M)[8]					5-12-77(F/P)[27]	
2-03-77(F)					5-15-77(F)	
2-10-77(F)					5-31-77(F)	
2-13-77(F/M)[13]					7-08-77(F/M)[27]	
4-25-77(F/M)[8]					7-08-77(P/M)[30]	
7-23-77(F/M)[8]					7-19-77(F/M)[31,32]	
7-23-77(F/M)[8]					7-29-77(F/M)[27]	
9-01-77(F/M)[11]					8-06-77(F/M)[33]	
12-08-77(F/M)[13]					8-23-77(P/M)[2]	
					9-14-77(F/M)[32]	
					10-07-77(F)	
					10-20-77(F/P)[36]	
					10-21-77(F)	
					10-28-77(F/M)[37]	
					10-31-77(F/M)[37]	
					12-08-77(F/M)[36]	
					12-09-77(F/M)[37]	

Table A-2 (Cont.)

LERs BY SYSTEM AT ZION UNIT 1 - 1976 and 1977

Failed Fuel Detection System	Reactor Core	Feedwater System	Gas Radioactive Waste Management System	Airborne Radioactive Monitoring System	Emergency Core Cooling System	Fire Protection System
2-25-76(P)	2-26 76(O/P)	3-05-76(F/M)	3-12-76(F/M)	3-24-76(M)[2]	4-01-76(F/M)	4-27-76(M)
	7-16-76(O/P)	8-08-76(F)	2-01-77(P)	7-01-76(F/M)	6-23-76(O/M)[4]	5-04-76(F/P)
	7-30-76(O/M)[6]	12-03-77(F)		7-30-76(F/M)	9-16-76(F)	
		12-08-77(F/M)[39]		8-16-76(O/F)	10 19-76(M)[10]	
				11-12-76(F/M)[12]	1-26-77(F/M)[17]	
				4-09-77(O/M)[22]	1-28-77(F/M)[17]	
				4-13-77(O/M)	2-18-77(P)	
				4-19-77(F/M)[26]	12-18-77(P)[2]	
				5-21-77(F/M)[26]		
				6-14-770F/M)[26]		
				7-27-77(P/M)		
				8-28-77(F/M)[12,34]		

Table A-2 (Cont.)

LERs BY SYSTEM AT ZION UNIT 1 - 1976 and 1977

Process Sampling System	Circulating Water System	Hangers, Supports Shock, Suppressors	Main Steam Isolation System	Containment Combustible Gas Control System
11-23-76(F)	12-07-76(O)	2-03-77(F)	10-07-77(F)	11-30-77(F)
	12-14-76(O)[2]	8-06-77(F)	12-03-77(F)	
	1-31-77(O/M)	9-19-77(F/M)[35]		
	1-31-77(O/M)	9-21-77(O/M)[35]		
	1-31-77(O/M)	10-04-77(F/M)[35]		
	2-09-77(O/M)	10-04-770F/M)[35]		
	3-09-77(P/M)	11-01-77(F/M)[35]		
		11-09-77(F/M)[34,38]		

Table A-2 (Cont.)

LERs BY SYSTEM AT ZION UNIT 1 - 1976 and 1977

Area Monitoring System	Emergency Generator System	Containment Air Purification Cleanup System	Containment Heat Removal System	Reactor Coolant System	Residual Heat Removal System	Liquid Radioactive Waste Mgt. System
5-13-76(F)	6-21-76(F)	9-14-76(O/M)[5]	9-23-76(F/M)[5]	10-04-76(P/M)[9]	10-06-76(F)[2]	10-20-76(F)
12-10-76(F)	9-24-76(F)	1-21-77(M)[16]				6-03-77(P/M)[28,29]
12-12-76(F/M)[14]						10-28-77(P/M)
12-15-76(F/M)[15]						

NOTES: FOR TABLE A-2

1. Vendor error in accident analysis - no immediate action required.
2. Violation of technical specifications.
3. Identical to 1-07-76 event.
4. This event appears to be related to the 4-01-76 event. Management didn't follow up on 4-01-76 event to substantiate the cause. Had they done so, it appears this event would not have occurred.
5. Similar events occurred in a previous period of record.
6. Related to previous events 2-26-76 and 7-16-76 in that operating personnel are having difficulties handling xenon oscillations.
7. Identified by licensee as a repetitive occurrence - a check of this record period provides no indication of the repetitive event.
8. Related to previous event 4-07-76 in that this event had potential generic implications which were not identified by the licensee.
9. This event was improperly classified in LER file under "Reactor Core Isolation Cooling System."
10. Failure of management to follow up on 4-01-76 event to which this is identical.
11. This event related to event of 9-21-76 in that the 9-21-76 event cause was identified in such a way that a permanent fix was not utilized.
12. Event of 7-01-76 indicated licensee understanding that air monitoring systems which operate continuously require a preventive maintenance program - the understanding does not appear to have been applied beyond the containment purge monitoring system.
13. Similar to 11-04-76 event.
14. Similar to 12-10-76 event.
15. A result of preceeding 12-10-76 and 12-12-76 events.
16. Similar to 9-14-76 event.
17. Identical to previous event 1-26-77 in a redundant system.

18. Similar to 6-18-76 event which occurred in a redundant piece of equipment.

19. Improperly classified in LER file as "Condensate Storage Facility."

20. Similar to 2-26-77 event.

21. Appears related to 10-06-76 event filed under "Residual Heat Removal System" - the maintenance performed for previous event may have been incomplete.

22. Similar to 8-16-76 event in same component group. Had management followed up on generic cause of fuse failure in 8-16-76 event this event would probably not have occurred.

23. The type of fix implemented for this event denotes lack of management attention to detail of plant design, i.e., where else in plant would a failure of this type occur due to overheating; is the problem generic?

24. Improperly classified in LER file as "Main Steam Supply System."

25. Related to previous event 3-19-77 in that both events occurred in the same steam generator instrumentation package (ID) with the indication of failure for both events being the same, i.e. low flow for the first event, zero flow for the second. Inadequate review of first event, probable cause of second event.

26. Related to 4-09-77 event. Improper review and resolution of previous event res[illegible] in this event.

27. Maintenance and cause identification performed to resolve previous event of 4-16-77 was apparently incomplete resulting in this event.

28. Related to 10-20-76 event - management didn't follow up on previous event.

29. Event improperly classified under "System Code Not Applicable."

30. Event improperly classified under "Feedwater Systems."

31. Event improperly classified under "Reactor Core Instrumentation."

32. Previous event 11-17-76 was due to failure of Hagan lead/lag module - the licensee stated "cause of module failure will be documented...after repairs are made." Apparently no generic follow up by management.

33. During previous maintenance to rectify 7-19-77 event not all pressurizer instrumentation was rechecked and recalibrated. Only the affected equipment received maintenance.

34. Appears that preventive maintenance program identified in 7-01-76 event has not been carried out.

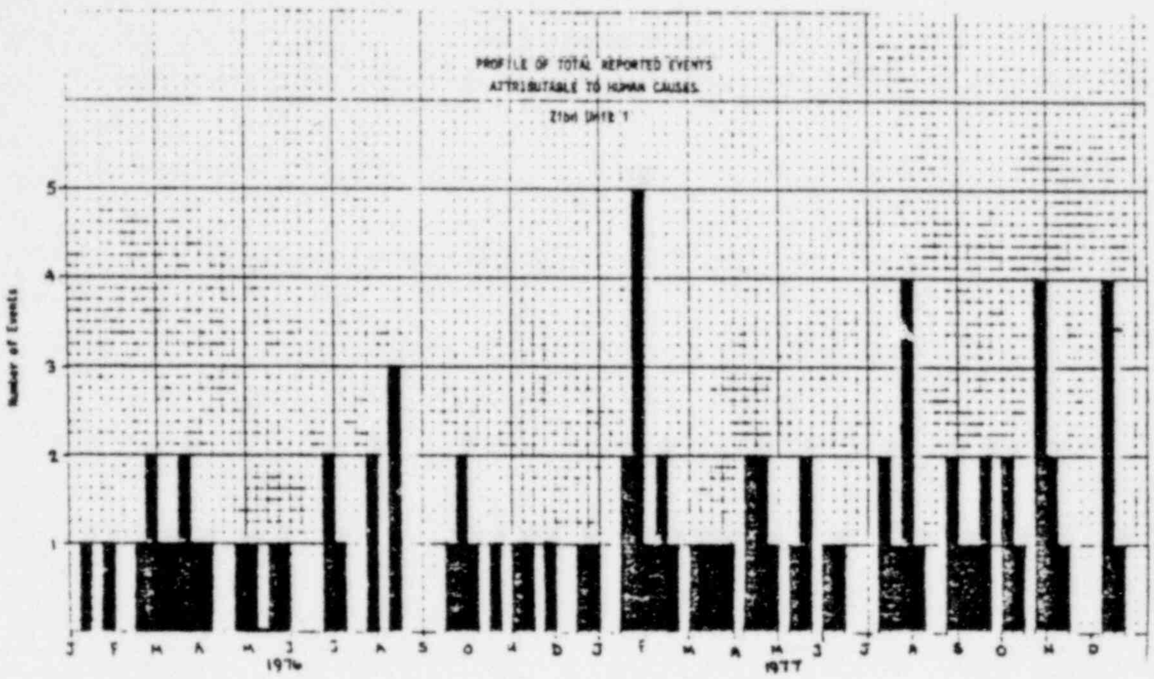
35. Related to 8-06-77 event in that management did not apparently view the problem generically.

36. Management failed to view 8-06-77 as generic and repeatable.

37. Management failed to view 10-21-77 event as generic and repeatable.

38. Event in this system occurred previously 2-03-77.

39. Similar to previous event 12-03-77.



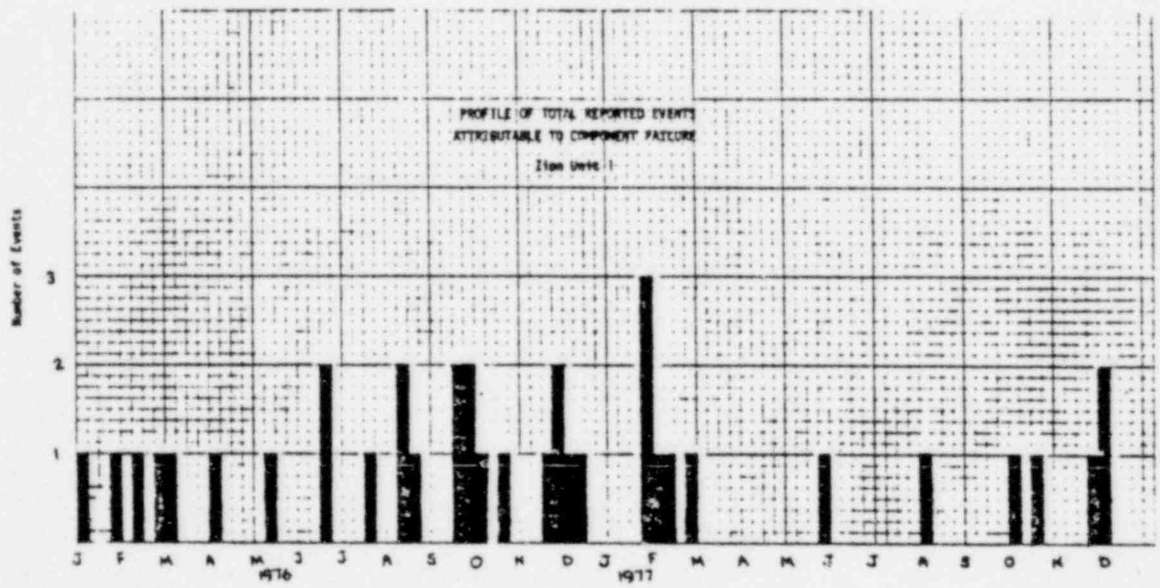


FIGURE A-8

Zion Unit 1 Performance Profiles

system would be replaced with mechanical ones, "since the fluid probably leaked out due to high temperature environment."

As a result of the 11-02-77 event of hydraulic snubber failure, the licensee stated that "inspections each refueling cycle identify leaking seals. No further corrective action is deemed necessary."

The 11-02-77 event and the 11-09-77 event present an interesting view of facility management perception of and response to generic event causes.*

Figure A-8 on the previous page shows the Zion Unit 1 profiles of total reported events attributable to human causes together with the profile of events attributable to component failure.

Review of Inspection Reports and 766 System Data File for Zion Unit 1

When we reviewed the 766 system data file and associated inspection reports for 1976 and 1977, we found 60 inspection reports detailing NRC I&E inspector findings. Twenty-seven of these reports identify a total of 78 items of noncompliance. Two of these reports resulted in civil action against the licensee. Of the 78 items of noncompliance, ten involve physical protection and are identified in two separate inspection reports.

*Point Beach Unit 1 also reported an event in this "system" on 10-21-77. They stated the cause as personnel error. The event itself was described as "During...testing of safety-related shock suppressors according to T.S. 15.4.13.2...snubber did not lock up when specified load rate was applied." Their cause description and response: "C[illegible] valve...found to be improperly set. Control valve was properly set, and [illegible]ber retested satisfactorily. Similar snubber control valves are being rechecked." The response of Point Beach Unit 1 in checking similar snubber control valves shows that some licensees look for generic implications beyond the "conventional" system level.

Matrix A-2 summarizes the findings of each inspection report and associated 766 system data file entries that resulted in noncompliances. Two reports in which LERs were reviewed and two reports covering management inspections are also included. Not including noncompliances due to physical protection and those for which reports were not available, 33 of 62 items were assignable to ERC-M, and 25 were assignable to ERC-P.

There was generally good agreement between the noncompliance cause code as listed in the 766 system and the detailed discussions in the "Report Details" section of the available inspection reports. Less than nine percent of the noncompliance cause codes either were ambiguous or did not agree with the inspection report details. There was also strong agreement between the enforcement text provided for each item of noncompliance identified in the 766 system and the "Enforcement Actions" section of the associated inspection report. However, there was less agreement between the noncompliance cause code and the 766 enforcement text. Approximately 47 percent of the items bore either an ambiguous or irrelevant relationship to each other. There is not enough detail in the 766 enforcement text and the associated noncompliance cause code (without analyzing the supporting inspection report) to provide a sufficiently comprehensive understanding of the noncompliance and the circumstances of its origin.

We reviewed possible sources of cues that may have aided inspectors in identifying noncompliance items. In approximately 32 percent of the cases, a noncompliance resulted from inspector followup of an LER. Almost 20 percent of the noncompliances resulted from inspector followup on a licensee-identified matter. Thus for Zion Unit 1, more than 50 percent of the noncompliance items resulted from inspector cues.

MATRIX A-2

Review of 766 File and Inspection Reports for

Zion Unit 1

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code In 766 Agree With IE Report	Does NC Cause Code In 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated In IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
76-02	FCS2	M	YES	CAN'T TELL	YES	NO	NO	YES	YES (4 ITEMS) NO (2 ITEMS)	
	FCS2	O	YES	CAN'T TELL	YES	NO	YES	YES		
	FMY3	M	NO	NO	YES	YES	NO	YES		
	FMY3	M	YES	CAN'T TELL	YES	NO	YES	YES		
	FDP2	M	YES	CAN'T TELL	YES	NO	NO	YES		
76-03	JAY3	M	YES	NO	YES	NO	NO	IN SUBSEQUENT LETTER	INCOMPLETE (1ITEM) YES (1 ITEM) NO (2 ITEMS)	
76-07	ESB2	M	NO	YES	YES	NO	YES	YES	YES (6 ITEMS) NO (6 ITEMS)	3 ITEMS/DISAGREE

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
	FDG2	P	YES	YES	YES	YES	NO	IN SUBSEQUENT LETTER		
	FCA2	P	YES	CAN'T TELL	YES	YES	NO	YES		
	FJP3	P	YES	YES	YES	NO	YES	YES		
	FJR3	-	CAN'T TELL	CAN'T TELL	YES	YES	NO	YES		
	FJR3*	P	YES	YES	YES	YES	NO	IN SUBSEQUENT LETTER		
	FPE3*	P	YES	CAN'T TELL	YES	NO	NO	IN SUBSEQUENT LETTER		
76-10	FCL2	M	NO	NO	YES	NO	NO	IN SUBSEQUENT LETTER	NOT INSPECTED	2 IDENTIFIED IN 766, BUT NOT EVIDENT IN IE REPORT
76-11	MGT. INSP.							-DISCUSSED COMMON CAUSE FACTORS CONTRIBUTING TO OPERATING PROBLEMS. -CONTINUED LICENSEE EFFORT TO MINIMIZE FUTURE INCIDENTS CAUSED BY OPERATOR ERROR AND IMPORVE PLANT PERFORMANCE.		

NOTES

(*) Repeat noncompliance

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement items	LER's Reviewed Adequacy of Response (Disagree?)
(1) 76-12	ASA1	M	YES	CAN'T TELL	YES	YES		NO	NONE REVIEWED DURING INSPECTION	
	FPG2	M	YES	YES	YES	YES		NO		
	ABC1	M	YES	CAN'T TELL	YES	Y[illegible]		NO		
76-13	FPG2	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER	YES (1 ITEM) NO (2 ITEMS)	
	FDG2	P	YES	YES	NO	YES		YES		2 ITEMS/AGREE
	ABC2	P	NO	YES	YES	NO	YES	YES		
	FDG2	F	YES	CAN'T TELL	YES	YES		IN SUBSEQUENT LETTER		
76-17	FDN2	P	YES	NO	YES	YES		YES	YES (11 ITEMS)	10 ITEMS/AGREE 1 ITEM/OPEN

NOTES

(1) Licensee fined

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
	FPE2	P	YES	YES	YES		YES	YES		
76-20	FJG3	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER	NONE INSPECTED	
	FJG3	P	YES	YES	YES	NO	NO	YES		
76-21 (PHYS. PROT.)	FPH2	P	YES	YES	YES	NO	NO	IN A SUBSEQUENT LETTER	NONE INSPECTED	
76-22	FJP3	P	YES	YES	YES	YES		YES	YES (3 ITEMS) NO (2 ITEMS)	
	FCS2*	P	YES	YES	YES	NO	NO	YES		
76-25	FPE3	P	YES	YES	YES	NO	NO	YES	NONE INSPECTED	4 ITEMS/AGREE

NOTES

(*) Repeat noncompliance

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
(2) 76-26	FJE2	P	YES	NO	YES	YES		IN SUBSEQUENT LETTER	NONE INSPECTED	
76-29 (Phys Prot.)	RMC2	P	YES	CAN'T TELL	YES	NO	NO	YES	YES (3 ITEMS) NO (2 ITEMS)	
	RLL2	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
	RME2	M	YES	CAN'T TELL	YES		YES	YES		
76-30	FJP3	M	YES	CAN'T TELL	YES	NO	YES	YES	NONE INSPECTED	
76-31	ASE2	M	YES	CAN'T TELL	YES	NO	NO	YES	NONE INSPECTED	1 ITEM/DISAGREE
	FDG2*	M	YES	CAN'T TELL	YES	NO	YES	IN SUBSEQUENT LETTER		
	FDG2	M	YES	NO	YES	NO	NO	NO		

NOTES

(*) Repeat noncompliance

(2) Inspection to follow-up on an LER, unexplained boron dilution event, October 2, 1976.

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
	FCJ2	F	YES	YES	YES	YES		YES		
	FJP3	M	NO	NO	YES	YES		NO		
76-32	FCA2	M	YES	CAN'T TELL	YES	YES		YES	NO (1 ITEM) YES (1 ITEM)	13 ITEMS/AGREE 5 ITEMS/DISAGREE
(3) 77-05									YES (4 ITEMS)	7 ITEMS/AGREE
77-07	FPH2	P	YES	YES	YES	NO	NO	YES		
77-08	FJJ3	M	YES	NO	YES	NO	NO	NO	YES (1 ITEM) 1 ITEM	1 ITEM/AGREE
	FJE2	P	YES	YES	YES	NO	NO	NO		
	FJE2	P -	YES	YES	YES	NO	YES	NO		

NOTES

(3) Inspector noted that LER write-ups were scant and that all facts available were not presented to make a complete evaluation.

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enfoncement Items	LER's Reviewed Adequacy of Response (Disagree?)
	FJK3	M	YES	NO	YES	NO	NO	NO		
77-09	FJC3	M	YES	CAN'T TELL	YES	NO	NO	YES	YES (2 ITEMS) NO (1 ITEM)	
77-10		REPORT NOT AVAILABLE (4 ITEMS OF NONCOMPLIANCE)								
77-11	"RB3	M	YES	YES	YES	NO	NO	NO		
77-15	FDG3	M	YES	NO	YES	YES		YES	YES (7 ITEMS) 1 ITEM	1 ITEM/AGREE
	FJE2	M	YES	NO	YES	YES		YES		
	FCJ2	M	YES	NO	YES	YES		YES		
	FDG2	M	YES	NO	YES	YES		YES		

Teknekron, Inc.

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
	FPG2	M	YES	YES	YES	NO	NO	NO		
	FPG2	M	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
	DOH2	M	YES	YES	YES	YES		NO		
	ART3	M	YES	YES	YES	NO	NO	NO		
	FJP3	M	YES	YES	YES	YES		YES		
	FDF2	M	YES	YES	YES	NO	NO	NONE REQUIRED		
	FDF2	M	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
(1) (4) 77-16	FJP3	M	YES	YES	YES		YES	CAN'T TELL		

NOTES

(1) Licensee fined

(4) July 8, major event water hammer, safety injection event due to human error.

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
	FPE2	P	YES	NO	YES		YES	CAN'T TELL		
	FJF2	P	YES	YES	YES	YES		NO		
	FPF2	P	YES	YES	YES		YES	NO		
	FES2	P	YES	YES	YES		YES	NO		
77-17	FJG2	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
77-18	FJH2	M	YES	YES	YES	NO	NO	NO		8 ITEMS/CAN'T TELL
77-19 (Phys. Prot.	RLL2	M	YES	CAN'T TELL	YES	NO	NO	YES	YES (4 ITEMS) NO (2 ITEMS)	
	RLD3		CAN'T TELL	CAN'T TELL	YES	NO	NO	NO		

NAME ZION UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On'a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enfonce-ment Items	LER's Reviewed. Adequacy of Response (Disagree?)
	RMD2	M	YES	CAN'T TELL	YES	NO	NO	YES		
	RML3	M	YES	CAN'T TELL	YES	NO	NO	YES		
	RLE3	P	YES	YES	YES	NO	NO	YES		
	NOE3	M	YES	YES	YES	NO	NO	YES		
	NOE3	M	YES	YES	YES	NO	NO	YES		
77-20		REPORT NOT AVAILABLE (2 ITEMS OF NONCOMPLAINCE)								
77-26										[illegible]9 ITEMS/AGREE
77-27	MGT. MTG.				DISCUSSED NEED FOR IMPROVED MANAGEMENT CONTROL					

For 50 percent of the noncompliance items, remedies specified by the licensee to prevent recurrence of the event were identified in the inspection report. Twenty-one percent of the items were addressed in a subsequent followup letter. However, the licensee's action on previously identified enforcement items was generally deficient. Nearly 70 percent of the inspection reports that specifically discuss "Licensee Action on Previously Identified Enforcement Items" indicated one or more items for which the licensee had not yet achieved compliance.

The inspector found the licensee's reporting of LERs unacceptable in 12 percent of the 74 total cases addressed in the inspection reports. This was because of the inspector's judgment that the licensee provided insufficient detail to substantiate the event. For 36 percent of the events, not enough detail was present in the inspection reports to make it clear whether the inspector had reviewed the LERs in detail.

Our review of the inspection reports revealed three events due to human failure that were serious from the regulatory point of view. The identification of these events and the subsequent determination of their seriousness was made possible by the inspection process. These events are summarized individually.

Radiation Exposure Incident - March 18, 1976 (as described in I&E Inspection Report No. 050-295/76-12)

On March 18, 1976 an employee received an 8.05 rem dose when he entered the cavity beneath the reactor vessel to determine the location of a water leak from the refueling cavity into the reactor cavity. The referenced inspection report describes the details of the event and the circumstances of its occurrence; we will not duplicate that information. However, part g of the inspection report, "Problems Revealed by this Incident," was enligntening and is reproduced here in its entirety:

g. Problems Revealed by this Incident

This incident revealed the follow' . apparent problems related to radiation protection:

(1) The unlighted, difficult-to-reach tunnel and cavity beneath the reactor were not recognized and treated as an extremely hazardous high-radiation area.

(2) Neither station management nor Radiation Protection personnel understood the source of the high radiation levels beneath the reactor. Radiation levels were vaguely attributed to the reactor vessel, not to the incore system. No effort had been made to relate the position of the withdrawn incore thimbles to the bottom of the vessel.

(3) None of the tunnel entries, which resulted in 3.5 man-rems of dose in addition to Employee A's 8 rems, produced very meaningful exposure rate data. Employee A knew only that exposure rates greater than 10 R/hr probably existed and that doses received during the previous entries by Employees C and D had exceeded the range of their 0-200 millirem pencil dosimeters.

(4) Radiation Protection neither prohibited Employee A from making a solo entry nor provided monitoring assistance, even though high radiation levels were known to exist in the area. Nor, as required by Procedure No. RP-253, was a special work permit issued to ensure proper monitoring, protective equipment, instructions, and approvals. Procedure No. RP-253 requires preparation of a special work permit for work resulting in a daily whole-body dose greater than 50 millirems, unless the work is otherwise approved in writing by the Radiation Protection Supervisor or the work is continually monitored by a Radiation Protectionman.

(5) Despite the known existence of high-radiation areas, Employee A was provided no high-range dosimetry, other than his film badge.

(6) There are indications that this incident may have been caused or at least contributed to by an ineffective working relationship between Radiation Protection and certain station management personnel.

The cause of the event was a performance deficiency assignable to the management "circle" in the FPM model. However, the manifestations of the event appear as either incorrect (paragraphs 1, 4, and 6 of the description) or missing components (paragraphs 2, 3, and 5 of the description) of the information flow along one or more of the arrows in the FPM model.

This occurrence resulted in a citation for three items of noncompliance and the institution of a civil penalty.

Boron Dilution Incident - October 3, 1976 (as described in I&E Inspection Report No. 050-295/76-26

On October 3, 1976, licensee personnel observed that pressurizer level changes and boron analysis over the previous 24 hours indicated that an unexplained dilution was in progress in Unit 1. The inspection report describes the details of the event and the circumstances of its occurrence, but the relevant section of the inspection report entitled "Management Interview" is reproduced here in its entirety:

Management Interview

An exit interview was conducted on October 15, 1976, with (Mr. X) and other members of the staff. The following items were discussed:

A. The inspector asked the licensee why valve 1IW0153 was open. The licensee stated there was no reason for the valve being open and did not know how it was opened. The inspector stated that valve 1IW0153 being open without justifiable reason was contrary to the requirements of Procedure SOI-7 and constituted an infraction against Technical Specification 6.2.A. (Paragraph 2.e, Report Details)

B. The inspector asked when the suspected leaking valve 1MOV-VC-8106 would be tested. The licensee stated the valve would be type C leak tested by October 16, 1976. The inspector requested that the licensee telephone in the results of the test by October 18 and the licensee agreed to do this.* (Paragraph 2.e, Report Details)

C. The inspector stated that it took six hours after a sample had revealed 864 ppm of boron in the reactor coolant system before boration was accomplished. The inspector stated that this was not considered to be a timely response and that during discussions with operating personnel regarding actions to be taken in future events that a more timely response should be emphasized. The licensee stated that from hindsight more timely boration would have been indicated but that during the event the emphasis was on finding the cause of the dilution. (Paragraph 2.3, Report Details)

D. The inspector suggested that the design of the injection seal water system be reviewed to determine if the alarm on the injection seal water tank level might be adjusted to give an earlier indication of undue flow out of the system. The licensee stated that if the level alarm was adjusted to alarm at a higher level in the tank, normal leakage out of the system would cause alarms and diminish usefulness of the level alarm. The inspector asked what the value of the normal leakage was. The licensee responded that the leakage was measured but did not recall the exact value.

The cause of the event is clearly assignable to management. However, the manifestations of the event and its aftermath appear as either <u>incorrect</u> (paragraphs B and C of the description) or <u>missing</u> components (paragraphs A and D of the description) of the information flow along one or more of the arrows in the FPM model.

The occurrence resulted in a citation for one item of noncompliance.

<u>Water Hammer and Safety Injection Event - July 8, 1977 (As described in I&E Inspection Report No. 50-295/77-16)</u>

The "Report Details" section describes this event:

1. On July 8, 1977, during performance of a periodic test by a licensed operator, a momentary distraction caused the operator to omit several steps of the procedure resulting in a reactor trip.

*The licensee notified the inspector October 21 of the results of the test. Test results revealed no significant leakage.

2. In response to the reactor trip, all systems functioned as designed. However, the auxiliary feedwater system flow control had been incorrectly adjusted after a previous test of the system; the maladjustment resulted in flow rates approximately three times higher than required (or desired) by current operating procedures.

3. Due to a clerical error, the current operating procedures had not been distributed for use, and the flow control adjustment had been performed with outdated procedures.

4. This series of events caused a system water hammer when the auxiliary feed pumps came on automatically. The water hammer was of sufficient magnitude to shake various transmitters located in the immediate vicinity; the shaking transmitters initiated a spurious safety injection.

5. When a safety injection is initiated, the system is designed to operate for 60 seconds in that mode. After 60 seconds, the operator is to reset the safety injection in accordance with a procedure for recovery from a false or inadvertent safety injection. Contrary to these procedures, personnel manually defeated the safety injection for 30 seconds prior to resetting it. This manual defeat of the safety injection signals preclude receipt of additional safety injection signals.

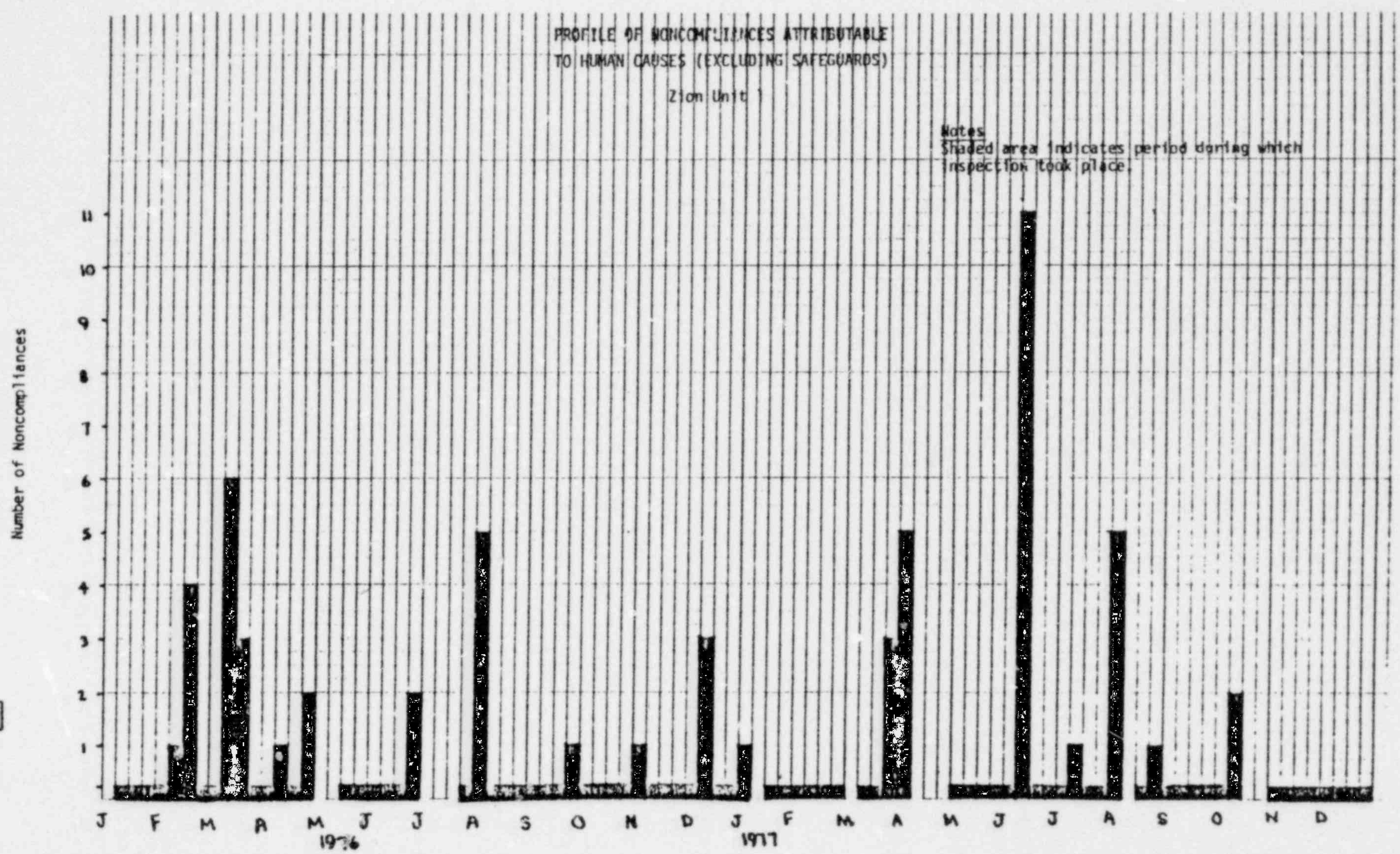
This event was caused by peformance deficiencies assignable to both management and personnel. However, the manifestations of the event preceded it in time and appear as either incorrect or missing components of the information flow along one or more of the arrows in the FPM model. The occurrence resulted in a citation for two items of noncompliance.

Including the last occurrence described, three serious events occurred at Zion Units 1 and 2 between July 8 and 12, 1977, two water hammers with consequent safety injection events and a pressurizer draining event. At the exit interviews following the management meetings held to investigate these events, inspectors informed the licensee of:

- the seriousness with which NRC viewed these events;
- observations involving the breakdown of management controls.

The NRC levied a civil penalty in a subsequent enforcement action.

Figure A-9 shows the noncompliance profile for Zion Unit 1.



PROFILE OF NONCOMPLIANCES ATTRIBUTABLE
TO HUMAN CAUSES (EXCLUDING SAFEGUARDS)
Zion Unit 1
Notes
Shaded area indicates period during which
inspection took place.
Number of Noncompliances
11
10
9
8
7
6
5
4
3
2
1
J F M A M J J A S O N D J F M A M J J A S O N D
1976
1977

FIGURE A-9

POINT BEACH UNIT 1 CASE STUDY

Review of the LER File for Point Beach Unit 1

During 1976 and 1977, 26 events occurred in 16 systems at this unit, as shown in Table A-3 on page A-63. Nineteen of these were reported as component failures; we reclassified one to Teknekron ERC-M. Two events were reported as "other" and we reclassified one as ERC-M. The remaining events were reported as human error (personnel error or defective procedures), which we converted to ERC-M or ERC-P. However, none of these conversions required reclassification on the basis of our review.

Five of the systems had more than one event; these systems averaged three events each over the 24-month period. A detailed review of the events in each system indicated only two causally linked groups of events.

The first group of causally linked events was in the Engineered Safety Features Instrumentation System. On 12-29-76 a differential reading was noted between the "B" steam generator steam line pressure instrument 1PT-478 and the redundant instruments 1PT-479 and 1PT-483. Investigation revealed a frozen point in the sensing line where the tubing exits the facade to enter the main building. The licensee stated "insulation on sensing line had a gap which allowed the line to freeze. Gap repaired and heat lamp installed." On 12-11-77 an identical event occurred.

The second group of causally related events occurred in the Air Conditioning, Heating, Cooling, and Ventilation System. On 4-30-77 an air damper did not operate properly. The licensee stated: "foreign matter in Johnson Service Company Model R-130-1 air regulator which obstructed orificed exhaust line.

Table A-3

LERs BY SYSTEM AT POINT BEACH UNIT 1 - 1976 and 1977

Feedwater System	Containment Isolation System	Engineered Safety Features Instrumentation System	Chemical, Volume Control & Liquid Poison System	Control Room Habitability System	Station Service Water System
1-08-76(F)	1-08-76(F)	1-10-76(F)	3-08-76(F)	3-10-76(F)(1)	6-16-76(F)
		11-30-76(F)	12-30-76(F)		6-15-77(F)
		12-29-76(M)(4)			10-31-77(P)
		10-10-77(M)(5)			12-21-77(F)
		12-11-77(O/M)(7)			

Table A-3 (Cont.)

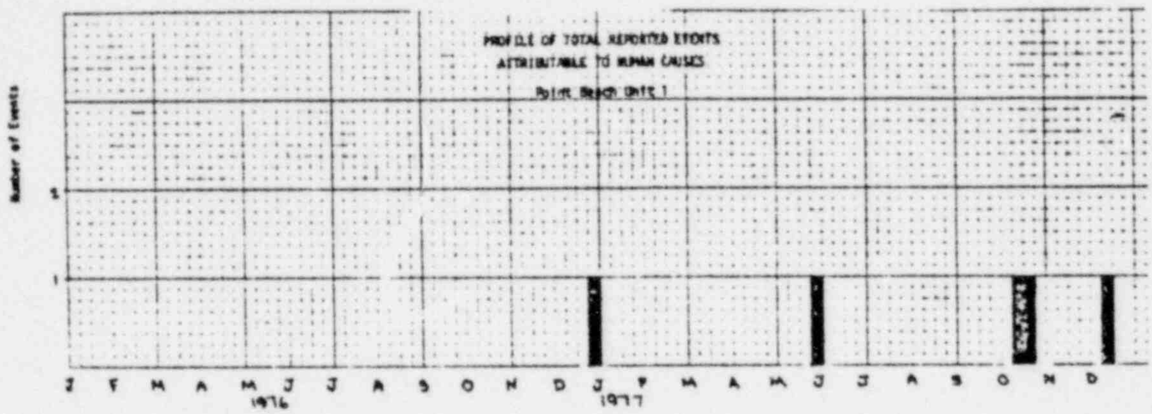
LERs BY SYSTEM AT POINT BEACH UNIT 1 - 1976 and 1977

Circulating Water System	System Code Not Applicable	Reactor Trip Systems Instrumentation	Reactor Core Fuel Elements	On Site Power System	Main Steam Supply System	Air Conditioning, Heating, Cooling, & Ventilating System
7-06-76(F)	8-06-76(0)(2)	11-30-76(F)(3)	12-22-76(F)	2-09-77(F)(5)	2-26-77(F)(5)	4-30-77(F)(5)
						5-28-77(F/M)(5,6)

Coolant Recirculation System	Emergency Generator System	Hangers, Supports Shock Suppressors
6-20-77(F)	6-29-77(F)	10-21-77(P)(5)
6-23-77(F)		

NOTES: FOR TABLE A-3

1. Component failure to meet technical specification requirement during a test.
2. Error in vendor safety analysis - licensee evaluated impact and determined that continued operation is acceptable.
3. Appears similar to power supply failure in event 1/10/76(c) under Engineered Safety Features Instrumentation Systems.
4. Appears to be a design error. Clearly causally linked to previous events in this category.
5. Discovered during routine test.
6. Appear to be causally related to 4/30/77(c) event in that the cause is generic.
7. Identical to 12/29/76 event as to component and cause.



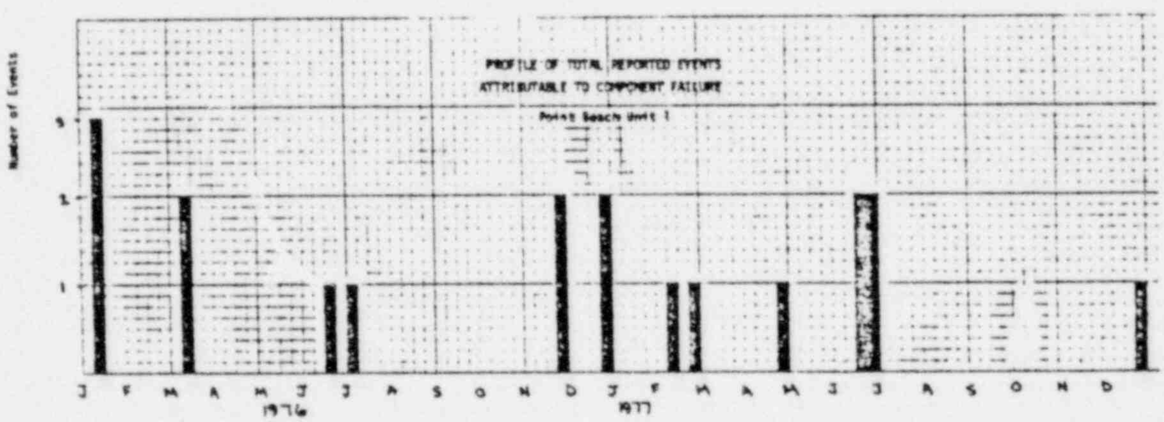


FIGURE A-10

Point Beach Unit 1 Performance Profiles

Regulator was cleaned and adjusted." On 4-28-77 an identical event occurred. The licensee identified the same cause but ordered a new regulator to replace the repaired regulator.

In summary, the reported events that appear to be causally linked are too few to suggest a pattern of deficient licensee performance. The limited total number of events both isolated and causally linked in the LER file suggests a pattern of facility operation virtually unimparied by management or personnel error. The patterns of management and personnel performance at Point Beach Unit 1 contrast sharply with those identified in other case studies.

Figure A-10 on the previous page shows the profile of total reported events due to human causes and the profile of events due to component failure.

Review of Inspection Reports and 766 System Data File for Point Beach Unit 1

When we reviewed the 766 system data file and associated inspection reports for 1976 and 1977, we found a total of 38 inspection reports detailing the results of NRC I&E inspector findings. Thirteen of these reports identify a total of 25 items of noncompliance. Nine of these 25 items involve physical protection and are identified in three separate inspection reports.

Matrix A-3 summarizes the findings of each inspection report and associated 766 system data file entries that identify noncompliances, as well as one report in which LERs were reviewed. Not including those noncompliances due to physical protection, ten noncompliances were assignable to ERC-M and six were assignable to ERC-P.

In general, there was strong agreement between the noncompliance cause code as listed in the 766 system and the detailed discussion in the "Report Details"

MATRIX A-3

Review of 766 File and Inspection Reports for Point Beach Unit 1

NAME POINT BEACH UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
76-06	FDP3	P	NO	NO	YES	NO	YES	NO		
	FDP3	M	YES	YES	YES	NO	NO	YES		
76-07 (Phys. Prot.)	RMA2	P	YES	YES	YES	NO	NO	YES	YES	
	RME2	P	YES	YES	YES	NO	NO	YES	YES	
76-08	NONE							-		2 EVENTS/AGREE
76-09	ASA2	M	YES	CAN'T TELL	YES	NO	NO	YES		
76-11	OAW3	M	YES	YES	YES	NO	NO	YES		2 EVENTS/AGREE
76-13 (Phys. Prot.)	RMC3	M	YES	YES	YES	NO	NO	CAN'T TELL	YES (2 ITEMS)	

NAME POINT BEACH UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
76-13	RLL2	P	YES	YES	YES	NO	NO	CAN'T TELL	YES (2 ITEMS)	
	RRA3	P	YES	YES	YES	NO	NO	CAN'T TELL	YES (2 ITEMS)	
76-15	FPE	P	YES	YES	YES	NO	NO	YES	YES (3 ITEMS)	
76-18	FPF	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER	NOT REVIEWED	
77-03	FCS2	P	YES	CAN'T TELL	YES	NO	YES	YES	YES	2 EVENTS/AGREE
77-09	FPG3	M	YES	YES	YES	NO	YES	IN SUBSEQUENT LETTER		2 EVENTS/AGREE
	FMY2	M	YES	CAN'T TELL	YES	NO	NO	IN SUBSEQUENT LETTER		
	FMY2	M	YES	YES	YES	NO	NO	NO		

NAME POINT BEACH UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated in IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed. Adequacy of Response (Disagree?)
77-09	ASB2	M	YES	YES	YES	NO	NO	YES		
77-13 (Phys. Prot.)	NED2	M	NO	NO	YES	NO	NO	IN SUBSEQUENT LETTER	YES (2 ITEMS)	
	NED2	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
	NEB3	P	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
	NDE3	M	YES	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
77-16	EMA2	P	NO	YES	YES	NO	NO	IN SUBSEQUENT LETTER		
	EEB2	M	YES	CAN'T TELL	YES	NO	NO	IN SUBSEQUENT LETTER		
	EJF2	M	YES	CAN'T TELL	YES	NO	NO	IN SUBSEQUENT LETTER		

NAME POINT BEACH UNIT 1

Insp. Rpt.	Non Comp.	Teknekron Cause Code	Does NC Cause Code in 766 Agree With IE Report	Does NC Cause Code in 766 Agree With 766 Text	Does 766 Text Agree With IE Report	Did N/C Result from Insp. Follow-Up On LER	Did N/C Result from Insp. Follow Up On a Licensee Identified Action	Has Licensee Specified Remedies to Preclude Recurrence as Stated iq IE Report	Licensee Action on Previously-Identified enforcement Items	LER's Reviewed Adequacy of Response (Disagree?)
77-17	FPG2	P	YES	CAN'T TELL.	YES	NO	NO	YES	YES (1 ITEM)	2 EVENTS/AGREE
77-19	FDJ2	M	YES	NO	YES	NO	NO	IN SUBSEQUENT LETTER		

Teknekron, Inc.

section of the inspection report. Less than 12 percent of the noncompliance cause codes either were ambiguous or did not agree with the associated inspection report details. The inspector's perception of the underlying cause of the noncompliance and his ability to communicate that perception in terms of the available cause codes (Primary Cause of Violation) listed in enclosure D of MC 0535 is readily apparent. In general, there was strong agreement between the enforcement text provided for each item of noncompliance identified in the 766 system and the "Enforcement Actions" section of the associated inspection report. There was less agreement between the noncompliance cause code in the 766 system and the 766 enforcement text: approximately 44 percent of the items bore either an ambiguous or irrelevant relationship to each other. This lower level of agreement was due largely to a lack of supporting detail in the 766 enforcement text. This lack of agreement between the noncompliance cause code and the 766 enforcement text means that a review of the 766 enforcement text and the noncompliance cause code without the supporting I&E report would not provide a sufficiently comprehensive understanding of the noncompliance and the circumstances of its origin.

We also reviewed possible sources of cues that may have aided inspectors in identifying noncompliance items. In no case did a noncompliance result from inspector followup on an LER. Only three noncompliances resulted from licensee identification of new or modified procedures to the inspector. In this case study, only about 12 percent of the noncompliances resulted from possible inspector cues; cues did not play a substantial role in identifying noncompliance items.

For 36 percent of the noncompliance items, licensee remedies to prevent recurrence of the event were specified in the inspection report, while forty-four percent of the noncompliance items were addressed in a subsequent letter. Generally, those items for which an immediate remedy was identified were those for which the licensee was in strong agreement with the inspector's findings.

The licensee's action on previously identified enforcement items was always tirely and complete at each inspector visit in which these items were reviewed. In reviewing LERs, the inspector never disagreed with the licensee's reporting. There were no events due to human failure that were serious from the regulatory point of view.

Figure A-11 shows the noncompliance profile for Point Beach Unit 1.

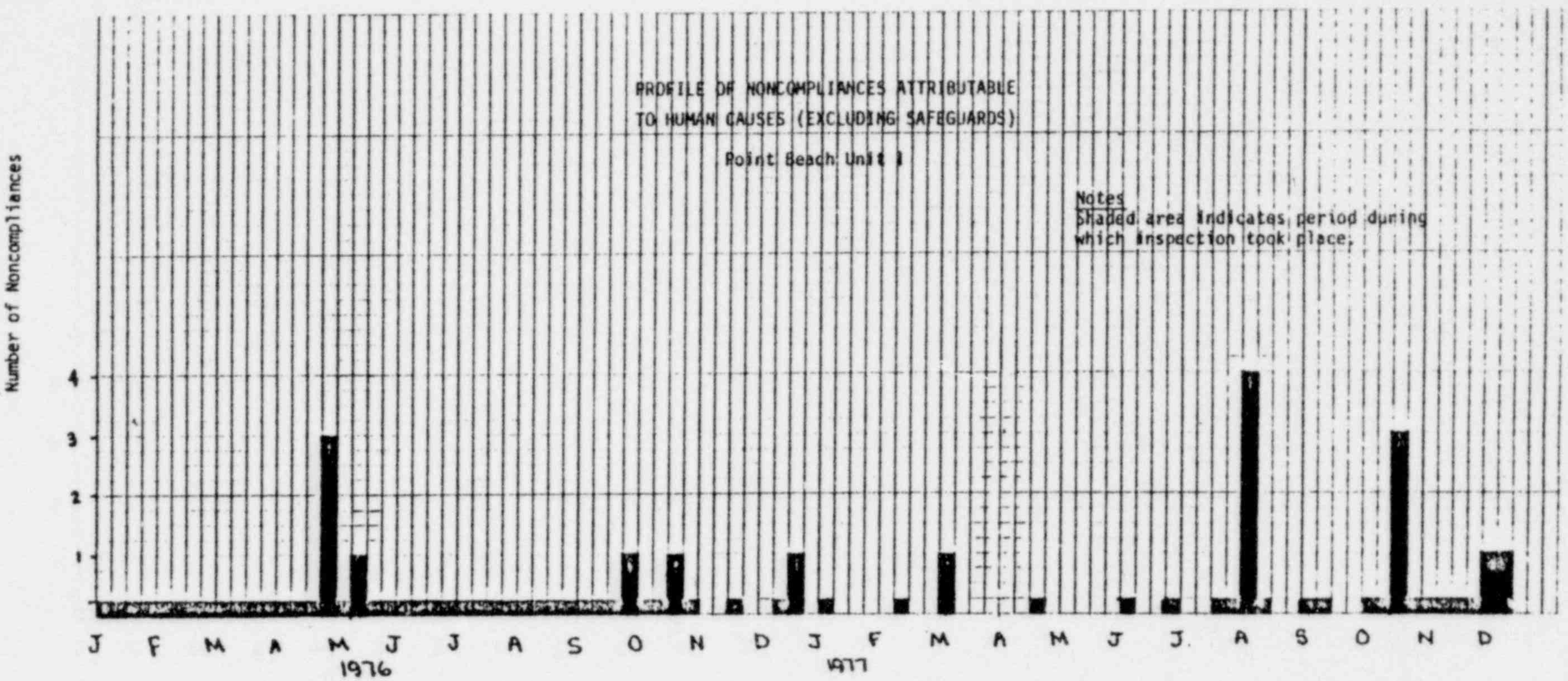

PROFILE OF NONCOMPLIANCES ATTRIBUTABLE
TO HUMAN CAUSES (EXCLUDING SAFEGUARDS)
Point Beach Unit 1
Notes
Shaded area indicates period during
which inspection took place.
Number of Noncompliances
4
3
2
1
J F M A M J J A S O N D
1976
J F M A M J J A S O N D
1977

FIGURE A-11

Conclusion

The FPM model and methodology, using existing LER and 766 file data, appear to have both the capacity and sensitivity to differentiate "poor" from "good" performers. Figure A-12 presents the profiles of total reported events attributable to human causes for the three licensees; the profiles for Prairie Island Unit 1 and Point Beach Unit 1, the "good" performers, are clearly different from that for Zion Unit 1. Figure A-13 shows the profiles of noncompliances (excluding safeguards) attributable to human causes, and again the differences are clear.

We found the LER file data essential to gaining insight into why the licensees perform as they do. As discussed in Sections 3.3.2.1 and 3.3.2.2, LERs promptly report real events occurring within facility systems. This close link to the "plant operating reality" offers the insight into management and personnel response to actual situations. The 766 file data was a less meaningful and sensitive performance indicator than we had anticipated at the start of our work. The cause codes in the data file are not precise and their use sometimes reflects inspectors' interpretations; the enforcement text is often too brief to establish the actual content of a noncompliance. Also, the discovery of noncompliances through the inspection program is often widely separated in time from their actual occurrence, due to the structuring of the program into time-dependent modules. These findings are discussed fully in Sections 3.3.3.1 and 3.3.3.2.

Differences in reporting requirements and technical specifications appeared to have little impact on the performance analysis results. We had expected little impact, since the FPM model is not inherently influenced by differences in technical specifications. But the empirical proof was in the performance profiles, as shown in Figure A-12. The LER performance profiles for Point Beach Unit 1 and Prairie Island Unit 1, with different technical specifications, were relatively similar to each other. Zion Unit 1 technical specifications are similar to those for Prairie Island Unit 1, but Zion's

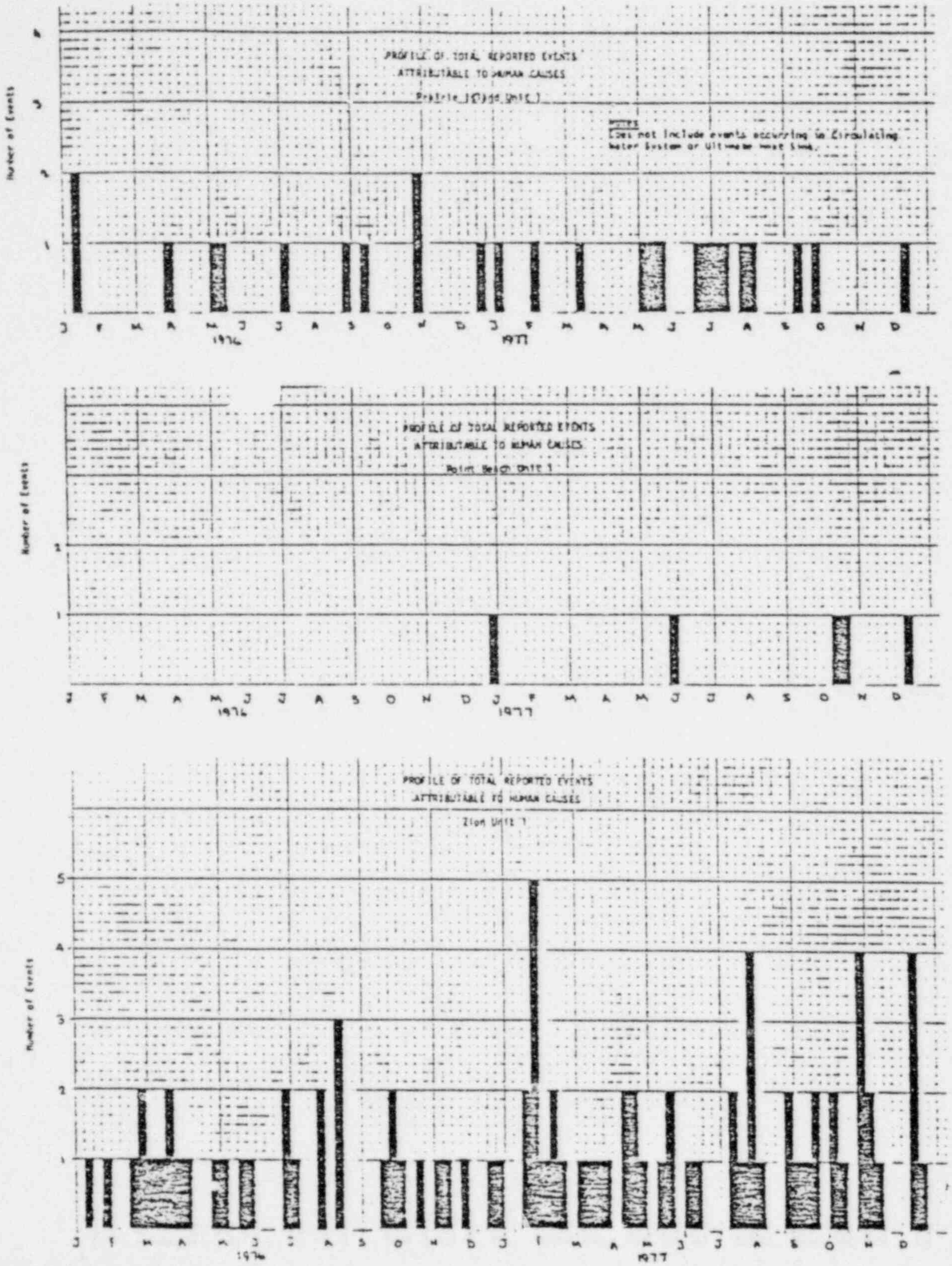


FIGURE A-12
Comparison of LER Profiles

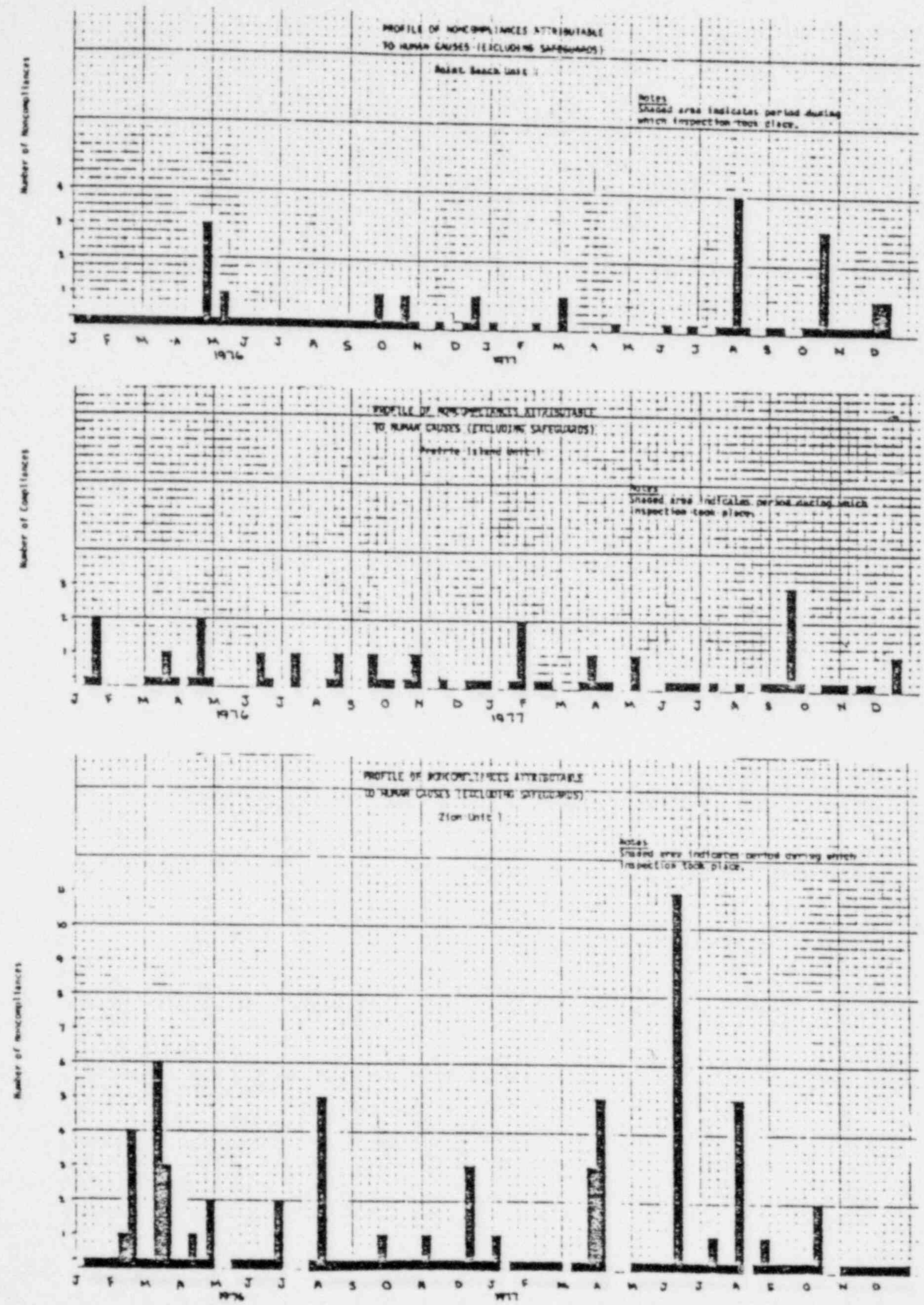


FIGURE A-13
Comparison of Noncompliance Profiles

LER profile is substantially different from both Prairie Island's and Point Beach's. Table 2, on page 38, establishes that technical specifications had little effect, at least for these three licensees. Further case studies will provide more indication of the sensitivity of the model to reporting and technical specification differences. We also expect that case studies of BWRs will permit comparisons that have until now been difficult.

Finally, we found that comparing the LER profile and noncompliance profile for a licensee provides insight into the capability and effectiveness of the regulatory process in managing the licensee's performance. This regulatory/licensee relationship may vary from region to region. Figure A-14 shows these profiles for Zion Unit 1: the differences in phasing and frequency between the LER and noncompliance profiles are apparent, and the LER profile continues to show high levels of human error. Figures A-15 and A-16 show the profiles for Point Beach Unit 1 and Prairie Island Unit 1, where phasing and frequency are more similar.

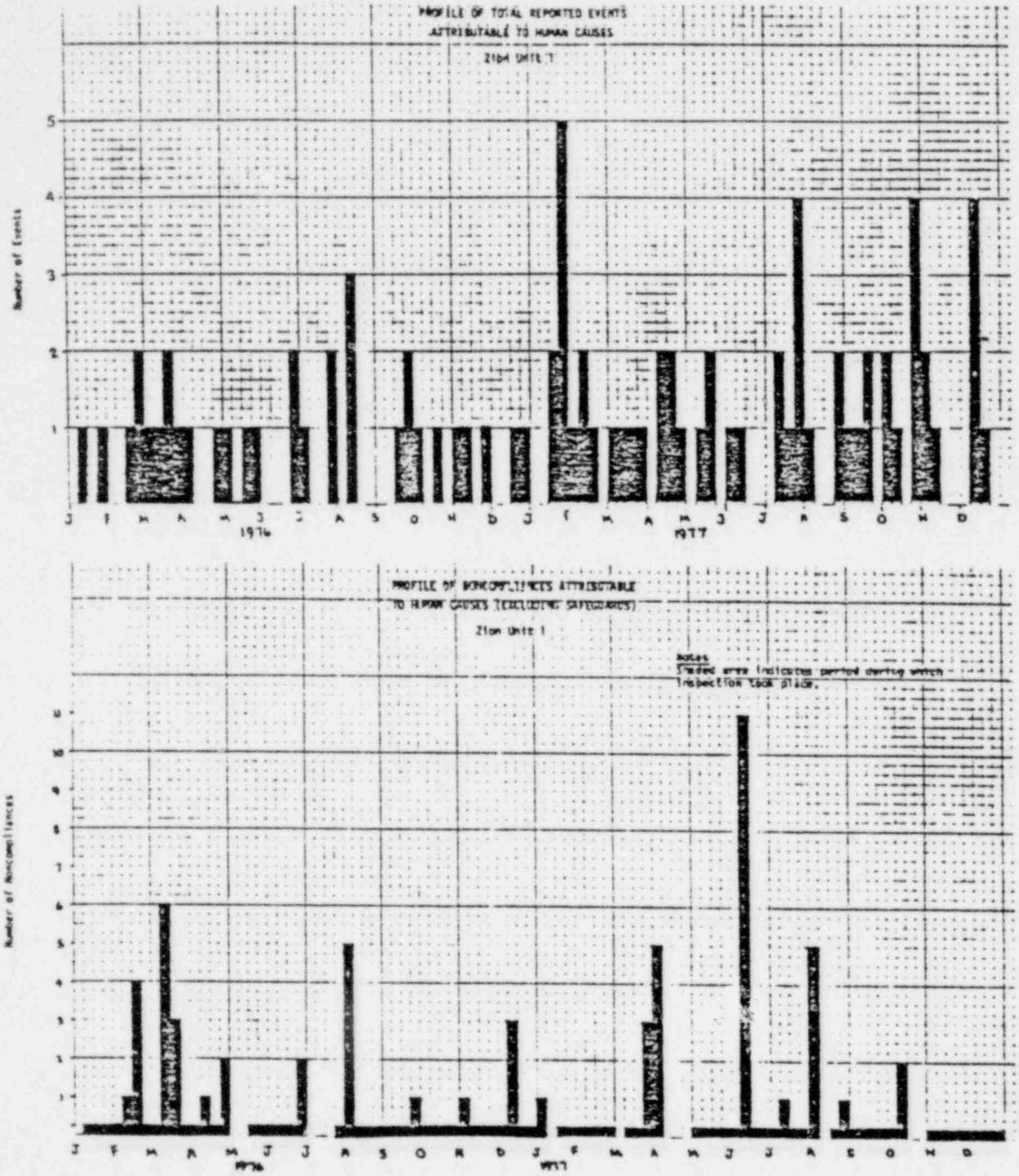


FIGURE A-14

Zion Unit 1 LER and Noncompliance Profiles

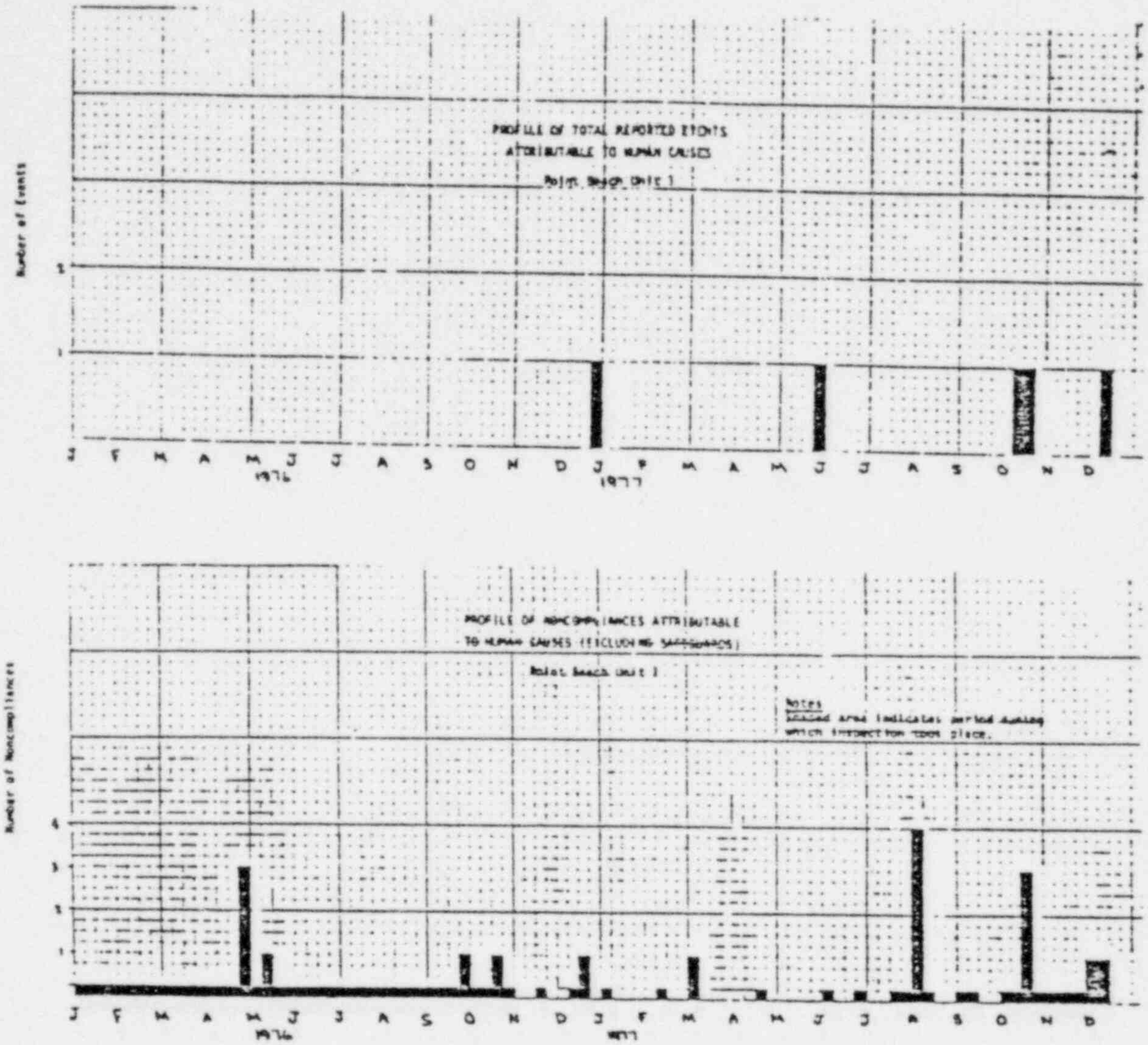
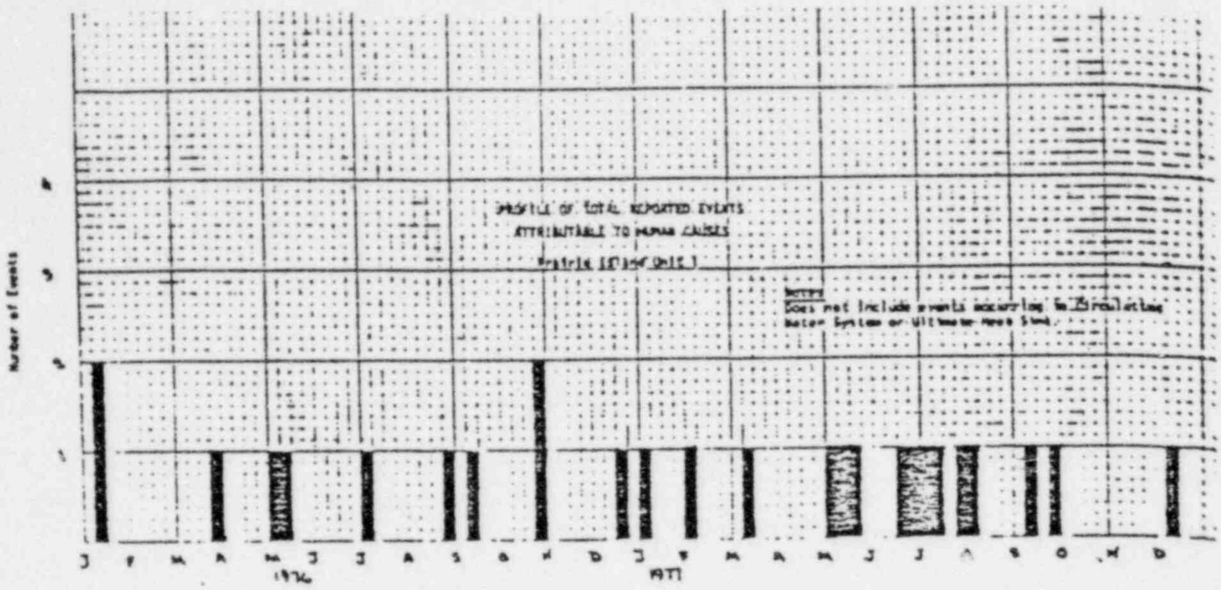


FIGURE A-15

Point Beach Unit 1 LER and Noncompliance Profiles



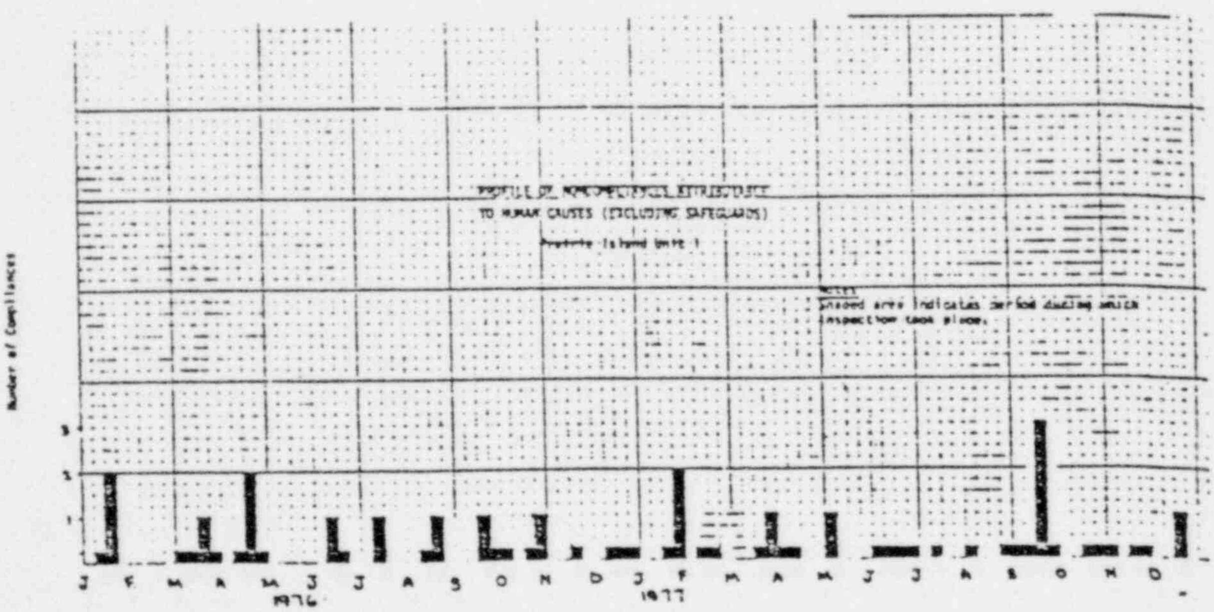


FIGURE A-16

Prairie Island Unit 1 LER and Noncompliance Profiles

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

DEC 1 4 1978

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Samuel E. Bryan, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION TO BE CONSIDERED FOR BOARD NOTIFICATION - LICENSEE REGULATORY PERFORMANCE EVALUATION (LRPE)

This supplements our memoranda dated November 6, 1978 and November 15, 1978 on the above subject. Enclosed is a letter which was sent to all recipients of the performance evaluation information. This letter transmitted a memorandum dated October 26, 1977 which contains additional information on this matter.

We request that copies of the enclosed documents be supplied to those who received the original LRPE information.

Samuel E. Bryan

Samuel E. Bryan, Executive Officer
for Operations Support, IE

Enclosures:
1. Transmittal Letter
2. Memo EMHoward to EVolgenau
   dtd 10/26/77

cc: J. G. Davis, IE w/o encls
H. D. Thornburg, IE w/o encls
N. C. Moseley, IE w/o encls
G. C. Gower, IE w/encls
IE Files w/encls

Dear

On November 15, 1978, I sent you information describing three approaches tried by the NRC staff for evaluating the regulatory performance of operating nuclear power plants. One of these approaches was called the "statistical method." It was based on evaluating two measures of performance - the number of noncompliance findings and the number of events considered directly controllable by the licensee. With the information sent you were the results of the statistical method evaluation for forty operating plants on sites based on data for 1976.

Enclosed with this letter is additional information developed in the trial of the statistical method. The enclosed information -- a copy of an October 26, 1977 memorandum entitled LICENSEE INSPECTION AND ENFORCEMENT INDICATORS UPDATE -- is based on noncompliance data (not event data) for the period January 1, 1976 through June 30, 1977 and for the period January 1, 1977 through June 30, 1977.

As was discussed in the information previously sent you, the purpose of licensee regulatory performance evaluation, if successful, is to give NRC staff the ability to distinguish between levels of licensee regulatory performance. This could lead to more effective use of the agency's inspection and enforcement resources and to identification of plants that need further examination by the agency.

Sincerely,

Enclosures:
As Stated

October 2?, ??77

MEMORANDUM FOR: Ernst Volgenau, Director
Office of Inspection & Enforcement, HQ

FROM: E. Morris Howard, Director, Region IV, IE

SUBJECT: LICENSEE INSPECTION AND ENFORCEMENT INDICATORS UPDATE

Enclosed are four figures depicting inspection and enforcement indicators, based solely on noncompliance, covering the periods January 1, 1976 through June 30, 1977 and January 1, 1977 through June 30, 1977, for both BWR's and PWR's.

It is interesting to note that unusually low or high indicators in the long term (January 1, 1976 through June 30, 1977) are not off-set by drastically improved performance in the short term. Indian Point is an excellent example of short term improvement with the long term record continuing to reflect the unusually bad record in Calendar Year 1976. The long run trend is a valuable tool in determining the improvement or degradation of a site's record when compared with a short term evaluation. These trends might also be used to determine the effect of significant enforcement action, which is what occurred at Indian Point and Zion in the second-half of 1976, with Indian Point showing marked improvement in the first-half of 1977 and Zion showing a marked down trend in the same period. It will be interesting to determine the effect of the IE enforcement activities on Zion's record in subsequent evaluations.

There are unlimited possibilities which could be investigated with a strong possibility that reasonable, statistically supportable, conclusions could be reached concerning the licensee's activities, program effectiveness, and regional inspection performance.

There may also be a hint as to how the inspectors perceive the licensee management attitudes, particularly where there is a low subjective rating and only a single deficiency for over four hundred hours of inspection effort. The converse also occurred.

It is possible to draw conclusions over the individual functional areas, most clearly by inference from the total score for each functional area.

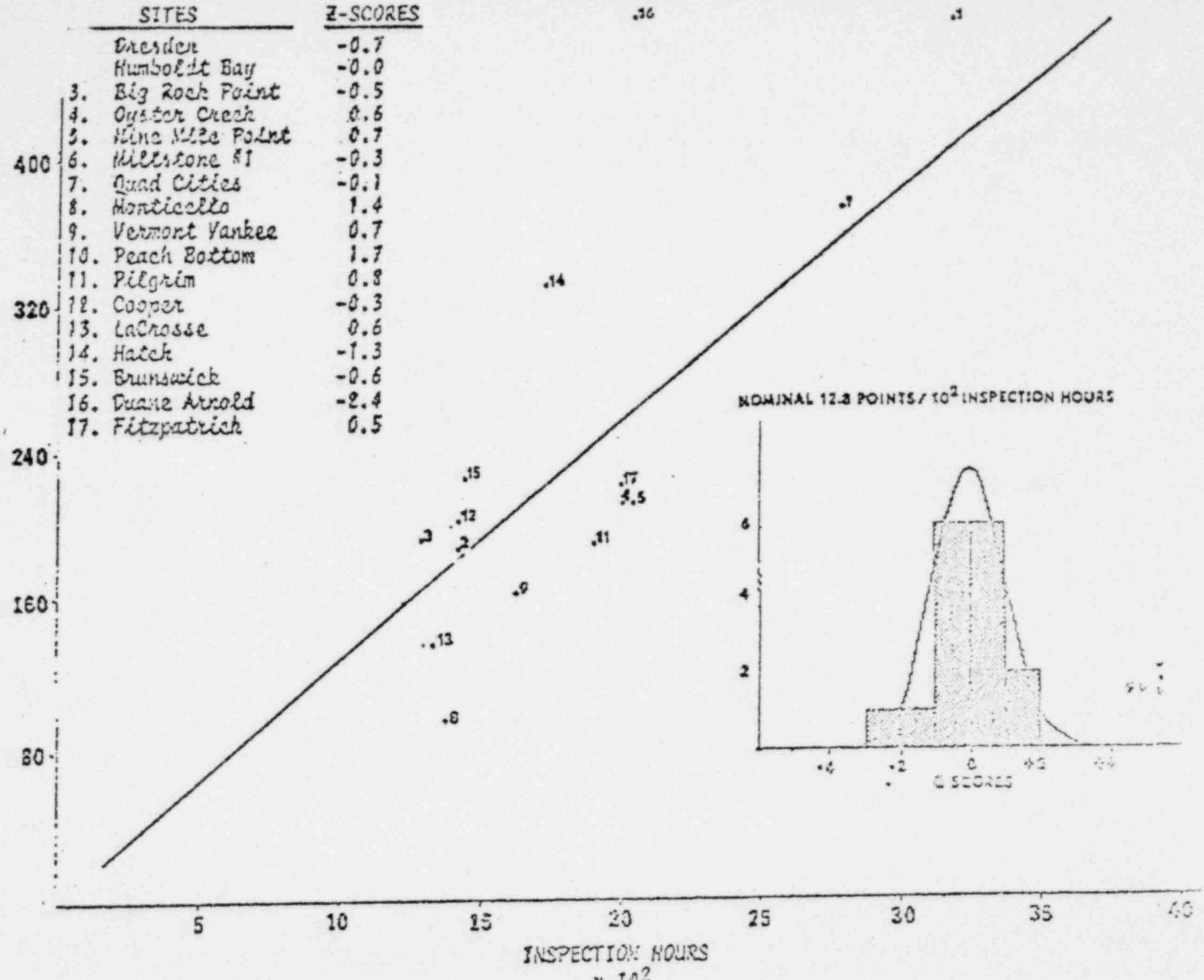
I will be glad to discuss any or all of these areas at your convenience.

Original Signed By
E. Morris Howard

E. Morris Howard
Director

Enclosures:
1. Boiling Water Reactors - January 1, 1976 Thru June 30, 1977
2. Boiling Water Reactors - First Half 1977
3. Pressurized Water Reactors - January 1, 1976 Thru June 30, 1977
4. Pressurized Water Reactors - First Half 1977

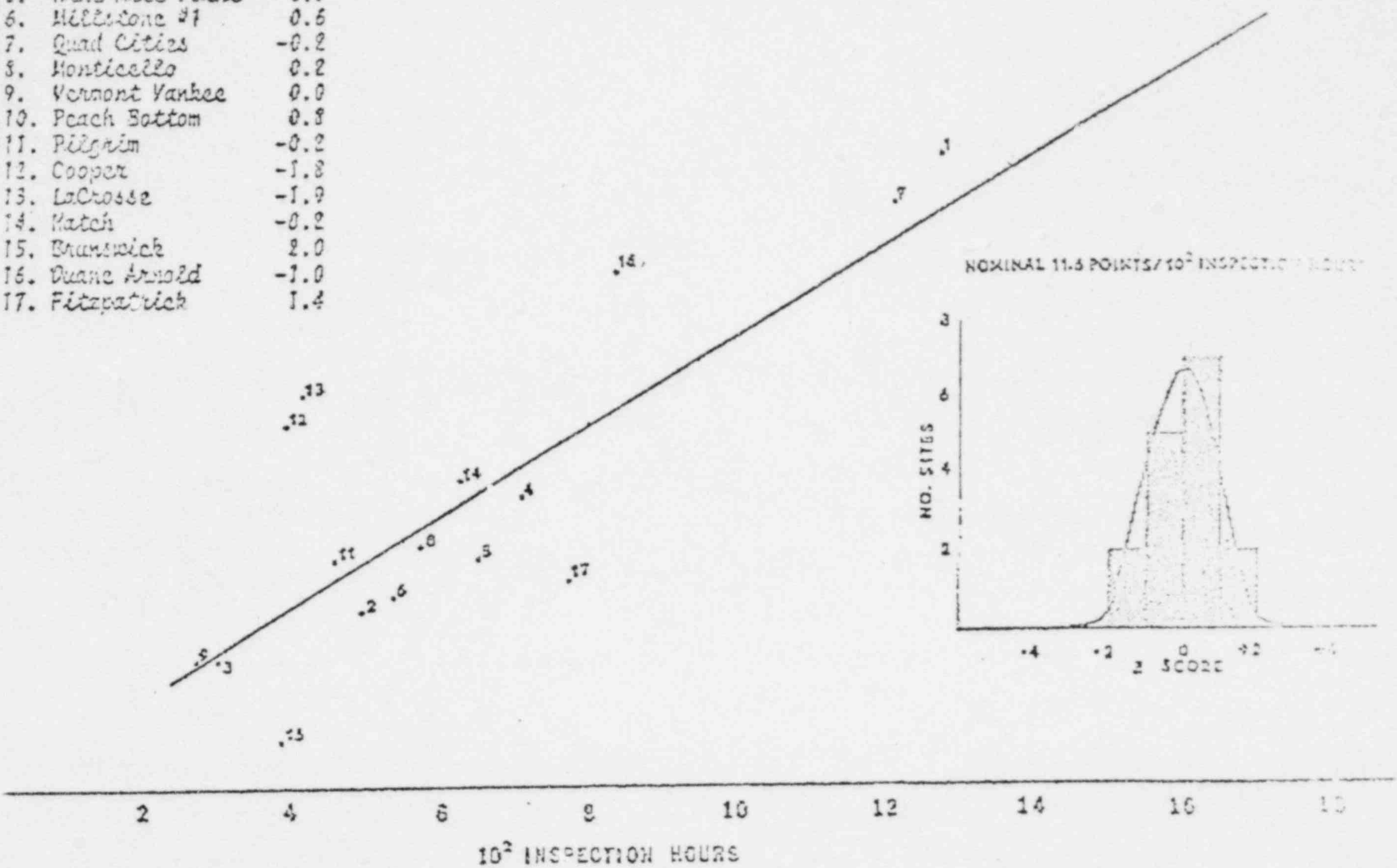
cc: J. G. Davis, IE:HQ, w/enclosures



SITES	Z-SCORES
Dresden	-0.7
Humboldt Bay	-0.0
3. Big Rock Point	-0.5
4. Oyster Creek	0.6
5. Nine Mile Point	0.7
6. Millstone #1	-0.3
7. Quad Cities	-0.1
8. Monticello	1.4
9. Vermont Yankee	0.7
10. Peach Bottom	1.7
11. Pilgrim	0.8
12. Cooper	-0.3
13. LaCrosse	0.6
14. Hatch	-1.3
15. Brunswick	-0.6
16. Duane Arnold	-2.4
17. Fitzpatrick	0.5

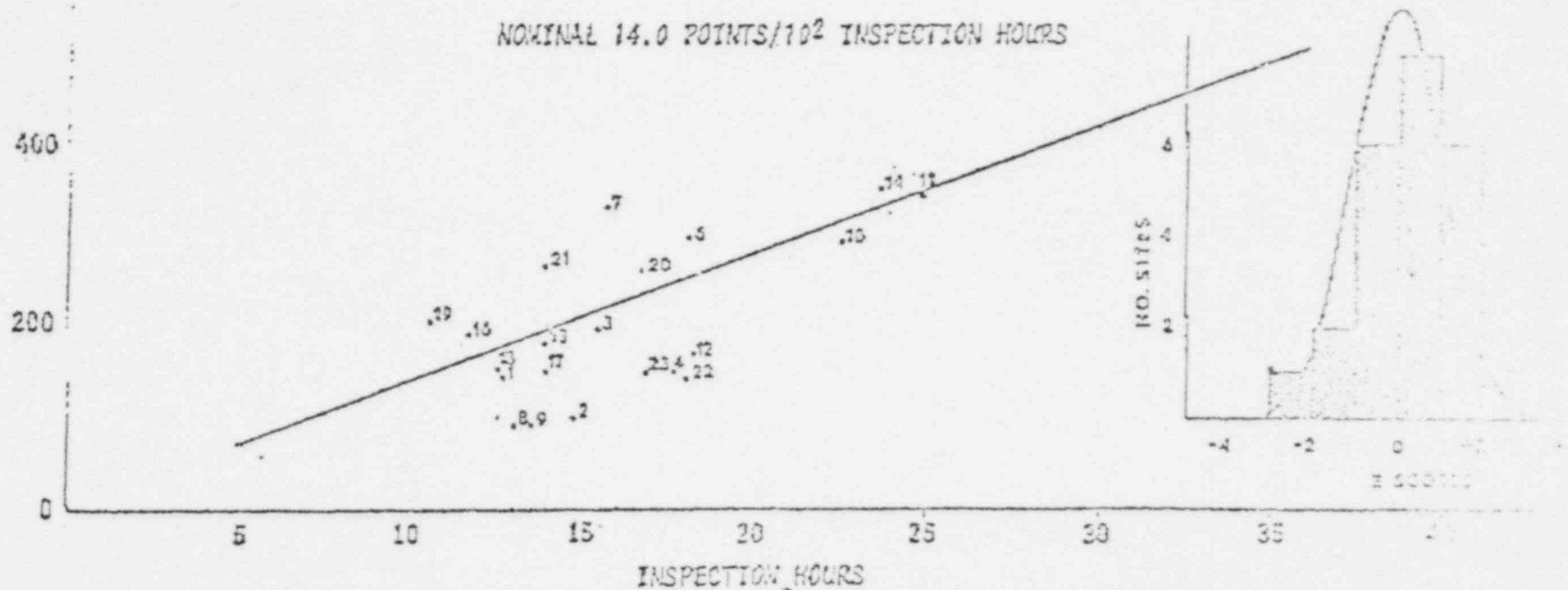
BOILING WATER REACTORS - JANUARY 1, 1976 THRU JUNE 30, 1977

	SITES	Z-SCORES
1.	D[illegible]n	-0.3
2.	Hu[illegible]dt	0.6
3.	Big Rock	0.1
4.	Oyster Creek	0.3
5.	Nine Mile Point	0.7
6.	Millstone #1	0.6
7.	Quad Cities	-0.2
8.	Monticello	0.2
9.	Vermont Yankee	0.0
10.	Peach Bottom	0.8
11.	Pilgrim	-0.2
12.	Cooper	-1.8
13.	LaCrosse	-1.9
14.	Hatch	-0.2
15.	Brunswick	2.0
16.	Duane Arnold	-1.0
17.	Fitzpatrick	1.4

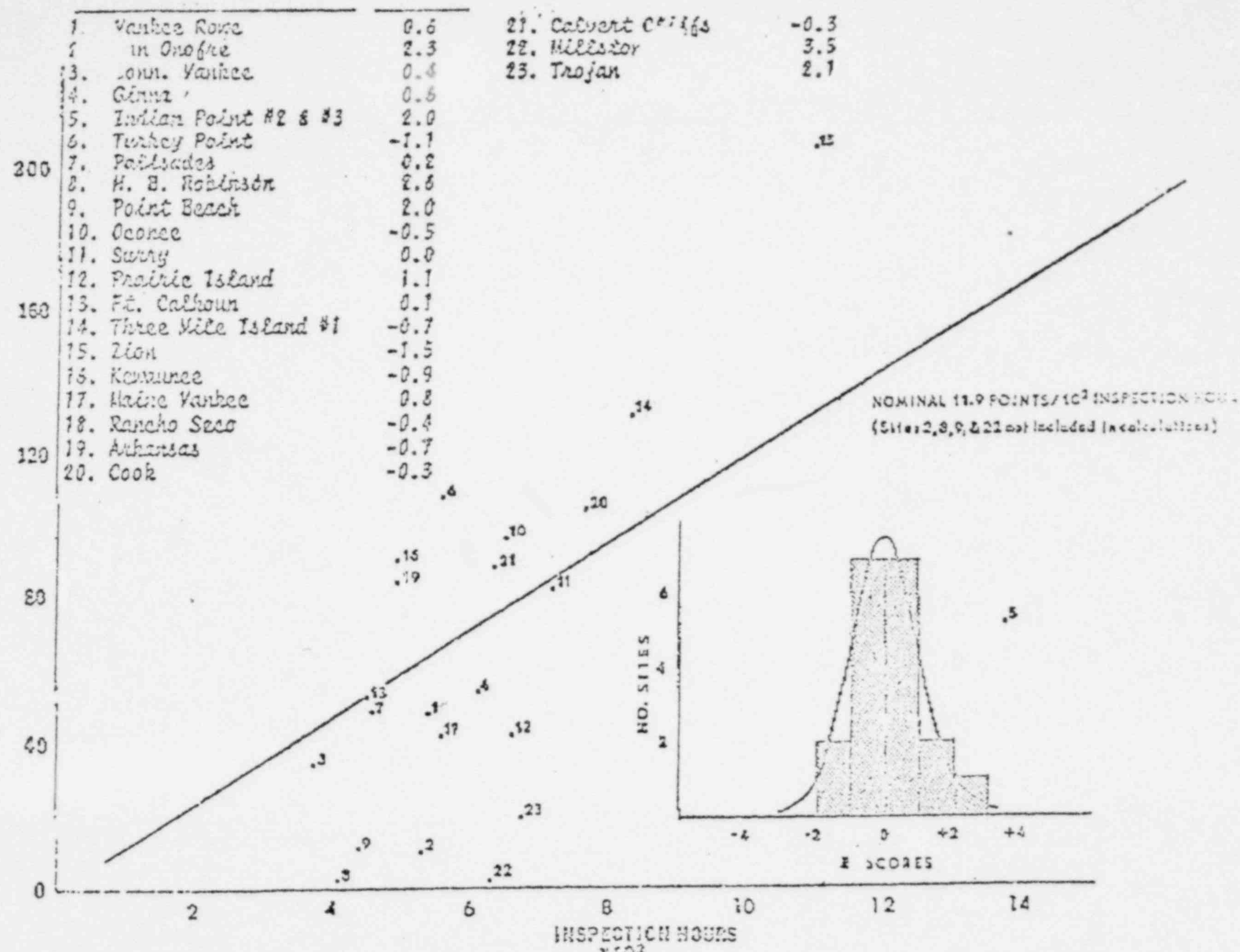


BOILING WATER REACTORS - FIRST HALF 1977

	Yankee Rowe	0.4	20.	Cook	-0.3
	San Onofre	1.6	21.	Calv Cliffs	-0.3
	Conn. Yankee	0.2	22.	Mill #2	1.4
4.	Ginna	1.3	23.	Trojan	1.9
5.	Indian Point #2 & #3	-2.2			
6.	Turkey Point	-0.5			
7.	Palisades	-1.8			
8.	H. B. Robinson	1.4			
9.	Point Beach	1.5			
10.	Oconee	0.3			
11.	Surry	0.1			
12.	Prairie Island	1.1			
13.	Ft. Calhoun	0.2			
14.	Three Mile Island #1	-0.2			
15.	Zion	-1.5			
16.	Kewaunee	-0.4			
17.	Maine Yankee	0.6			
18.	Rancho Seco	0.4			
19.	Arkansas	-0.7			



PRESSURIZED WATER REACTORS - JANUARY 1, 1976 THRU JUNE 30, 1977



PRESSURIZED WATER REACTORS - FIRST HALF 1977