Limiting Factor Analysis of High-Availability Nuclear Plants

NP-1138
Research Project 894-3

Final Report, September 1979

Prepared by

BABCOCK & WILCOX
Nuclear Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

Principal Investigator
R. H. Clark

and

DUKE POWER COMPANY
P.O. Box 33189
Charlotte, North Carolina 28482

Principal Investigator
W. E. Barrow

Prepared for

Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, California 94304

EPRI Project Manager
R. E. Swanson
Nuclear Power Division

ORDERING INFORMATION

Requests for copies of this report should be directed to Research Reports Center (RRC), Box 50490, Palo Alto, CA 94303, (415) 961-9043. There is no charge for reports requested by EPRI member utilities and affiliates, contributing nonmembers, U.S. utility associations, U.S. government agencies (federal, state, and local), media, and foreign organizations with which EPRI has an information exchange agreement. On request, RRC will send a catalog of EPRI reports.

NOTICE

Prepared by
Babcock & Wilcox
Lynchburg, Virginia
and
Duke Power Company
Charlotte, North Carolina

EPRI PERSPECTIVE

PROJECT DESCRIPTION

This report presents the findings and recommendations of one of four studies by EPRI to define the factors that limit the availability of nuclear power plants. As such, it should be of interest and help to utility personnel involved in plant engineering, operation and maintenance and responsible for improving plant availability.

PROJECT OBJECTIVES

In this study of a Babcock and Wilcox plant, as in the other three studies, a mature, high-availability plant was selected. The reason for this was to prevent obscuring the results with one-of-a-kind failures, major selection problems that had already been solved or were on the way to resolution, and problems associated with unusually severe, and therefore atypical, break-in periods. The findings from the plant being investigated were modified by experiences at a reference plant to prevent the findings from being excessively plant specific. The results from similar studies on Westinghouse, Combustion Engineering, and General Electric plants are given in NP-1139, NP-1137, and NP-1136 respectively.

PROJECT RESULTS

The methodology used in the study was to organize a team comprised of representatives of the nuclear steam supplier, the architect-engineer and the operating utility, who observed plant events, maintenance and outage records. These observations became the basis for problem area identification, prioritization and--depending on the nature of the problem--the possible need for research and development.

In using this report, the following should be noted:

- The report was not intended to address or judge in any way utility management matters or regulatory requirements.

- Operation of any power plant is based on system and economic needs and considerations that do not necessarily dictate the achievement of maximum unit productivity.

- Nuclear plant shutdown schedules are complex and are frequently impacted severely by unanticipated events. The consequences can sometimes be mitigated by judicious contingency planning.

- The study results are not component or system reliability oriented; i.e., since the scope of the study did not go significantly beyond a single plant, meaningful failure rate information cannot be derived.

- The report presents guidelines that in many cases are plant specific. Different equipment, available interfaces, procedures, and regulatory requirements will have marked effects on each plant's performance.

Roy Swanson, Project Manager
Nuclear Power Division

FOREWORD

The reader should use this report in the context in which it was prepared and written. The report concentrates upon a particular nuclear steam suppliers plant and reviews its outage and maintenance records. While comparisons have been made with some other plants by the same NSS, the report still in major part represents the findings connected with the studied plant. As such, it should not be interpreted as a reliability document. The sampling of data is too small for a reliability base and the appearance of a large number of a particular vendors product in the tables may only indicate that the original equipment purchases were placed in a major part with that vendor.

Plant methods and corrections are included so that other utilities may benefit from the actions taken by an experienced plant staff. The study did not have the staff or funding to determine other or all utility actions in similar circumstances. It is our opinion that the information presented will provide valuable information to other utilities since it is concentrated on a reasonably high performance experienced plant. As other plants mature through the check-out/break-in period, it is expected that they will find their performance limited by the same or similar generic basic problems identified in these reports.

It is our intent to utilize this report to assist in determining priorities and needs for R&D efforts. It represents an effort by a team composed of NSS, AE and utility personnel and provides a good insight into the operation limitations which will probably have the most long term effects on utility plant performance capabilities.

ABSTRACT

An Electric Power Research Institute (EPRI) -sponsored study to identify availability limiting factors in plant having nuclear steam supply systems supplied by Babcock & Wilcox (B&W) is described. Oconee Nuclear Unit 1, owned and operated by Duke Power Company, was the reference study plant. The study was conducted by a team from B&W representing the nuclear steam system supplier and from Duke representing the owner/operator and architect-engineer.

The operating and maintenance records from Oconee Unit 1 were collected for the period July 1, 1974, through December 31, 1977. These data were identified as historical data. During the data collecting phase of the study (January 1 through December 31, 1977), onsite team members obtained, from plant records and personal interviews, additional availability-related information on plant outages from Oconee 1 and also from Oconee Units 2 and 3. These data were identified as current data. The Oconee data were supplemented with similar but less detailed data from the Sacramento Municipal Utility District's Rancho Seco plant and to a lesser degree with data from the Metropolitan Edison Three Mile Island 1 (TMI-1) plant. At both Oconee 1 and Rancho Seco, team members observed and obtained data on availability-related activities during the 1977 refueling outages. Documented records of 1977 refueling outages at Oconee 3 and TMI-1 and of a "B&W projected Standard" refueling outage were also included as part of the study. As a supplementary study, data were obtained and analyzed on 17 valves that have had a history of impacting plant availability. Finally, as an effort to avoid misleading conclusions due to a study of data from a limited sampling and with limited details, additional data were obtained by interviews with engineers, operators, and maintenance personnel.

The operation and maintenance data were assigned to one of 48 systems/components; refueling data were assigned to one of 17 work events. Limiting factors for operation, maintenance, and refueling were calculated. The formula used for calculating the limiting factor for operation considered the number of events, the loss of power, and the time to repair. The latter included indirectly related time, such as access times, acceptance times, delays, etc. Approximate formulas

were also developed to calculate limiting factors for maintenance and for refueling. Each of the systems/components was ranked according to its calculated importance. Discussions were included for each system, refueling work event, and each of the 17 key valves.

Combined equipment limiting factors based on the current operational, historical operational, and refueling limiting factors were determined for each system/component and work event.

Finally, the report recommends steps that could reduce the impact of the availability limiting factors.

CONTENTS

ILLUSTRATIONS

TABLES

SUMMARY

LIMITING FACTOR ANALYSIS OF HIGH AVAILABILITY NUCLEAR PLANTS

B&W, as the NSS supplier, and Duke Power Company, in the dual role of both architect-engineer and owner/operator, performed this study of factors that limited the availability of the Oconee Nuclear Station power plants.

After project team organization, this study pursued the following EPRI-prescribed scope of work:

1. Identify limiting factors, from collected data, that prevent better availability.
2. Categorize as to the root cause of unavailbility into groupings established by EPRI before the study began.
3. Perform added plant comparisons as a check of limiting factor validity.
4. Evaluate current programs which aim to alleviate limiting factors.
5. Evaluate selected top limiting factors in depth.

Identification of limiting factors was based primarily on written records, both published and unpublished, including plant operating records at the plant sites. Included were components and activities that had actually caused power reduction and also those whose failure, malfunction, or maintenance (1) could have caused a power reduction, (2) have or could have caused extension of a refueling/maintenance outage, (3) have or could have caused workers to receive high occupational radiation doses, and (4) whose maintenance was deemed to be excessive.

Conclusions and recommendations are based primarily on data from Oconee 1 for the period July 1, 1974 through December 31, 1977, and from Oconee 2 for calendar year 1977.

The prescribed categories by which limiting factors were grouped according to root causes were as follows:

1. Utility management policies, operating philosophy, and maintenance practices.
2. Regulatory requirements.
3. Design requirements.

Design was the only category to be further investigated. Design was defined as broadly inclusive so that it included factors not strictly within the other more narrowly defined categories. As the study proceeded, it became apparent that the written records were frequently not adequate to determine a root cause. Often, records contained a proximate, apparent cause although a series of cause and effect relationships was involved. Analysis revealed that elements of more than one of the categories — many times all four — were root causes to limiting factors; i.e., the predefined categories were not mutually exclusive.

Identification and analysis of the limiting factors shows the following:

- Seventy-six percent of the outages involved five components: steam generators, fuel handling equipment, control rod drives, reactor coolant pumps and motors. The balance of the outages were due to many other components/systems, each with a low impact on availability.
- The refueling/maintenance outage was significant as an availability-limiting activity.
- Most limiting factors have multiple causes, e.g., operational utility management, regulatory requirements, and/or design requirements.
- Improved maintenance practices could reduce equipment and refueling outages. Cost-effective areas of improvement include planning, spare parts stockage, procedures, tools, and training.

For additional plant comparisons, data were obtained from Rancho Seco and Three Mile Island Unit 1. Beyond that, comparisons were made with B&W Equipment Outage Factor (EOF) records for six plants, including Oconee 1, 2, and 3, with the same basic NSS design. This comparison showed the following:

- Limiting factor and EOF data correlate; i.e., the same items appear in approximately the same order.
- Essentially all steam generator problems occurred at the Oconee plants.
- The balance-of-plant problems were less severe at the Oconee units than at the others. This, however, was heavily influenced by turbine problems at Arkansas Nuclear One (Unit 1) and Rancho Seco.
- The NSS problems (excluding the steam generator) were more severe at the Oconee units than at the other plants.

- In both sets of data and except for steam generator losses at Oconee, the refueling outage activities are by far the largest single contributor to loss of plant availability.

- The EOF comparisons show significant differences in plant availability among the different plants. These differences appear to be due not only to low-frequency, high-impact events such as turbine problems but also to plant experience, operating practices, balance-of-plant (BOP) design and performance, training, spare parts inventories, and maintenance practices. These comparisons demonstrate that the limiting factors identified in this study apply to the particular plants covered by this project. Care should be used in generalizing the results from this study to other plants.

Programs to alleviate design category limiting factors are given. Each limiting factor includes proposed resolutions, and the resolution is categorized as to whether a remedy is ready to be implemented, is known but not developed, or is in need of further study before development.

A further analysis of the principal limiting factors was performed. Also included are the results of a study of 17 key valves that had been identified as impacting plant availability. The principal conclusions of this study are as follows:

1. Availability improvement is possible at the plants studied. Intensive effort on a relatively few problem areas could produce a favorable return on investment. Availability improvement could result from both improved equipment and improved maintenance practices.

2. Efforts should continue to reduce refueling outage span times. This should include modifications to fuel handling equipment, better training, and better checkout. Since refueling will be performed many times over the plant life, improvement could significantly reduce total outage time.

3. Studies such as this are a sound and useful step to improved availability because, in contrast to earlier efforts, this study covered the entire plant, extended over a representative period, and included in-depth analyses.

4. The results revealed that most availability limiting factors have multiple causes. In-depth analysis is often necessary to identify these causes. Efforts to improve plant outage data at the source to understand the causes of plant outages should continue. The data indicate that it is possible to be selective by concentrating the effort on a relatively few items of equipment which cause most of the outages.

Section 1

BACKGROUND AND INTRODUCTION

BACKGROUND

The availability of a power plant to produce electricity is important to the utility company's system reliability and economic operations. Even small im-improvements in availability can result in a significant reduction in the need for high-cost replacement power and can lead to a savings in reserve capacity requirements. Sustained improvement in plant availability may delay the need for large capital outlays for new plant construction.

Recognizing the importance of plant availability, the Electric Power Research Institute (EPRI) sponsored this study, which analyzes the factors that may limit the availability of electrical generating plants having nuclear steam systems (NSSs) supplied by The Babcock & Wilcox Company (B&W). The Oconee Nuclear Station, Unit 1 (Oconee 1) plant, owned and operated by Duke Power Company (Duke), was selected as the reference plant for this study. The study was conducted by a project team comprising representatives from B&W as the NSS supplier and Duke as both the owner/operator and the architect-engineer of the reference plant. Activities were performed under the overall direction of the EPRI Project Manager.

PROGRAM OBJECTIVES

The objectives of this program were to identify specific factors that have limited the availability of certain nuclear power plants with B&W-supplied NSSs and to assess the extent of their impact on plant performance and to suggest what may be done to improve future designs or modify existing plants to improve performance. Another objective was to supply a data base and analyses that could serve as a focal point for future R&D projects to improve plant reliability and availability.

We have met these objectives by identifying the systems/components that have significtnaly impacted plant availability in the plants cooperating in the study by quantifying the impact with numerical values, by identifying the root causes (where possible) of the limitations, by drawing conclusions from the data, and by making suggestions to reduce the plant availability impact of these limiting factors. A similar treatment was given to those systems/components identified as requiring high maintenance or repair even though their impact on plant availability may be only indirect.

A separate study of refueling outage activities was included to identify factors that have caused such outages to take longer than normally scheduled or expected. Special emphasis was also given to selected key valves identified as having significant impact on plant availability.

This EPRI availability study was intended to be the first phase of a series of availability improvement programs that would improve plant operating performance and reduce the amount of time and manhours to perform maintenance. These reductions translate into increased plant reliability and availability and less personnel radiation exposure.

DEFINITIONS OF BASIC TERMS

Appendix I defines terms used in this report. However, certain of these terms are basic to the limiting factor identification and analysis. To facilitate early reader use of this information, definitions of these terms are repeated below.

Availability — The amount of time the plant was available for power production, represented as a percentage of the time the plant could be available.

Combined Equipment-Limiting Factor (CELF) — The loss of plant availability in effective full-power hours (EFPH) per unit-year for a given system/component. The CELF is a normalized, one-reactor unit-averaged value which includes limiting factor for operation (LFO) "historical" and "current" data, and the actual outage extension portion of the limiting factor for refueling (LFR) for those systems/components which are not directly a part of the refueling activity. It is the single figure-of-merit factor for availability evaluation. The CELF is determined from the following formula:

$$\text{CELF} = \frac{\text{LFO}_H + \text{LFO}_C}{2} + \text{refueling outage extension}$$

Refer to Appendix C for definition of terms.

Current Data — Non-refueling data on Oconee Units 1, 2, and 3 for 1977.

Failure — Termination of the ability of an item to perform its required function. Failures may be unannounced and not detected until the next test (unannounced failure), or they may be announced and detected by any number of methods at the instant of occurrence (announced failure).

Historical Data — Data collected on Oconee Unit 1 from 7/1/74 through 12/31/77.

Limiting Factor for Maintenance (LFM) — The manhours of labor for maintenance or repair per unit-year for a given system or component. The LFM is determined for the Oconee 1, 2, and 3 current data from the formula

$$\text{LFM} = \sum_{\substack{\text{No. of} \\ \text{events}}} (\text{No. of men} \times \text{MTTR}) \frac{1}{3} \qquad \text{(mh/unit-year)}$$

Refer to Appendix C for definition of terms.

Limiting Factor for Operation (LFO) — The loss of plant availability in EFPH per unit-year due to failure or malfunction of a given system or component. This factor includes power losses (EFPH) due to reactor shutdown and startup and component access as well as the power losses during actual maintenance or repair work. LFO is determined for the Oconee 1, 2, and 3 current data from the following formula:

$$\sum_{\substack{\text{No. of} \\ \text{events}}} \left[\begin{matrix}\text{power loss} \\ \text{factor}\end{matrix} \times \left(\text{MTTR} + \begin{matrix}\text{additional} \\ \text{outage time}\end{matrix}\right)\right]\frac{1}{3} \qquad \text{(EFPH/unit-year)}$$

Refer to Appendix C for definition of terms.

Limiting Factor for Refueling (LFR) — The difference between the actual time (clock hours) to perform a given refueling outage activity and the B&W-projected standard time to perform that activity. LFRs have also been determined for certain components that undergo maintenance during the refueling activities. The LFR is expressed in terms of effective full-power hours (EFPH). LFR is determined from the following formula:

$$\text{LFR} = (P - S)F_p \qquad \text{(EFPH/unit-year)}$$

Refer to Appendix C for definition of terms.

Reliability — The characteristic of an item expressed by the probability that it will perform a required mission under stated conditions for a stated mission time.

Section 2

STUDY METHODOLOGY

EPRI STATEMENT OF WORK

The EPRI Statement of Work sets forth objectives, organizational guidelines, and data gathering and analysis criteria regarding nuclear plant availability limitations. The work is identified by the phases listed below.

I. Project Organization — Formulation of detailed work plans and procedures for project organization.

II. Limiting Factors Identification — Evaluation of Oconee 1 operational history to identify causes of outages and power reductions.

III. Limiting Factors Analysis — Analysis of limiting factors by categorization, priorities, and evaluation.

IV. Additional Plant Comparisons — Investigation of operating histories and design features of at least two additional B&W units to test the conclusions resulting from analysis of the Oconee 1 data.

V. Evaluation of Current Programs — Identification of activities to reduce or eliminate each limiting factor on the Phase III priority list.

VI. In-Depth Analysis of Limiting Factors — In-depth analysis of selected limiting factors from the Phase III priority list.

VII. Final Technical Report — Preparation of a report describing the project and the results derived.

The following paragraphs explain certain phases as noted.

PHASE II — LIMITING FACTORS IDENTIFICATION

Sources of Data

The project team collected Oconee 1 operations and maintenance records for the time period July 1, 1974, through December 31, 1977, and compiled data from these records. In this report, these data are referred to as "historical data." 1977 operations and maintenance data from Oconee 2 and 3 were also collected to provide comparative information. The 1977 non-refueling data for the three Oconee units are referred to as "current data" in this report.

The project team observed and took notes on the work activities on or near the critical path during the 1977 Oconee 1 refueling outage. These data were supplemented by observations of selected operations during the 1977 refueling of Rancho Seco, the Sacramento Municipal Utility District (SMUD) plant. Records of both plants were studied for additional data, as were the 1977 refueling activity records for Three Mile Island Unit 1 (TMI-1) and Oconee 3. Collectively, these data are called "refueling outage data."

Both public (published) and private (unpublished or verbal) sources of information were searched to identify factors that have limited the availability of commercial power plants with B&W-supplied NSSs. The data sources are identified in Appendix B. Data were collected on those components and/or activities:

- Whose failure or malfunction caused or could have caused a plant shutdown or power reduction.
- Whose failure or malfunction extended or could have extended a plant shutdown or power reduction.
- Whose maintenance or use during the refueling/maintenance outage was on, or could reasonably have been on, the critical path.
- Whose maintenance would cause workers to receive high doses of radiation.
- Whose maintenance frequency or manhour requirements was deemed to be excessive.

To ensure proper interpretation and to supplement the written records and data, interviews were held with operations, maintenance, and engineering personnel at the plants studied and with B&W engineering personnel.

As an addition to the contract, 17 "key valves" that had been identified as significantly impacting plant availability were studied in greater detail. Data for this special study were obtained from the Oconee, Rancho Seco, and TMI-1 plants.

Limiting Factors

For convenience and to ensure that no available data were omitted, the data were grouped and analyzed by system, component, and/or refueling work activity. For each system and in some cases for each component, a limiting factor for operation (LFO), a limiting factor for maintenance (LFM), and/or a limiting factor for refueling (LFR) were calculated. Also, a combined equipment limiting factor (CELF) was developed to include a combination of the LFO and the actual outage extension term of the LFR for those systems/components which are not directly a part of the refueling activity. Appendix C gives details on the definition and application of the formulas for calculating the four limiting factors — LFO, LFM, LFR, and CELF. Pertinent comments on application of specific limiting factors are given below.

Limiting Factor for Operation (LFO) — In many cases, work on more than one component occurred in a given power reduction or plant outage, but our data analysis treats each component work event as though it had forced the power reduction or shutdown independently. Thus, the loss of plant availability during the necessary shutdown/cooldown and heatup/startup time is charged to each activity in the outage. (Table C-1 in Appendix C gives the average plant availability loss in EFPH for shutdown/cooldown to the various levels necessary for repair work activities on Oconee 1, 2, and 3; Table C-2 gives similar information for Rancho Seco.) Simultaneous repair activities are also charged fully to each component or activity. Sometimes repairs are made just because the unit is down. Including these data tends to overestimate the LFO.

Limiting Factor for Maintenance (LFM) — Only a fraction of this work is on the critical path for plant operation and directly impacts plant availability. We have calculated the manhours from the number of men used to perform a given task and the clock hours it took them as shown on the Station Work Requests. In cases where our data are inadequate, we have estimated either the number of men or clock hours or both to determine the manhours.

PHASE III — LIMITING FACTORS ANALYSIS

Categorization

The CELFs are categorized as follows in Table 3-2:

O: Utility management operating philosophy and practice

R: Regulatory requirements (NRC, OSHA, environmental, state, and local)

D: Design requirements

The design requirements are analyzed in detail in Sections 3 and 4 and include "design traps," which could cause operator errors/maintenance problems.

Assignment of Priorities

In Table 3-2, the design category items are listed on a CELF-priority basis for analysis. The priority sequence is the same as given in Table 3-1.

Evaluation

Table 3-1, "Summary of Calculated Limiting Factors," has 29 entries with CELFs for 28 of these. The other entry gives the 1977 refueling outage losses. For analytical purposes, the low-priority entries have been omitted from Table 3-2, "Limiting Factor Categorization," and Table 3-3, "Problem/Solution Summary."

PHASE IV — ADDITIONAL PLANT COMPARISONS

Oconee 1 is the reference plant for this availability evaluation. Additional data were obtained from Oconee 2 and 3, Rancho Seco, and TMI-1 to test the Oconee 1 data that were subsequently used in the limiting factors calculations.

PHASE V — EVALUATION OF CURRENT PROGRAMS

Programs to reduce or eliminate design category CELFs are listed in Table 3-3. Each CELF includes a proposed solution, and the solution is categorized as follows:

- Fixes that are already developed, available, and ready to be implemented.
- Solutions that are known but not fully engineered or tested.
- Solutions that need further study, development, resolution, etc.

In addition, this table indicates whether the proposed solution is applicable to operating plants.

PHASE VI — IN-DEPTH ANALYSIS OF LIMITING FACTORS

Section 3 identifies the major limiting factors found in this study. The status of plant modifications was considered in determining which limiting factors to recommend for further study.

Section 4, which is the bulk of this report, identifies the plant availability limiting factors (LFOs, LFMs, and LFRs) and includes detailed discussions, conclusions, and recommendations for the plant systems and components and for refueling activities. The results of the key valve study are reported in Section 4 along with radiation exposure data and several other topics.

Tables 4-1 and 4-2 list the Oconee 1 systems and components according to the calculated values of their LFOs and LFMs, respectively. Table 4-3 lists the systems according to the average LFO for the three Oconee units in 1977 (exclusive of refueling outage activities). Figures D-1, D-2, and D-3 and Table D-1 give further details on the current data. Table 4-4 shows the LFRs for Oconee 1 and 3, Rancho Seco, and TMI-1.

PHASE VII — FINAL TECHNICAL REPORT

This report contains four major sections and 10 appendices. The first three sections are an executive summary. Collectively, these three sections summarize the rationale and methodology for this study and present the major conclusions and recommendations of the report.

Details of the rationale and study methodology for collecting and analyzing the data are given in Appendices A, B, and C. The basic data are given in the figures and tables in Appendices D, E, F, and G. Abbreviations, definitions, and references are given in Appendices H, I, and J.

Section 3

ANALYSIS AND CONCLUSIONS

3.1. LIMITING FACTOR IDENTIFICATION

Equipment and Refueling

Table 3-1 lists the LFO historical (from Table 4-1), the LFO current (from Table 4-3), and the portion of the LFR (from Table 4-4) that caused refueling outage extensions due to equipment problems. These three factors are combined by formula 1 to give an equipment limiting factor (CELF):

$$(1) \qquad \text{CELF} = \frac{\text{LFO historical} + \text{LFO current}}{2} + \begin{matrix}\text{refueling}\\ \text{outage}\\ \text{extension}\end{matrix} \qquad \text{EFPH/unit-year}$$

In formula 1, the historical and current limiting factors are averaged to reduce the statistical spread of the data. Since both the historical and current data contain approximately the same amount of coverage (3½ years on Unit 1 for historical data versus one year on Units 1, 2, and 3 for the current data), it seems appropriate to give equal weight to each set of data. In formula 1, the refueling outage extension time is added to the combined limiting factors since these are the best available data on the losses during the refueling outage due to equipment. Based on formula 1, column 4 of Table 3-1 gives the CELFs. The last column of this table shows the individual and total losses due to normal 1977 refueling activities.

Based on the data in Table 3-1, we draw the following conclusions:

- Most of the equipment outage time losses (and especially 1977 losses) were due to forced outages on a relatively few components. These include steam generators, fuel handling equipment, control rod drives, RC pumps, and RC pump motors. The five items account for 76% of the total equipment outages. As will be shown in section 4.2.1.6, the Oconee units are the only B&W operating units that have experienced significant plant outages for steam generator problems. If steam generators are discounted, the four remaining items account for 62% of the total equipment outages.

- Except for the five items mentioned above, the remaining equipment outage time is caused by numerous low-impact problems throughout the plant. It should be noted that in Table 3-1, item 7, core physics and reactor safety includes such delays as xenon hold, which result from a shutdown for any reason and which are directly related to the number of shutdowns.
- The refueling outages were extended by maintenance activities for the following equipment: steam generators, fuel handling equipment, reactor coolant pumps, incore instrumentation, and reactor building polar crane. In addition, main turbine maintenance, though not on the critical path at Oconee, could cause refueling outage extensions depending on the time needed to perform the work.

Other availability studies involving critical path activities and repair times during a shutdown sometimes assign shutdown and startup times in a sequence to one item at a time. This weights each item in a sometimes arbitrary manner and skews the priority assigned to each item. Eliminating the first item in the sequence may significantly affect the item next in the sequence. The critical path savings due to eliminating the first item may be reduced or made nonexistent. To overcome this skewing effect, we assume in this limiting factor study a shutdown and startup for each item as if it were the only item and cause of the shutdown. Therefore, caution must be applied in any direct comparison of the data for the equipment and refueling outages in Table 3-1. As discussed in section 2 in the data analysis of equipment outages, the data are treated as though each simultaneous equipment outage independently caused a plant outage. Also, the startup, shutdown, and component access time is charged to each activity as though it independently caused an outage. This treatment preserves all the information originally contained in the data and avoids random variations in the final result. However, it also results in a total equipment outage time that is greater than the plant outage time. A similar treatment was given the refueling outage data. However, the effect is greater for the equipment outage data than for the refueling outage data. For instance, comparison of the CELF and the refueling outage data in Table 3-1 might seem to indicate that equipment outages caused 3.4 times as much loss of availability as did refueling. This was not the case, as can be seen by inspection of the plant operating records in Appendix D.

The limiting factors in Table 3-1 reflect the outage contribution from all causes including equipment failure, maintenance practices, operating philosophy and practice, and regulatory requirements. Section 3.2 assigns an outage cause category to each limiting factor and discusses the relative contribution of each category. Section 3.4 evaluates programs that could reduce or eliminate those limiting factors assigned to the "Design Requirements" category.

Maintenance and Radiation Exposure

The annualized maintenance manhours (limiting factor for maintenance, LFM) for each system/component are tabulated in section 4 along with tables showing radiation exposure dose levels for general categories of maintenance activities. The limiting factors for maintenance are based on Oconee 1 historical data (7/1/74-12/31/77) and are shown in Table 4-2. The radiation exposure dose levels are 1977 data for Oconee 1, 2, and 3. Table 4-10 gives dose levels by quarter for routine work, and Table 4-11 gives dose levels by unit for special shutdown work.

Comparison of the historical LFOs (Table 3-1) and the historical LFMs (Table 4-2) shows that the LFO and LFM results are generally consistent; equipment that resulted in a loss of plant availability generally had high maintenance hours. An exception to this is the main turbine work, which caused little loss of plant availability at Oconee but has been a relatively high maintenance item at all plants.

Equipment or activities that caused a loss of availability and which were also identified as a significant source of radiation exposure are steam generator, refueling, reactor coolant pumps, reactor coolant pump motors, primary valves, incore instrumentation, and inservice inspection. A major source of radiation exposure which had little impact on plant availability is radioactive waste handling, filter changes, and system modification. Table 4-10 shows that these three items accounted for over 20% of the total 1977 routine work dose. Other items that required high maintenance and/or caused high radiation exposure but which did not cause significant loss of availability include general station maintenance/surveillance, station modifications, secondary valves, hydraulic suppressors, and heat exchangers.

The radiation exposure data are not sufficiently detailed to draw more specific conclusions. Further studies involving acquisition and analysis of radiation exposure data for specific maintenance activities would be necessary to proceed further. Components and work activities of particular interest should include steam generators, major component inspection, reactor coolant pumps (seals and bearings), radiation waste disposal (filters, demineralizers, and evaporators), control rod drives, and primary valves.

3.2. LIMITING FACTOR CAUSES

Categorization

Table 3-2 categorizes the combined equipment limiting factors and the refueling outage work activities according to whether the loss was due to

- O: Utility management operating philosophy and practice,
- R: Regulatory requirements (NRC, OSHA, environmental, state, and local),
- D: Design requirements.

Table 3-2 gives further definitions and subcategories of these items. Each limiting factor category assignment is made on the basis of a positive (yes) answer to any of the subcategory questions given in Table 3-2.

In our analysis of limiting factors, we considered two factors. First, the root cause(s) of the event or work activity. The root cause of the outage may be one of the three categories or (more often) some combination of them. Second, we considered the time required for correction of the problem and return to service. This time is also affected by one or more of the three categories.

Conclusions Regarding LF Categorization

As shown in Table 3-2, most limiting factors are judged to be caused by more than one of the three categories. The available data do not permit a percentage allocation of the total equipment outage time to each of the three categories. It is our judgment, supported by observation of specific instances, that the equipment and refueling outage times (limiting factors) could be reduced by improved maintenance practices. These include outage planning, spare parts, improved procedures, processes, and tooling. We also note that the training of plant maintenance personnel has received comparatively little attention relative to that given operators, designers, etc. The training of maintenance personnel is also complicated by the need to limit personnel exposure.

The regulatory categories primarily involve safety inspections and core power restrictions. The former generally involved plant shutdown even though the plant was capable of generating power without the test.

In most cases, we also determined that the regulatory category factor could be reduced through improved operating practices, improved designs, or both. Changing regulatory criteria have not always allowed the designer to produce the equipment that could most efficiently meet today's criteria.

3.3. EVALUATION OF DATA

Adequacy of Data

As shown elsewhere, data were collected and to a degree analyzed in three discrete packages — current data, historical data, and refueling data. Of these three, we believe the current and refueling data to be more complete and more accurate than the historical data because these data were based not only on documented records but also on observations and interviews during or soon after the outage. The current and refueling data also more accurately reflect current problems rather than historical problems, some of which may have been solved. Although the condition of reflecting current problems is generally advantageous, it can have undesirable aspects. For example, in this study steam generator problems were by coincidence at their peak at the Oconee units, especially at Oconee 1 during 1977.

The pre-1977 historical data, in contrast to the current and refueling data, were based almost entirely on documented records. Work requests were the primary source of documented records; in earlier plant outages there were cases where work requests could either not be identified or not correlated to the outage. As discussed in Appendix B, work requests were originally used by planners and schedulers to assign and implement maintenance and repair tasks and not as a device for recording work performed. Consequently, older work requests often gave only estimated repair hours and estimated numbers of men needed to complete a task. Problem symptoms were often given with little or no information on the nature of the failure or repairs. This situation was complicated by the many forced outages which involved multiple equipment repairs (LFs). To each repair, we assigned additional outage time for shutdown, cooldown, startup, etc. These assignments were made from estimates based on current data and engineering judgment. These potential historical data problems were minimized by comparing work request data with plant outage and other records as discussed in Appendix B. In addition, interviews were held with engineers and operators to help identify systems/components that may have operating histories different from those shown by the data. The end result, we believe, shows reasonably good correlation between current and historical data as shown by Table 3-1.

Plant Comparisons

Since the data evaluated in this study represent a small sample relative to the total, the incidence of low-frequency, high-impact problems could influence the conclusions to be drawn. To provide more insight into this possible skewing of our conclusions, we have analyzed and studied unpublished availability data from

B&W equipment outage factor (EOF) plant availability records. The EOF data are based on oral reports, internal B&W status reports, and internal B&W site problem reports. These data differ from the data obtained under this study in the following ways:

- The EOF data concentrate on primary side (NSS) problems.
- In the EOF data, when a shutdown is attributed to more than one component, the outage time is shared evently among all involved components.
- In this limiting factor study, operating practice and human error factors are included with equipment-related factors. In the EOF data, operating practice and human error are separated from equipment-related factors.

The EOF data are internally consistent and permit one to draw meaningful conclusions on these data alone. Since the method of analysis, component/activity and breakdown of these data are different from that used in the present contract, the numerical values of the EOF and the LF data are not directly comparable. However, conclusions to be drawn from the two sets of data are comparable.

In the EOF data, we studied only operating data from the six oldest (mature) B&W plants starting with the Rancho Seco plant and including all B&W plants having earlier commercial operating dates. These six plants all have the same basic NSS design, including a 177-fuel assembly core, two-loop plant with design electrical ratings between 819 and 918 MWe. To minimize the influence of problems that occur during the plant shakedown phase, we studied only recent data over a reasonable time span. Specifically, our study covered the period from January 1, 1977, through December 31, 1978. From a study of the EOF data we note the following:

- Comparisons of the EOF data and the LF data show that, in general, the same items appear in both sets of data and in approximately the same order.
- Essentially all steam generator problems occurred at the Oconee units.
- The balance-of-plant problems were less severe at the Oconee units than at the others. This, however, was heavily influenced by turbine problems at ANO-1 and Rancho Seco.
- The NSS problems (excluding the steam generator) were more severe at the Oconee units than at the other units.

- In both sets of data and except for steam generator losses at Oconee, the refueling outage activities are by far the largest single contributor to loss of plant availability.

The EOF comparisons also show that there are significant differences in plant availability among the different plants; this difference may be due not only to low-frequency, high-impact events but also to such factors as plant experience, operating practices, balance-of-plant (BOP) design and performance, training, spare parts, maintenance practices, and the like. These factors illustrate that the limiting factors identified in this study are the factors for a few particular plants operating in a particular manner; caution should be used in relating the results from this study to other plants.

Our analysis of the EOF data showed that the "best composite demonstrated performance by any mature plant" and the "best demonstrated performance by one single plant" during this two-year period were as follows:

	Availability loss, EFPD/unit-yr	
Item	Best composite performance by any plant	Best performance by one plant
Refueling outage	30.7	33.1
NSS equipment	3.1	11.5
Balance-of-plant equipment	6.2	12.8
Other (including human error)	6.9	13.3
Total	46.9	70.7

A detailed study of refueling outages (1) has concluded that with all recommended improvements, a typical refueling outage length for mature plants can be reduced to a goal of 21-22 days. We think that with identifiable availability improvements implemented, the average mature plant can use as a goal an equipment performance midway between the best composite plant and the best single plant (about 27 EFPD equipment loss). This plus 22 EFPD refueling outage gives a total availability loss of about 49 EFPD/unit-year. We recognize that unusual events and one-of-a-kind occurrences can have major impact on planned schedules and will generally prevent idealistic performance.

3.4. PROGRAMS TO REDUCE LIMITING FACTORS

Table 3-3 summarizes the engineering/R&D programs we believe can best reduce or eliminate the design-related limiting factors identified in this study. The limiting factors shown here are primarily from Table 3-2, and the related programs are primarily a summary of the major programs discussed in section 4. Table 3-3 categorizes the program status as to whether the design/development is

C — complete, ready to be implemented,

U — underway, or

N — a new program that may need (1) a sponsor, (2) more engineering/definition, (3) funding, (4) testing, and (5) implementing.

Table 3-3 also indicates whether or not the recommended programs can be used to backfit operating plants.

Table 3-1. Summary of Calculated Limiting Factors, EFPH/Unit-Year

System/component/work activity	LFO hist. Oconee 1[a]	LFO curr. Oconee 1, 2, 3[b]	Ref outage extension: Oconee 1, 3, RS, TMI-1	CELP	Refl outage[c]: August 1977 refl loss, Oconee 1, 3, RS, TMI-1
	Equipment outage from all causes (operational, regulatory, and design)				
Steam Generator	733	1119	388	1314	
Refueling outage work activities[c]					
Fuel handling operations	--	--	--	--	245
Containment leak tests	--	--	--	--	136[d]
Shutdown/startup	--	--	--	--	124[e]
Core physics, reactor safety	--	--	--	--	105[e]
Secure/reinstall CRDM	--	--	--	--	85
Detension/retension RV head	--	--	--	--	62
ARIS work	--	--	--	--	52[d]
Clean transfer canal	--	--	--	--	36
Fill/drain transfer canal	--	--	--	--	31
Move equip in/out of RX building	--	--	--	--	28
Remove/reinstall RV head	--	--	--	--	28
Remove/reinstall plenum	--	--	--	--	24
Reactor building purge	--	--	--	--	20
Install/remove canal shield plate	--	--	--	--	19
Remove/install shield blocks	--	--	--	--	13
Install/remove stud hole plugs	--	--	--	--	13
Remove/reinstall RV head insul'n	--	--	--	--	12
HP survey	--	--	--	--	7
Total					1040
Fuel handling equipment	294	--	118	412	
Control rod drives	356	434	--	395	
RC pumps	165	94	220 NC	350	
RC pump motors	335	109	--	222	
Core physics and reactor safety	72[f]	187[g]	--	130	
Cont and monitoring equipment	69	10	79	118	
Electrical systems	225	0	--	112	
Reactor and internals	205	0	--	102	
Turbine lubricating oil	45	75	--	60	
Pressurizer	40	65	--	52	
Feedwater	13	66	--	40	
Condensate	29	49	--	39	
Makeup and purification/HPI	0	59	--	30	
Turbine EHC	19	34	--	26	
Main turbine	41	5	NA[h]	23	
Decay heat/LPI	0	42	--	21	
Heater drains	11	27	--	19	
Liquid waste	0	30	--	15	
Generator stator cooling	3	18	--	10	
Coolant storage	9	10	--	10	
Main steam	10	8	--	9	
Suppressors and hangers	18	0	--	9	
Chemical addition and sampling	0	16	--	8	
Reactor building spray	0	10	--	5	
Polar crane	6	0	NA[i]	3	
Instrument air	0	2	--	1	
Plant protection equipment	1	1	--	1	
Total equipment outage time, EFPH				3536	

[a] Historical: July 1, 1974 through December 31, 1977, data.

[b] Current: 1977 data.

[c] Includes those activities (from Table 4-4) related specifically to refueling outage activities. Equipment maintenance activities performed during the outage are included with the system/components.

[d] Not annual tests.

[e] Includes a factor of 0.5 to accout for activity being performed at part power.

[f] Does not include 235 EFPH for startup physics.

[g] Does not include 253 × 0.5 = 126 EFPH for startup physics tests.

[h] Main turbine work not near critical path at Oconee. Could be near critical path at other plants.

[i] Not identified as a refueling work activity to be followed, but is known to have caused refueling delays (see section 4.2.8).

Table 3-2. Limiting Factor Categorization

Limiting factor, system/component/activity	CELF	Outage category (see code definitions at end of this table) O[a]	R[a]	D[a]	Basis for categorization (See code definitions at end of this table)
1. Steam generator	1314				
Leaking tubes		x	x	x	O-1,O-2,R-2,D-2,D-3
Eddy-current inspection		x	x	x	O-2,R-2,D-2
2. Refueling outage work activities[b]	1040				
Fuel handling operation (P=245)[c]		x		x	O-2
Containment leak tests (P=136 NC)		x	x	x	O-2,R-2,D-2
Shutdown/startup (P=124)		x		x	O-2,D-2
Core physics and reactor safety (P=105)		x	x	x	O-2,R-2,D-2
Secure/reinstall CRDM (P=85)		x		x	O-2,D-2,D-3
Detension/retension RV head (P=61)		x		x	O-2,D-2,D-3
ARIS work (P=52)			x		R-2
Clean transfer canal (P=36)		x		x	O-2,D-2,D-3
Fill/drain transfer canal (P=31)		x		x	O-2,D-2
Move equip into/out of reactor bldg (P=28)		x		x	O-2,D-2,D-3
Remove/reinstall RV head (P=28)		x		x	O-2,D-2
Remove/reinstall plenum (P=24)		x			O-2
Reactor building purge (P=20)		x			O-2
Install/remove canal seal plate (P=19)		x		x	O-2,D-3
Remove/install shield blocks (P=13)		x			O-2
Install/remove stud hole plubs (P=13)		x			O-2
Remove/install RV head insulation (P=12)		x		x	O-2,D-2
Health physics survey (P=7)		x			O-2
Total (P=1040)					
3. Fuel handling equipment	412				
Fuel handling bridges		x		x	O-2,D-2,D-3
Transfer system				x	D-1,D-2,D-3
4. Control rod drive system	395				
Ratchet trip				x	D-1,D-2,D-3
Failed stators		x		x	O-1,D-2,D-3
Falure of absolute position indicator		x		x	O-1,D-2,D-3
Low cable insulation resistance		x		x	O-1,D-2,D-3
Vent valve leakage		x		x	O-1,D-2,D-3
5. Reactor coolant pumps	350				
Seal leakage/failure		x		x	O-1,D-2,D-3
Pump balancing		x			O-2
6. RC pump motors	222				
Lube oil level and leaks				x	D-1,D-2,D-3
7. Core physics and reactor safety	130				
Power ascension delays		x	x	x	O-2,R-2,D-2
Core power tilt		x	x	x	O-2,R-2,D-2
Xenon hold		x	x	x	O-2,R-2,D-2
8. Control and monitoring equipment	118				
Integrated control system		x		x	O-2,D-1
Non-nuclear instrumentation				x	D-1,D-5
Incore detectors				x	D-1,D-3
9. Electrical systems	112				
Generator				x	D-1,D-5
Exciter				x	D-1,D-5
10. Reactor and internals	102				
Surveillance specimen holder tube				x	D-1,D-2
11. Turbine lubricating oil	60				
Oil purifier				x	D-1,D-2,D-3
Turning gear lift pumps				x	D-1,D-2,D-3
Oil leaks				x	D-1,D-2,D-3

Table 3-2. (Cont'd)

Limiting factor, system/component/activity	Outage category (see code definitions at end of this table) CELF	O[a]	R[a]	D[a]	Basis for categorization (see code definitions at end of this table)
12. Pressurizer	52				
Valves		x		x	O-2,D-2,D-3
13. Feedwater	40				
Valves		x		x	O-2,D-2,D-3
Heaters				x	D-1,D-2,D-3
14. Condensate	39				
Valves		x		x	O-2,D-2,D-3
15. Makeup and purification/HPI	30				
Water chemistry		x		x	O-2,D-1,D-2,D-3
Valve lineup		x		x	O-2,D-2,D-3
Valves		x		x	O-2,D-2,D-3
Filters/water purifiers				x	D-1,D-2,D-3
16. Suppressors and hangers	9				
Suppressors			x	x	R-2,D-2,D-3
Valves	?				
Pressurizer		x		x	O-2,D-2,D-3
Valve operators				x	D-1,D-2,D-3
Body-to-bonnet leaks				x	D-1,D-2,D-3
Valve application		x		x	O-2,D-1,D-2,D-3
Water chemistry	?				
Primary		x		x	O-2,D-1,D-4
Secondary		x		x	O-2,D-1,D-4

[a]Data do not permit allocation of percent cause for each category.

[b]Includes those activities (from Table 4-4) related specificatlly to refueling outage activities. Equipment maintenance activities performed during the outage are included with the system/component.

[c]See Table 4-4.

Code Definition and LF Outage Categorization

O — Utility-Controlled Operating Philosophy and Practice

Was the limiting factor produced by matters that are mostly controlled by management policy, grid and system operating requirements and/or indirectly through plant operating and maintenance practice and philosophy?

O-1 — Was the limiting factor affected by management policy or decisions, including system and grid or other interwoven considerations which resulted in out-of-conformance or less-than-expected operating procedures for the identified LF item?

O-2 — Was the individual LF adversely affected by operating philosophy or practices, management decisions on such items as levels of spare parts, availability of contracted or trained or qualified manpower on all shifts or other considerations that resulted in the individual item being classed as an LF without judgment as to the overall acceptability of the practice or decision?

R — Regulatory Requirement (NRC, OSHA, Environmental, State, Local)

Was the limiting factor due to regulatory requirements?

R-1 — Did a regulatory authority require a plant shutdown or reduction in power?

R-2 — Did a regulatory requirement (regulatory guide, technical specification, etc.) require a shutdown or power reduction when the plant was capable of generating power or returning to power?

D — Design Requirements

D-1 — Was the LF due to items other than "operating" or "regulatory" categories?

D-2 — Is a design change or new design required to reduce the LF without judgment to the overall acceptability of the original design?

D-3 — Was the LF adversely affected by equipment performance not known to be caused by installation, operation, or maintenance practices?

D-4 — Was the LF a reault of requirements for exceptional skill in operation or maintenance?

D-5 — Was the LF a result of a reasonable number of random failures without excessive repair time and with no design change suggested?

Table 3-3. Problem/Solution Summary

Limiting factor[a] and related engineering/R&D programs	Program/solution status C[b]	U[b]	N[b]	Solution applicable to operating plants
1. Steam Generator				
a. Leaking Tubes				
(1) EPRI/utility/NSS mfr steam generator research programs		x		x
(2) Develop improved tube leak detection, plugging techniques			x	x
(3) Develop techniques and licensing criteria to permit continued operation with greater leak rates			x	x
(4) Develop means to collect and prevent escape of gaseous activity after cooldown and access for maintenance			x	x
b. Eddy-Current Inspection				
(1) Develop improved inspection techniques to avoid refueling outage delays			x	x
2. Refueling Outage Work Activities				
a. Fuel Handling Procedures				
(1) Develop improved critical components to improve reliability and speed up operations		x	x	x
(2) Develop automatic bridge/trolley to speed up operation			x	x
(3) Employ an improved, multi-function mast		x		x
(4) Develop equipment capable of full checkout prior to start of fuel movement			x	x
b. Containment Leak Tests				
(1) Develop improved equipment/techniques to shorten test interval			x	x
d. Secure/Reinstall CRDM				
(1) Develop improved equipment/techniques to shorten activity times			x	x
e. Detension/Retension RV Head				
(1) Develop improved equipment/techniques to speed up operation			x	x
g. Clean Transfer Canal				
(1) Develop improved equipment to minimize cleaning time			x	x
h. Fill/Drain Transfer Canal				
(1) Develop improved canal filling techniques to give fast fill without turbidity problems			x	x
i. Move Equipment Into/Out of Reactor Building				
(1) Develop improved polar crane design to minimize impact of equipment failure			x	x
(2) Employ polar crane redundancy principles (jib crane, conveyors, etc.) to improve efficiency of use			x	x
(3) Develop procedures and licensing criteria to allow containment hatch to be left open during start of refueling outage			x	x
j. Remove/Install Reactor Vessel Head				
(1) Employ device to permit quick head rigging	x			x
m. Install/Remove Canal Seal Plate				
(1) Install new inflatable seal design		x		x
p. Remove/Install Reactor Vessel Head Insulation				
(1) Develop improved equipment and procedures to shorten activity			x	x
3. Core Physics and Reactor Safety				
a. Power Ascension Delays				
(1) Develop technology to eliminate power ascension limits due to PCI criteria			x	x

Table 3-3. (Cont'd)

Limiting factor[a] and related engineering/R&D programs	Program/solution status C[b]	U[b]	N[b]	Solution applicable to operating plants
b. Core Power Tilt				
(1) Develop movable incore detector calibration probe to resolve anomalous detector readings			x	x
c. Xenon Hold				
(1) Use feed-and-bleed mode of operation			x	x
(2) Develop improved accessible computer codes to minimize precautionary "holds."			x	x
d. Restart Physics Tests				
(1) Develop codes to permit making tests at non-equilibrium conditions			x	x
4. Fuel Handling Equipment				
a. Fuel Handling Bridges (see item 2.a above)				
b. Transfer System (see item 2.a above)				
5. Control Rod Drives				
a. Drive Ratchet Trip				
(1) Install circuitry to prevent restoration of power during rod drop	x			x
b. Failed Stators				
(1) Install improved O-rings to prevent wetting	x			x
(2) Install varnish insulation and monofilar windings to minimize failures	x			x
(3) Install interlock to prevent energizing stators without cooling water	x			x
c. Failure of Absolute Position Indicator				
(1) Develop improved indicators employing redundancy			x	
(2) Develop improved repair techniques/hardware			x	x
d. Low Cable Insulation Resistance				
(1) Install moisture-resistant connectors	x			x
e. Failure of Vent Valve				
(1) Install improved valve design	x			x
6. Reactor Coolant Pumps				
a. Seal Leakage/Failure				
(1) Implement comprehensive seal improvement program		x	x	x
7. Reactor Coolant Pump Motors				
a. Lube Oil Level and Leaks				
(1) Develop device to maintain constant lube oil level without lube oil leaks and to reduce inventory			x	x
8. Control and Monitoring Equipment				
a. Integrated Control System (none suggested)				
b. Non-Nuclear Instrumentation (none suggested)				
c. Incore Detectors				
(1) Implement incore detector improvement program to reduce failures and improve reliability		x	x	x
9. Electrical				
a. Generator (none suggested)				
b. Exciter (none suggested)				

Table 3-3. (Cont'd)

Limiting factor[a] and related engineering/R&D programs	Program/solution status C[b]	U[b]	N[b]	Solution applicable to operating plants
10. Reactor and Internals				
a. Surveillance specimen holder tube (no further effort suggested.)				
11. Turbine Lube Oil				
a. Oil Purificer				
(1) Implement or follow field installation of improved designs		x		x
b. Turning Gear Lift Pumps				
(1) Implement or follow field installation of improved designs		x		x
c. Oil Leaks				
(1) Implement gasket improvement program			x	x
12. Pressurizer				
a. Valves (see valve program summary below)				
13. Feedwater				
a. Valves (see valve program summary below)				
b. Heaters				
(1) Implement feedwater heater study program for improved performance			x	x
14. Condensate				
a. Valves (see valve program summary below)				
15. Makeup and Purification/HPI				
a. Water Chemistry (see water chemistry probrams below)				
b. Valve Lineup				
(1) Enforce operating procedures or install valve interlock device	x			x
c. Valves (see valve program summary below)				
d. Filters/Water Purifiers				
(1) Improve equipment design, procedures, layout, etc. to minimize radwaste problems			x	x
16. Suppressors and Hangers				
a. Suppressors				
(1) Implement program to improve snubber performance and reduce inspection requirements			x	x
Valves				
a. Pressurizer Valves				
(1) Use hermetically sealed valves in critical locations, such as first-off valves and high-pressure valves			x	x
(2) Minimize seat leakage by minimizing crud in RC system			x	x
(3) Develop method to monitor seat leakage			x	x
(4) Locate valves away from pressurizer or use heat shields to minimize heat effects			x	x
(5) Ensure suitable refurbishing and handling procedures for safety/relief valves			x	x
(6) Replace pressurizer spray control valve with hermetically sealed solenoid-operated valve	x			x
b. Valve Operators				
(1) Implement a program to ensure valve/valve operator compatibility			x	x

Table 3-3. (Cont'd)

Limiting factor[a] and related engineering/R&D programs	Program/solution status C[b]	U[b]	N[b]	Solution applicable to operating plants
c. Body-to-Bonnet Leaks				
(1) Implement a program to study and minimize valve B/B leaks (gaskets)			x	x
d. Valve Application/Qualification				
(1) Develop valve application guidelines and qualification procedures for key generic valves			x	x
Water Chemistry				
a. Primary				
(1) Develop a program to identify and eliminate sources of high chlorides			x	x
(2) Develop improved means to remove corrosion products			x	x
(3) Implement program to minimize impact of chemical contaminants			x	x
(4) Implement program to control hydrogen in the RC system			x	x
b. Secondary				
(1) Implement study to identify sources of iron and means to eliminate or tolerate			x	x
(2) Identify ways to minimize impact of turbine steam cycle effluents			x	x
(3) Develop computerized on-line water chemistry monitor			x	x
(4) Implement study to identify causes of chemical addition and sampling system malfunctions			x	x

[a]Design category only.

[b]C: design complete, ready to be implemented; U: design/development underway; N: new program, more definition and funcing needed.

Section 4

LIMITING FACTOR IDENTIFICATION

4.1. GENERAL

Tables 4-1 and 4-2 list the Oconee 1 systems and components ranked according to their calculated limiting factor for operation (LFO) and limiting factor for maintenance (LFM), respectively. Table 4-3 lists the systems and components ranked according to calculated LFO for the average of the three Oconee units in 1977 (exclusive of refueling activities). Tables 4-1 through 4-3 are summaries of the non-refueling data given in Appendices D and E. The tables also show the factors used to calculate LFOs and LFMs. Table 4-4 lists the calculated 1977 limiting factor for refueling (LFR), the factors used in the LFR formula, and ranks the factors according to the calculated LFR. Table 4-4 is a summary of the refueling outage data given in Appendix F.

The subsections in section 4.2 list the calculated LFOs, LFMs, and LFRs and the calculated rank for each of the systems/components considered in this study. These subsections also discuss (by system) the data, the analysis, and the conclusions and recommendations for the systems/components and refueling activities. Section 4.3 gives further discussions, analysis, conclusions, and recommendations for the refueling outage activities. Section 4.4 gives discussions, conclusions, and recommendations on the key valve study. Radiation exposure data for routine and special shutdown work were evaluated for dose reduction. The data are presented and recommendations given in section 4.5. Special analyses of pumps/motors, heat exchangers, and other problems of interest are presented in section 4.6.

It should be noted that this is an availability study and not a reliability study. There is an insufficient number of items and failures to produce a reliability number; the report identifies items as installed, maintained, and operated in an individual plant.

4.2. SYSTEMS

4.2.1. Reactor Coolant Systems

4.2.1.1. Reactor and Internals (1A)

Study results	Limiting factor	Rank	
LFO (Oconee 1, historical data, Table 4-1)	295	5	
LFO (Oconee 1, 2, 3, 1977 data, Table 4-3)	0	0	
LFM (Oconee 1, historical data, Table 4-2)	1772	3	
LFR (four-plant avg, Table 4-4)	38	8	Detension/retension RV head
	10	11	Remove/reinstall RV head
	6	13	Remove/reinstall plenum
	3	16	Remove/reinstall RV head insulation

Discussion, Conclusions, and Recommendations

During normal refueling operations at TMI-1 in 1976, the surveillance specimen holder tubes (SSHTs) were found to have suffered flow-induced vibrational wear. The problem led to the removal of the SSHTs from all the operating B&W plants and modification of the SSHTs on new B&W plants during initial startup activities. This work had an impact on the availability of the operating B&W units in 1976, including Oconee 1, which was shut down in April and May 1976 for removal of the SSHTs from the reactor internals. The problem has been addressed on both operating and future B&W plants. We have no recommendations for other programs to improve the reactor and internals.

4.2.1.2. Fuel and Rods (1B)

Study results	Limiting factor	Rank	
LFO (Oconee 1, hist. data, Table 4-1)	0	--	
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--	
LFM (Oconee 1, hist. data, Table 4-2)	0	--	
LFR (four-plant avg, Table 4-4)	106	3	Refueling operation

Discussion, Conclusions, and Recommendations

Before 1978 the only apparent availability impact involving B&W fuel has been the effect of the slight fuel assembly bowing on refueling operations. This was handled by re-indexing the fuel handling equipment or by reordering the fuel loading scheme to fill the surrounding fuel assembly locations before trying to insert the bowed assembly.

4.2.1.3. Reactor Coolant Pumps (1C)

Study results	Limiting factor	Rank	
LFO (Oconee 1 hist. data, Table 4-1)	165	8	
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	94	5	
LFM (Oconee 1 hist. data, Table 4-2)	1603	5	
LFR (four-plant avg, Table 4-4)	124	2	Remove/reinstall RC pump seal

Analysis of Oconee 1 Operational Historical Data (See Table E-1)

The data show three events that forced or extended a plant shutdown: one to correct a pump shaft balance problem, one to replace pump seals, and one to inspect pump seals. (See further discussion of the latter two events under 1977 operational data below.) Two Westinghouse pumps were involved in these three events.

Analysis of Oconee 1 Maintenance Historical Data

The Oconee 1 pump maintenance work may be classified as follows:

Activity	No. of work events	Manhours
Seal maintenance/ replacement	11(65%)	4124(73%)
Seal inspection	4(23%)	1476(26%)
Vibration/balanc'g	1(6%)	8(1%)
Oil leaks at lower motor bearing	1(6%)	4(1%)

Maintenance problems were distributed among the four Westinghouse RC pumps as follows:

	RCP-1A1	RCP-1A2	RCP-1B1	RCP-1B2
Seal maintenance/ replacement	5	1	2	3
Seal inspection	--	1	1	2
Vibration	--	1	--	--
Oil leaks at lower motor bearing	--	--	--	1
Total	5	3	3	6

Analysis of Oconee 1, 2, and 3 1977 Operational Data (See Appendix D)

This limiting factor was caused by two events at Unit 1 near the end of the 1977 refueling outage identified above under Oconee 1 data. After the plant had undergone the scheduled refueling outage maintenance, it was found during plant startup that the seal on Westinghouse pump 1B1 had excessive leakage; it was replaced. The replaced seal had metal particles in the seal faces. During the same period the seal on pump 1B2 was inspected and found to be in satisfactory condition. It was later determined that the metal particles originated from repair of the seal injection throttle valve upstream of the seal and downstream of the seal injection filters.

Related Data From Other Plants

During the 1977 Rancho Seco refueling outage, debris was found in the Bingham pump seal after seal reassembly. The source of this debris was not identified. Cases are also documented (Oconee 2 January 1974, Bingham; H. B. Robinson May 1975; Indian Point 2 July 1977) where pump seal failures resulted in significant loss of coolant to the reactor building. All cases caused cleanup problems and loss of plant availability.

Interviews With Pump Engineers

Pump engineers note that the major problems with pump seals are not those represented by the events described above but rather are short seal life and inconsistency of seal performance. The engineers also note a lack of in-depth understanding of seal performance and a lack of ability to predict seal performance. Although not clearly illustrated in our data, pump balancing can also have some impact on plant availability. A discussion of these two items follows:

- Pump Seals — B&W operating plants use RC pumps from three pump manufacturers: Westinghouse, Bingham, and Byron-Jackson. The records show that seals from each of these manufacturers have problems. The following comments apply to RC pump seals used in B&W plants.

Working conditions for repairing RC pump seals are severe: high temperatures, poor lighting, and cramped quarters. Although the work request historical data show slightly higher numbers, we estimate that an average of about four men are required for 74 hours each to replace the seals in one pump. Also, our 1977 data show that the average pump seal work involved 7 man-rems of exposure per plant.

The frequency of inspection and maintenance/replacement of pump seals should be decreased because of economic considerations, impact on refueling critical path time, and high man-rem exposure. Also, proper seal performance is dependent on adhering to sensitive installation techniques. If such techniques are not followed, seal performance may be inadequate. Improvements are needed in this area.

- Pump Balancing — Balancing may be required after replacement of seals and/or motor bearings. This operation has caused some loss of availability in the past. With improved procedures, pumps can now be balanced in about 2 hours (average) per pump. We estimate that the equivalent of 6 to 8 EFPH are lost during each fuel cycle due to pump balancing.

  Effort has been given to reducing the need for balancing by careful alignment procedures, which has reduced the availability loss resulting from pump balancing. Emphasis on reducing the frequency of seal replacement is recommended to minimize pump balancing. (See section 4.2.1.4 for further discussion of balancing pump-motor combinations.)

Analysis of Refueling Outage Data

Refer to section 4.3.3.

Seal Injection Filters

Filters are used on the seal injection water supply lines to the RC pumps to prevent suspended matter (mainly crud) from damaging the seals. During plant startup from refueling operations, the filters have plugged up very rapidly, requiring frequent replacement of the filter elements in the presence of high radiation levels at Oconee and TMI-1. The high maintenance frequency has affected the plant startup schedule. It is proposed that a study be conducted to identify the specific causes of the problem and to develop other filtration techniques to eliminate the problem.

Conclusions and Recommendations

The historical data indicate that pump seal problems may be due to design considerations or the entry of foreign material. Pump seal leakage may also contribute to containment activity and increased waste disposal requirements.

A comprehensive pump seal improvement program should be implemented to support and complement other programs now underway. The program objectives should be to develop seals that have a longer life expectancy and low gas release. To realize the maximum benefit and to define the design improvement program, current knowledge of seal performance should be expanded by collecting additional performance data. More information is needed in the following areas:

- Seal leakage design criteria relative to anticipated plant transients need to be improved, optimized, and better understood.
- Further understanding of seal performance should be obtained by continuous monitoring of seal performance. Parameters to be evaluated include pump thrust (radial and axial), seal displacement, vibration, and temperature response.
- Methods to reduce or eliminate seal failure due to foreign debris, including crud, should be identified.
- The relationship between seal leakage, crud retention, and containment radioactive gas levels needs further study. This could be a source of xenon/iodine levels documented in the current Oconee operating data. Refer to section 4.6.3.
- The impact of seal leakage on radioactive waste reprocessing and waste disposal needs to be better understood. Refer to sections 4.2.7.1 and 4.2.7.2 for discussions of liquid and gaseous waste disposal.

In mid-1975 a program to upgrade the Bingham pump seals was intiated. The objective of the program is to develop predictable seal performance under both steady-state and transient conditions. The program includes evaluation of the seal design, laboratory qualification, and installation of a prototype in an operating pump. Following assessment of prototype performance, a seal upgrade program with the utilities is planned. Parameters such as seal leakage, seal cavity pressure, and seal outlet temperature will be evaluated for application to other installed Bingham pumps and to design concepts of other vendor pumps. To provide a data base for the prototype seal evaluation and to increase the understanding of the seal operating environment, a program of data accumulation from all B&W operating plants is underway.

Regarding working conditions for seal maintenance, balance-of-plant (BOP) criteria based on pump vendor accessibility requirements are supplied to utilities, so that the utility can include these provisions in its plant design.

4.2.1.4. RC Pump Motors (1D)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	335	3
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	109	4
LFM (Oconee 1 hist. data, Table 4-2)	2061	2

Analysis of Oconee 1 Opeational Historical Data (See Table E-1)

The Oconee 1 historical data show seven events totaling 21 separate actions on Westinghouse RC pump motors that affected plant availability. The following events are shown for each of the four motors:

- Three oil changes.
- Preventive maintenance.
- Installation of spare coolers.
- Clean oil pots.
- Collect oil samples (three motors only).

The following events are shown for one motor:

- Confirm indicated bearing temperature.
- Repair oil lift system.

Exclusive of other causes, 887 EFPH were lost over the 3½ years of data coverage because of these events.

Analysis of Oconee 1 Maintenance Historical Data (See Table E-1)

The 1974-1977 Oconee 1 data for all RC pump motor maintenance, including those events that did not cause power reductions, may be classified as follows:

Activity	No. of events	Manhours
Collect oil samples, change oil concurrent with preventive maintenance	28(27%)	2654(37%)
Motor preventive maintenance	4(4%)	1264(18%)
Thrust runner modifications	12(11%)	1152(16%)
Clean oil pots, coolers, vents	20(19%)	940(13%)
Oil lift system (piping, motors, filters)	14(13%)	374(5%)
Vibration (checks, corrective action)	6(6%)	132(2%)
Miscellaneous	21(20%)	697(9%)

These events were almost equally divided among the four RC pump motors. A breakdown of work events by year follows:

1974: 19 events (2276 manhours)*

1976: 43 events (1350 manhours)

1976: 4 events (∿214 manhours)

1977: 39 events (3373 manhours)

*July through December 1974.

Sixteen of the events included equipment modification, 48 were for preventive maintenance, and 41 were for repair or corrective action. During the Oconee 1 1977 refueling outage, the motor upper bearing thrust runner pumps were changed from a centrifugal to a viscosity type. This modification, along with other scheduled preventive maintenance, required 2968 of the 3373 manhours used during 1977.

Analysis of Oconee 1, 2, and 3 1977 Operational Data (See Appendix D)

The two pump motor events that impacted plant availability were on Unit 3. During a startup in June 1977 a high motor bearing temperature resulted from inadequate cooling of the bearing due to improper valve alignment. No damage was found, but startup was delayed about 2½ days.

The second event occurred in October 1977 when Unit 3 power was reduced to three-pump operation because of a low oil level on RC pump motor 3B1. We report a total of 326 EFPH loss of Unit 3 in 1977.

Motor Lubrication System

Most of the problems with the RC pump motors involve the lubricating oil system. The problems are common to pump motors supplied by Westinghouse, Allis-Chalmers, and General Electric in B&W operating plants. The designs of the different pump motor manufacturers are similar in that they require that the upper bearing oil reservoir level be maintained within an inch or two of a constant level. If the oil level is raised due to overfilling, oil foaming, turbulence, etc., the oil will spill over the upper radial shaft bearing standpipe and into the motor stator housing. If the oil level falls due to vaporization, leakage, etc., the upper shaft bearings will not be properly lubricated and could be damaged.

Therefore, oil level alarms are used to protect against high and low oil levels. Although the oil reservoir holds 150 to 200 gallons of oil, of which only 10 gallons is actually needed for lubrication, no variations greater than an inch or so in oil level are permitted. Snce the oil level sensors typically have a level sensitivity of only 0.5 inch, little margin exists for oil level variation. The plant must be shut down to permit access to the RC pump motors to add oil. Several fixes have been attempted, such as using more accurate level detectors, with varying degrees of success, but the oil level problem has not been fully solved.

The second lubricating oil problem is with leaks in the lower oil pan. The pan is a split design to permit installation around the pump/motor shaft. Installation of the gaskets between the pan halves is difficult, and these gaskets often leak. This is the most common maintenance problem on the RC pump motors. Oconee and other B&W plants have built deflectors or shields to prevent gasket oil leakage from falling on the hot RC pumps. This is an example where improved gasket material and a better gasket design would alleviate the problem.

Anti-Rotation Devices

Another problem with Allis-Chalmers RC pump motors is the failure of the Marland Clutch anti-reverse rotation devices. Two such devices failed in August 1973 on Arkanses Unit One with one pump rotor being damaged to the extent that it had to be sent back to the factory for repairs. Improvements were made to the anti-reverse devices at that time.

In March 1978 another Marland Clutch (with A-C motor) failure occurred at TMI-2, resulting in minor damage to the rotor shaft, while completely destroying the anti-reverse device. This failure was attributed to a temporary sticking of the device due to particulate contamination in the lubricating oil.

Discussion, Conclusions, and Recommendations

The objective of programs now underway is to reduce the oil level sensitivity. This includes identifying and correcting leaks. For operating plants, the program includes a review of vendor design disclosures, inspection of vendor-unique motors, preparation of recommendations, and equipment modifications. Some corrections identified to date include the following:

- The oil level problem in the upper bearing oil reservoir could be corrected with a remote control system for adding oil, but this would require reliable controls and level sensing.
- Installing a displacement-type device which would raise the oil level when required — since adequate oil remains in the reservoir when a low level is indicated, the suggested device would raise the oil to the desired level by forcing a flotation buoy into the oil. Such a device would require accurate and reliable level controls. Such a system may be feasible on both operating and future plants and could justify its costs in terms of improved plant availability. On future plants, a redesign of the pump motors to use a continuous-flow oil system for lubricating the shaft bearings may be a solution.
- For the future B&W reactors, the motor oil sensitivity problem has been resolved by increasing the allowable level fluctuation to approximately 4 inches. Also, the motor oil reservoir includes a 6-inch overflow line to accommodate a heat exchanger pipe break or an overflow problem. A remotely installed oil separator takes the overflow and retains the oil in the system.
- The leaking oil pan gasket problem can be alleviated by better gasket material, improved gasket design, and stiffer flanges. Gasket material and design are a general area of study with significant potential benefits for the industry.
- The anti-reverse rotation device failure problem has been addressed. The solution is a Formsprag (Dana Corp.) device, which has individually acting anti-reverse rollers on sprags rather than having these devices contained in and actuated by a single common carrier.

Although the balancing of RC pumps, done subsequently to pump seal repairs, is not considered a signficant availability limiting consideraiton, balancing of pump/motor combinations is an area where additional study is needed to understand the effects, interrelationships, and solutions to the pump/motor question. This program would collect data to evaluate the effects of pump/motor balance on seal performance and bearing wear and to recommend unbalance limits and improved balancing methods. Areas of concern include (1) hydraulic imbalance, (2) motor mechanical center versus electrical center, (3) couple imbalance, (4) moment imbalance, and (5) motor load versus unload effects.

4.2.1.5. Piping (1E)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-8	0	--
LFM (Oconee 1 hist. data, Table 4-2)	0	--

Conclusions

No historical operating, maintenance, or refueling data for the B&W-manufactured reactor coolant piping was identified. Thus, availability is not affected.

4.2.1.6. Steam Generator (1F)

Study results	Limiting factor	Rank	
LFO (Oconee 1 hist. data, Table 4-1)	733	1	
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	1119	1	
LFM (Oconee 1 hist. data, Table 4-2)	1606	4	
LFR (four-plant avg, Table 4-4)	268	1	OTSG tube inspection and repair

Analysis of Oconee 1 Historical Data

The 3½ years of data for the B&W-manufactured steam generator show a total of 12 events, nine of which impacted plant availability. Of these nine events, one involved an instrument leak, one involved a valve packing leak, and the other seven involved steam generator tube leaks (five in 1977 and two in 1976).

The instrument and valve leak repairs required 65 EFPH. The tube leak repairs required 2500 EFPH (1500 in 1977 and 1000 in 1976). Of the remaining three events which did not impact plant availability, one was to remove special test instrumentation (100 manhours), and the other two were to stop gasket leaks (40 manhours).

Analysis of Oconee 1, 2, and 3 Operational Data

The 1977 data for the three Oconee units show a total of nine events that impacted plant availability, i.e., five on Unit 1, one on Unit 2, and three on Unit 3. All involved leaking tubes as follows: eddy current testing (609 EFPH), trying to identify an elusive leak (896 EFPH), and plugging tubes including the required shtudown, cooling, etc. time (1852 EFPH) — a total loss of 3357 EFPH for 1977. This total includes the 1500 EFPH that is also included in the Oconee 1 historical data above.

Analysis of Refueling Outage Data

The four-plant refueling outage study identified steam generator inspection and repair as the largest limiting factor for refueling (LFR). The magnitude of this LF (268) was strongly influenced by the Oconee 1 efforts to locate and plug leaking steam generator tubes.

The data for TMI-1 also contributed LFRs for the steam generators as a result of precautionary tube plugging done while the plant was down for a feedwater pump repair.

Survey of Steam Generator Tube Leaks Vs Availability

Because of the importance of tube leaks, the number and frequency of leaks in B&W steam generators are reviewed here. Prior to mid-1976, B&W steam generators had performed with no leaking tubes. The number and frequency of steam generator tube leaks at the three Oconee plants are shown graphically in Figure 4-1. The failure rate decreased significantly after plant and operational modifications were made. These modifications include reductions in the severity and frequency of the turbine stop valve testing, improvements in feedwater chemistry by changes in system operating procedures, and closing the auxiliary feedwater nozzle that had directed water into the open tube lane. Present data do not give enough information to identify any one item or any particular combination of events as the cause for steam generator tube leaks.

In addition to those above, two steam generator-related shutdowns were identified: The first concerns the multiple shutdowns of Oconee 2 in late 1977 while trying to locate a leaking tube; 1164 EFPH were lost in repeated attempts to locate this tube. (The leaking tube was located and removed from service in January 1978.) The second event involved Oconee 2 in July 1977, when a file was accidentally dropped into a steam generator during a tube removal sequence. Four hundred EFPH were lost as a result of this maintenance incident.

Radiation levels at the OTSG manway opening on Oconee 1 have been as high as 2 rem/hour and in the range of 15 to 20 rem/hour at the upper tubesheet (2). During 1977, the average exposure per unit at the three Oconee units was 73 manRem for locating and plugging tubes and 20 manRem for eddy-current testing (see Table 4-11).

Discussion and Conclusions

Steam generator tube leaks, leaking tube identification, and removal of leaking tubes from service are the primary reasons for the unavailability and potential unavailability of the steam generator and thus the NSS.

- Tube Plugging — To locate and remove a leaking steam generator tube the plant must be shut down, cooled, and depressurized. On the average, 131 EFPH were required for these "additional" activities each time the plant was shut down due to a leaking steam generator tube. Table 4-5 gives additional-time data for the seven tube-plugging events at Oconee 1.

  The critical path times for the Oconee 1 OTSG A and B tube plugging events amount to almost 21 days (∿215 + 293 hours) of lost plant availability. This lost time is directly related to the number of tubes that were inspected, i.e., 16% (1491) tubes for generator A and 33% (5125) tubes for generator B. The planning for the Oconee 1 refueling outage allowed 120 hours of non-cricial path time for inspection of 3% of the total tubes in each generator.

- Eddy-Current Inspection — Forty-two tubes were plugged in the two generators during the 1977 Oconee 1 refueling outage because of indicated tube anomalies. Attempts were made to understand the nature of the tube condition causing the indications, but the attempts were often unsuccessful and decisions were made to plug the tubes rather than extend the shutdown or risk a future tube leak with a resultant forced outage.

- Current Status and Programs Underway — The loss of availability of nuclear plants because of steam generator tube leaks is an industry problem. EPRI, the utilities, and the PWR manufacturers have a comprehensive ongoing program to resolve the tube leak problem. These programs are aimed at understanding design and operational factors affecting tube integrity and recommending modifications to improve tube integrity.

No tube leaks have been reported at B&W plants other than Oconee, but changes in system designs and operating procedures have been made. Tube inspections by eddy-current methods have been performed as required, and a limited number of tubes have been plugged as a precautionary step.

Recommendations

EPRI, the utilities, and the reactor vendors have comprehensive programs aimed steam generator tube integrity. From this study, several additional recommendations are made for activities to emphasize and supplement existing programs.

1. Programs of analysis and test should be initiated to develop better methods to quantify leak severity by improving correlations between leak rate and the secondary radioactivity level. Development of new or improved instrumentation as well as equipment or analytical methods to interpret the data should be included. A successful program could eliminate shutdowns to find a small leak that may be difficult to locate after the plant has cooled.

2. A test and evaluation program should be initiated to develop better ways of locating leaking tubes. The objective is to minimize the time to locate the leaking tube.

3. Evaluate the problem of radioactive gas entrapment in the reactor coolant system to minimize the resultant possible release to the containment during subsequent venting.

4. Develop automated eddy-current data interpretation and alternate inspection techniques to minimize the total time for tube inspection.

5. Present procedures require plugging of leaking tubes from the top and bottom. A program to develop equipment and procedures that would allow plugging of both ends from the top would shorten the repair time and lessen the exposure of inspection and repair personnel.

4.2.1.7. Pressurizer (1G)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	40	12
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	65	8
LFM (Oconee 1 hist. data, Table 4-2)	156	21

Analysis of Oconee 1 Historical Data

The Oconee 1 historical data for the B&W-manufactured pressurizer show a total of 24 events for the 3½ years of data. Of these, only one event (a leaking valve) impacted plant availability and cost 140 EFPH. Twenty-three of the total were related to pressurizer valves. These 23 events cost a total of 536 manhours of maintenance. The pressurizer valve problems may be classified as follows:

Events	No. of events	Manhours
Internal repairs	6(26%)	152(28%)
Replacement	4(17%)	234(44%)
Operator	3(13%)	14(3%)
Packing	3(13%)	52(10%)
Other	7(30%)	84(16%)

Analysis of 1977 Oconee 1, 2, and 3 Operational Data

The 1977 data show two events that impacted plant availability. Both involved work on pressurizer valves and cost a total of 200 EFPH.

Discussion, Conclusions, and Recommendations

The B&W pressurizer system comprises three components that could impact plant availability and/or plant maintenance: the vessel, the valves, and the heaters. Our data show no problems with the pressurizer vessel, negligible problems with the pressurizer heaters, and some problems with the valves.

Since these data were obtained, leaking heater bundle gaskets have been observed at two units, which caused loss of availability. This condition resulted from radial O-ring gasket motion during heatup and cooldown.

Except for the pressurizer sample valves (RC-5, RC-6, and RC-7) all the valves identified in this pressurizer study are also identified as key valves and are included in the key valve study. Section 4.4 is a report of this key valve study and of generic conclusions regarding other (non-key) valves. Refer to sections 4.4.6 and 4.4.7 for a comparison of pressurizer valve limiting factors relative to valves in other systems.

Except for heater bundles and valves no recommendations are made regarding the components in the pressurizer system. Recommendations regarding pressurizer valves are given in section 4.4.4.

4.2.1.8. Core Physics and Reactor Operation (1H)

Study results	Limiting factor	Rank	
LFO (Oconee 1 hist. data, Table 4-1)	307	4	
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	313	3	
LFM (Oconee 1 hist. data, Table 4-2)	0	--	
LFR (four-plant avg, Table 4-4)	70	6	Physics tests
	5	14	Health physics survey

Analysis of Oconee 1 Opeational Historical Data

Power Ascension — Five events (four in 1977) were reported in which power increases were temporarily halted because of power ascension restrictions. These power level "holds" were between 65 and 75% full power and resulted in an average of 2.4 EFPH loss per event.

Xenon Hold — A Technical Specification "hold" on power escalation exists at the "power level cutoff" of 90% to ensure that xenon is close to its full-power equilibrium value and that xenon buildup/burnout has nearly stabilized prior to escalation above 90% full power. This restriction ensures that power peaking does not exceed the more restrictive power peaking limits assumed in the core analysis. The data also show that power escalation was temporarily halted at 85 to 95% full power on 46 occasions because of the xenon Technical Specification hold. The average power hold of nearly 24 hours resulted in about 2.5 EFPH loss per event, or a total of 4.8 full-power days.

Restart Physics Tests — Restart physics tests are performed after each refueling to ensure that measured core physics parameters are in agreement with the values assumed in the safety analyses for the fuel cycle. In 1975 the restart physics tests took 509 hours with a loss of 363 EFPH; in 1976 they took 298 hours (loss of 196 EFPH); and in 1977, 383 hours (loss of 265 EFPH). The 1975 tests took longer than usual because they were first-of-a-kind tests for both B&W and Duke Power Company, occurring after the Oconee 1 first refueling. The 1977 tests took longer than anticipated because of an unexpected quadrant power tilt; additional tests were performed to confirm and define the tilt.

Analysis of Oconee 1, 2, and 3 1977 Operational Data

Power Ascension — The 1977 data for the three Oconee units show that each plant experienced four power level "holds" because of the power ascension restrictions discussed above with an average loss of 1.9 EFPH per event.

Core Power Tilt — The Oconee 2 data show two events in which power was restricted to 96% full pwoer because of differences between measured and predicted core power distributions in two fuel assemblies. This program resulted in a 4% full power loss for about 147 hours in 1977 in Oconee 2. (The restriction was lifted in 1978 after sufficient burnup in cycle 3 had occurred to reduce the measured peak to below acceptable values.)

Although quadrant power tilts have been indicated on several occasions, no significant power generation losses were identified prior to cycle 4 at Oconee 1. Earlier quadrant power tilt indications were traceable to malfunctioning incore detectors or computer problems. However, during startup physics testing on cycle 4 at Ocone 1, a quadrant power tilt was observed. Power was held at 75% FP while the problem was investigated. The cause was found to be nonuniform burnup in cycle 3, and the problem was resolved by making additional core power distribution analyses, which allowed a Technical Specification revision. This single event cost Oconee 1 almost 102 EFPH of power generation. The data are reported as three events because of intervening unrelated power outages.

Analysis of Refueling Outage Data

Refer to section 4.3.3.

Conclusions and Recommendations

Fuel Maneuvering — Restrictions are placed on startup and power ascension rate following extended low-power operation as a conservative measure to protect fuel integrity. The time and EFPH losses can be reduced by limiting the power holds to the minimum time recommended, i.e., 5 hours. The real cure to losses in EFPH due to pellet-cladding interaction (PCI) concerns can only come from a clearer understanding of the pellet-cladding interaction problem and/or design changes to reactivity controls to minimize the causes of the PCI concerns.

The exact mode of PCI-induced failure is not well known and, while it has not been a significant problem in PWRs at present maneuvering rates and burnups, continued R&D is needed to study the mechanisms of PCI-induced fuel cladding failure and to quantify the suspected causes of PCI so as to determine fuel design features that eliminate or minimize these concerns. Existing R&D activity relative to the BWR problems should be monitored for application to PWRs. Improvements in incore detector performance would decrease the possibility of lost time due to core power distribution problems.

Core Power Tilt — At this time, the number of cases in which a real tilt has occurred is not sufficient to require additional R&D work.

Xenon Hold — The length of xenon hold is a function of the reactor control mode, i.e., either rodded (Oconee 2 and 3) or feed-and-bleed (Rancho Seco). Oconee 1 switched to the feed-and-bleed mode in cycle 4. The Rancho Seco Technical Specification allows continued power escalation after a 2-hour power hold between 87 and 92% full power with the reactor operating in the feed-and-bleed mode (rods out). Our data show that on the average, Rancho Seco lost half as much generation per startup due to the Technical Specification xenon hold as did Oconee 2 and 3. The loss of electric generation due to the xenon Tech Spec hold could be lessened by shortening the actual or required power level hold time. This can be accomplished in any of the following ways:

1. Change the principal mode of operation of the rodded Oconee-type plants to a feed-and-bleed control like Rancho Seco and obtain the corresponding change in Tech Spec wording.

2. Develop a more sophisticated on-line or off-line computer program readily accessible to the plant operator which would accurately calculate transient xenon based on power history and compare it with the Tech Spec requirements. This would minimize unnecessary precautionary-type delays in escalation.

3. A less accurate alternative to item 2 would be a set of tables or curves to predict xenon worth for numerous presupposed power histories.

4. Modify the reactivity controls in such a way that the power peaking restrictions on xenon which are the basis for the Tech Spec xenon hold are no longer necessary.

Restart Physics Tests — Both the 1975 and 1977 restart physics tests on Oconee 1 have been identified as being longer than usual. The 1976 restart physics test times on Oconee 1 are fairly typical, but we believe that these "typical" times could be shortened by improved planning, scheduling, and the like. We conclude that the restart physics test program is minimal in relation to the number of tests to be performed and see no immediate possibility of deleting tests. If sufficient test results can be obtained and analyses performed on the power imbalance-detector correlation test, it might be possible to eliminate this test during the initial power escalation following each reload. This would save about one day of restart time. This test is now and probably would still be repeated at 75% FP after the new fuel has been conditioned at 100% FP for a week. We also believe that the test time could be shortened if core power distribution test comparisons could be made at non-equilibrium xenon conditions. This would require the use of computer codes to more accurately predict core power distributions at non-equilibrium xenon conditions. To be useful such codes would have to have input/output access by long-lines from the reactor sites to processing centers. The development and testing of the linking of such computer codes with the sites is an area for potential improvement. See section 4.3.3 for additional information on reducing physics test time after refueling.

4.2.2. Auxiliary Fluid Systems (2)

4.2.2.1. Makeup and Purification/HPI (24)

The primary function of the makeup and purification system is to provide high-pressure injection makeup water and primary system "cleanup" water to the RC system during normal reactor operation. A second function of the HPI pumps in this system is to provide high-pressure injection water to the RC system during the early stages of a loss of coolant accident.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	59	9
LFM (Oconee 1 hist. data, Table 4-2)	613	11

Analysis of Oconee 1 Historical Data

1. Valves
   a. The data show no valve failures that resulted in plant shtudown.
   b. Fourteen of the 75 valve repairs (19%) were made on valves HP-26 and HP-27 (HPI valves). These are 4-inch Rockwell-Edward globe control valves with Limitorque SMB-1-25 operators. Nine of the

repairs were to replace packing, four to repair an operator, and one to repair a valve canopy ring. See Appendix E for additional failure information.

c. Valve problems in this system are classified as follows:

Events	No. of events	Manhours
Repacking	36(48%)	148(27%)
Operator repair	24(32%)	93(17%)
Valve repair	15(20%)	300(55%)

d. The maintenance record for valves in this sytem compared to valves in other systems is given in sections 4.4.6 and 4.4.7.

2. HPI Pumps and Motors (see System 2A, Appendix E)

a. The data show no pump or motor failures that resulted in plant shutdown.

b. Of eight motor repair incidents totaling 85 manhours, 40 manhours (47%) were used to remove, repair, and replace one Ingersoll-Rand pump motor. This required a power reduction to 70% power. Three items to change oil in the motors and bearings used 15 manhours (17%) of the total maintenance time. These were accomplished during refueling.

c. Of the seven Ingersoll-Rand pump repair incidents totaling 524 manhours, 117 manhours (22%) were used to complete one pump seal flow test. Inspecting seals (one incident) required 135 manhours (26%). Removing, repairing, and replacing one pump (one incident) required 128 manhours (24%). None of the pump repair items required a power reduction.

d. The LFM for pumps/motors in this system compared to other systems is given in section 4.6.3 and Table 4-13.

3. Letdown Filters

Four events required 28 manhours. Three of the events (21 manhours) were required to change or clean filters. The remaining event was related to checking a vent hose and seal.

4. Letdown Coolers

One event to install a new letdown cooler required 450 welder and 520 maintenance manhours (total 970 manhours). (See also section 4.6.4.)

5. Letdown Orifice — No data reported.

6. Seal Return Coolers — No data reported.

7. Letdown Storage Tank — No data reported.

8. Purification Demineralizers — No data reported.

9. RC Pump Seal Injection Filters — No data reported.

Analysis of 1977 Oconee Data

The 1977 Oconee data show four events that affected power operation. Sixty-three EFPH were lost on Unit 1 during feed-and-bleed operations; 12 EFPH was lost on the same unit replacing a Velan valve in the makeup system. A seal injection line on an HPI pump had to be replaced on Unit 2 and cost 93 EFPH. The single event on Unit 3 was the refilling of the purification demineralizer with resin at 95% power (see Appendix D).

Discussion, Conclusions, and Recommendations

Discussions with Duke Power Company personnel identified two areas for improvement on the makeup and purification (MU&P) system. They feel that better maintenance access to the numerous valves in this system would be helpful, particularly in the letdown storage tank room. Second, improvements need to be made in the handling of ion-exchange resins and letdown filters. Replacement of the large letdown filters now in use at Oconee with smaller filters would facilitate changes and reduce radiation exposure and disposal problems.

One reason for problems with valves in the MU&P system is the use of a light-weight valve bonnet, which allows body/bonnet leakage and subsequent corrosion of carbon steel body/bonnet studs by hot boric acid. Another cause of valve problems in this sytem is the use of a valve to control letdown flow. This mode of operation has required frequent valve internals replacement because of vibrational wear due to the large pressure drop across the valve. This problem is discussed in section 4.4.4.10.

A major cause of HPI pump failure is operation of the pump with improper valve lineup, i.e., failure to open either the suction or the discharge valve. An interlock/controller to protect the makeup pumps could reduce these failures. Such a device has been developed and is commercially available.

Primary system water chemistry, particularly the presence of high chlorides, has been a problem. The source of the chlorides has not been clearly identified although a number of possible sources have been suggested, such as (1) the ion-exchange resins, (2) the makeup water, (3) the hydrazine, and (4) the reclaimed boric acid. A study to identify and correct the sources of high chlorides in the primary system water could save substantial startup time, especially following refueling and other major outages when high chlorides cause the most trouble. Section 4.6.2 contains additional discussion of the problem of high chloride levels.

4.2.2.2. Decay Heat/LPI System (2B)

The decay heat system recirculates primary system water and removes decay heat when the RC pumps and the steam generators are not in operation. The decay heat pumps also provide low-pressure water injection to the RC system during a loss-of-coolant accident.

Study Results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3, 1977 data, Table 4-3)	42	11
LFM (Oconee 1 hist. data, Table 4-2)	65	26

Analysis of Oconee 1 Historical Data

1. Valves — Of 27 valve events, none caused a shutdown or extended an outage. Fourteen valves required repair or maintenance once, four required repair or maintenance twice, and one required repair or maintenance five times (three problems with operators, two with packing). Valve problems are categorized as follows:

Events	No. of events	Manhours
Valves needed repacking	10(37%)	48(30%)
Problems with electric operators	9(33%)	44(27%)
Valve replaced due to flow vibr'n	3(11%)	36(22%)
Packing needed tightening	3(11%)	20(12%)
Miscellaneous	2(7%)	14(9%)

Additional information on valve maintenance in this system compared to maintenance on valves in other systems is given in sections 4.4.6 and 4.4.7. Section 4.4.4.12 contains additional information on valve packing materials.

2. Pumps — Three pump events were reported in the maintenance data; two were gage problems, and one involved a seal (54 manhours). See section 4.6.3 for an analysis of pump/motor problems by system.

3. LPI Coolers — No events reported.

4. Borated Water Storage Tank (BWST) — Two events were reported in the maintenance data: one valve alignment and one level indicator transmitter calibration (14 manhours).

Analysis of 1977 Oconee Data

Three events are reported in the Oconee 1, 2, and 3 current data on the decay heat/LPI system. Replacement of a valve on the LPI system followed by performance of the LPI engineering safety test resulted in the loss of 96 EFPH on Oconee 2. The temporary repair of a leak in the extraction piping and replacement of an Ingersoll-Rand LPI pump on Oconee 3 in 1977 due to failure of a mechanical seal cost 30 EFPH.

Discussion, Conclusions, and Recommendations

High flow vibration on butterfly valves LP-12 and LP-14 on the decay heat cooler discharge has been reported at Oconee 1. Replacement of these valves with Fisher modulating flow control valves designed for this type of application may have solved the problem.

Valve leakage (packing and/or body to bonnet) in the LPI system is a possible source of primary coolant water leakage. Since primary coolant leakage must be controlled and held to a minimum, valve repacking is more commonplace in the LPI system than in others. Because of this, wider use of packless valves in this system should be considered.

Another area of concern is that operating plants have only one decay heat "drop line" from the RC system. The valves on the drop line must be tested periodically and, because of the need to have the decay heat system operational during fuel movement, LPI maintenance must be scheduled before and/or after fuel movement. On future B&W plants, the decay heat drop line splits upstream of the valves,and parallel lines with double valves will provide test capability during operation of the decay heat system. Duke Power personnel suggest that the drop line valves be packless.

Maintaining water chemistry specifications during shutdown has also been identified as an area of concern. Most B&W plants, including Oconee 2 and 3, have cross connects on the decay heat system to allow a portion of the RC system primary water to be run through the normal letdown filters and ion-exchange resins in the makeup and purification system. Duke plans to backfit this capability to Oconee 1. More complete use of this capability may help in maintaining chemistry specifications during shutdown.

4.2.2.3. Chemical Addition and Sampling System (2C)

The chemical addition and sampling system serves to help keep the primary system water within specifications by providing a means of adding the necessary chemicals.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	16	16
LFM (Oconee 1 hist. data, Table 4-2)	73	25

Analysis of Oconee 1 Historical Data

Nineteen of the 23 maintenance events involved pumps (see Appendix E, system 2C). Analysis of the data revealed the following:

1. Pumps

   a. Of the 19 work requests on pumps in this system, 11 were for the hydrazine pump (104 manhours) and seven were for the high-pressure boric acid pump (104 manhours). The other work request was for the lithium hydroxide pump (6 manhours).

   b. The pump problems can be categorized according to the repair made, as follows:

Event	No. of events	Manhours
Diaphragm replacement	6(32%)	66(31%)
Replacement or adjustment of relief or check valve	4(21%)	68(32%)
Oil leak repair	3(16%)	22(10%)
Miscellaneous	6(31%)	58(27%)

   c. The pattern of work requests indicates that the reported problem was repaired, but developing problems were not identified and fixed at the same time. Of the 19 work requests, nine were followed within a month by an additional work request on the same pump, usually for a different problem.

   d. The total manhours devoted to work on the key pumps in this system is 214 manhours in 3½ years, and none of the work required a power reduction.

   e. Section 4.6.3 and Table 4-13 give additional information on the performance of pumps/motors in this system compared to pumps/motors in other systems.

2. Tanks, Valves, and Mixers

   In two cases, the mixer shaft had to be replaced, and once the propeller fell off and had to be replaced. The single reported event on tanks was the adjustment of a relief valve on the boric acid storage tank. No other tank or valve problems were reported in the data for this system.

Analysis of 1977 Oconee Data

A review of 1977 data for Oconee 1, 2, and 3 reveals one case for Unit 3 where the chemical addition and sampling system is designated as the cause of a power reduction attributed to high chloride concentration. See sections 4.2.2.1 and 4.6.2 for a discussion of the high chloride problem.

Discussion, Conclusions, and Recommendations

The pumps could present a potential problem in the chemical addition system. If a pump diaphragm should rupture, oil could be pumped into the chemical system and eventually into the RC system. A high chloride problem in the concentrated boric acid storage tank (CBAST) on Unit 3 may have been caused by pumping oil into the CBAST from a pump with a ruptured diaphragm. We suggest that the key pumps in this system be replaced by double-diaphragm or plunger-type, positive displacement pumps. Pump "failure sensors" might also be considered.

Our data do not identify the cause of the mixer shaft and propeller problems on Oconee 1; either concentrations or temperatures outside the system design limits can cause the shaft to be twisted as the agitator tries to mix liquids that are too viscous. Rapid addition of boric acid crystals to the tank can cause a layer of undissolved crystals to form on the bottom of the tank, which would also cause the propeller and/or shaft to break. Care must be exercised to avoid adding crystals faster than they can dissolve, and the design limits on both temperatures and chemical concentrations should be strictly observed.

4.2.2.4. Spent Fuel Cooling System (2D)

The spent fuel cooling system circulates and cools the spent fuel pool water.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	8	40

Analysis of the 3½ years of data from Oconee 1 reveals the following:

Valves

Only one event was recorded for the valves. This event did not force or extend the outage (24 manhours to repair).

Pumps

Records indicate that only one event involving pumps occurred. This was minor and did not force or extend an outage (4 manhours to repair).

Discussion, Conclusions, and Recommendations

None.

4.2.2.5. Reactor Building Spray (2E)

The reactor building spray system sprays water into the containment following a loss-of-coolant accident to prevent containment overpressurization.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	10	19
OFM (Oconee 1 hist. data, Table 4-2)	12	39

Analysis of Oconee 1 Historical Data

The Oconee 1 historical data show no events for this system that forced or extended a power reduction. The six maintenance events were equally divided between valves and pump motor problems. The three reactor building spray valve problems were one blown fuse (1 manhour) and two valve packing leaks (8 manhours). The three RB spray pump motor events were one seal supply leak (16 manhours), one gasket replacement (8 manhours), and one motor inspection (8 manhours).

Analysis of 1977 Oconee Data

According to the 1977 Oconee power histories (Figure D-3 and Table D-1), Unit 3 lost 29 EFPH due to a single event on the reactor building spray system when the unit was shut down briefly to replace a reactor building spray pump that had a failed bearing. No events on this system were reported on Unit 1 or Unit 2.

Discussion, Conclusions, and Recommendations

None.

4.2.2.6. Core Flooding (2F)

The function of the core flooding system is to rapidly reflood the core following a loss-of-coolant accident. Flooding water is provided by water stored in the core flood tank.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	8	41

Analysis of the 3½ years of maintenance data is as follows:

Valves

Valve problems in this system are classified as follows:

Events	No. of events	Manhours
Repacking	1(25%)	10(45%)
Electrical operator problems	3(75%)	12(55%)

Flow Transmitter

Two events were recorded for the core flood system flow transmitter, one of which was a setpoint adjustment problem and the other a signal monitor calibration problem (4 manhours each).

Tanks

No work events were recorded for the core flood system tanks.

Discussion, Conclusions, and Recommendations

None.

4.2.2.7. Low-Pressure Service Water (2G)

The low-pressure service water system removes heat from intermediate coolers, such as the decay heat coolers, component coolers, RC pump oil and bearing coolers and transfers this heat to the heat sink.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	17	35

Oconee 1 historical data and the Oconee Unit 1, 2, and 3 data for 1977 show that this system has no record of forcing or extending a power reduction (Tables 4-1 and 4-3). Table 4-2 shows that the calculated LFM (17) is relatively unimportant (rank 35).

Discussion, Conclusions, and Recommendations

Since performance of this system has been essentially trouble-free, no recommendations are made for the low-pressure service water system.

4.2.2.8. Component Cooling (2I)

The component cooling system provides cooling to numerous electrical components during normal plant operation, including the control rod drive stators, intermediate and small pump motors, and backup cooling for the RC pump seals.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	97	24

Analysis of Oconee 1 Historical Data*

1. Valves

   None of the seven events caused a shutdown or extended outage time. Valve maintenance may be categorized as follows:

Events	No. of events	Manhours
Limit switch adjustments	2(29%)	29(67%)
Body-to-bonnet leaks	2(29%)	178(58%)
Seat leaks	2(29%)	102(33%)
Leak tests	1(13%)	8(3%)

- - - - -

*See Appendix E, system 2I.

See also sections 4.4.6 and 4.4.7 for additional information on valve problems.

2. Coolers

   The tubes in both component cooling water coolers were cleaned with air and water during the 1977 refueling outage (24 manhours).

3. Pumps/Motors

   One Ingersoll-Rand component cooling water pump motor was replaced in 1974 (8 manhours).

4. CRD Stator Cooling Pressure Switch

   One CRD stator cooling pressure switch required repair work in 1975 (7.4 manhours).

Discussion, Conclusions, and Recommendations

None of the events on the component cooling system forced or extended a power reduction or outage and, except for valve work, repair and maintenance on the component cooling system was not a significant limiting factor.

4.2.2.9. Penetration Room Ventilation and Reactor Building Purge (2J)

This system exhausts air from the reactor building and the penetration rooms.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	26	34

Analysis of the Oconee 1 maintenance data reveals that all the work requests available were written in late 1976 or 1977. Although work requests were probably written before late 1976, none could be obtained, and the data may be biased toward more recent problems. A total of 10 work requests were reported for the penetration room valves. In 70% of the cases, the valve diaphragm was replaced (71 manhours). The other 30% of the work requests were for miscellaneous valve repairs (21 manhours). See also Appendix E, system 2J and sections 4.4.6 and 4.4.7 for further detail.

Discussion, Conclusions, and Recommendations

Eight of the ten work requests were written against valve PR-2. This is a 48-inch Pratt butterfly valve. Most of the work was to repair or replace the valve diaphragm. This valve is located in penetration room 1 on the reactor building purge exhaust line. The high frequency of work on this valve suggests further investigation to identify and correct the cause of the problems.

4.2.3. Secondary Systems (3)

4.2.3.1. Main Turbine (3A)

At the Oconee plants, the General Electric main turbine comprises three low-pressure sections and one high-pressure section. At Rancho Seco, the Westinghouse main turbine consists of two low-pressure and one high-pressure section.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	41	11
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	5	22
LFM (Oconee 1 hist. data, Table 4-2)	3241	1

The Oconee 1 historical data show two events that impacted plant availability: a non-recurring vibration problem and a valve that malfunctioned during a feedwater pump turbine test and caused a turbine trip. These two events resulted in a loss of 142 EFPH in the $3\frac{1}{2}$ years of data. The 1977 operating data for the three Oconee units show one additional event — a defective mechanical trip solenoid on Unit 2 — which cost 14 EFPH. These few unrelated events result in a relatively low limiting factor for operation ranking for the main turbines at the Oconee units.

The Duke refueling outage data show that 3241 manhours were used for the 1977 Oconee 1 turbine/turbine valve overhaul work where one low-pressure turbine section was overhauled. These data were used as the basis for estimating the manhours required for the turbine maintenance work at Oconee (Table 4-2).

Additional turbine operational data were obtained at Rancho Seco. In May 1975, a required inspection revealed the turbines at Rancho Seco to be in satisfactory condition. Three months later, turbine vibration was detected and the unit was shut down and inspected. Eighteen rotor blades were found to be missing, and numerous turbine blade cracks were detected. An eight-month outage followed, during which major turbine repairs, including rotor replacement, were made. (See reference 9 for additional information on this turbine problem.)

Discussion

Main turbines normally have few operational problems. However, when turbine problems do arise, they can cause a major impact on plant availability. Routine turbine maintenance (overhaul) is normally performed during the "annual" refueling outage. This work requires considerable time and manpower and may on occasion become the refueling outage critical path.

During the refueling outage, most utilities disassemble, inspect, and overhaul on a rotational basis one of the main turbine sections (high-pressure or one low-pressure section). At Oconee this work is done by a Duke turbine maintenance crew, which specializes in turbine maintenance. Duke believes that this approach results in minimum time and manpower requirements. They estimate that normal turbine maintenance may be performed within 20 days and therefore should not impact the outage critical path. Other studies (1) have shown that if support facilities (laydown space, cranes, etc.) are inadequate and/or if improvements are made in the primary side critical path as expected, that turbine maintenance work could become the refueling outage critical path.

At Rancho Seco, because all turbine sections were overhauled during the 1977 outage, we were unable to make a meaningful estimate of the manhours or time required for normal turbine maintenance at that plant.

Conclusions and Recommendations

From this and other studies (1), we conclude the following:

- The manhour requirements and the possibility of turbine overhaul work impacting the outage critical path can and should be minimized (where practicable) by
  - Providing adequate site facilities (and components), including laydown space, spare parts, and jib cranes, and
  - Providing experienced supervisory overhaul personnel and adequate planning.

4.2.3.2. Main Steam (3B)

The main steam system consists of the pipes and valves that carry steam from the steam generator to the turbines and which vent excess steam.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	10	19
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	8	20
LFM (Oconee 1 hist. data, Table 4-2)	679	8

Of the 43 work events identified in the Oconee 1 historical data, two events (both associated with main steam stop valve tests) impacted availability. These two events resulted in a loss of 36 EFPH. An analysis of the 1977 data for Oconee 1, 2, and 3 (Tables 4-3 and D-2) showed that six events occurred that impacted plant availability. Of these, five were related to main steam stop valve tests. The total 1977 Oconee loss was 25 EFPH. The Oconee 1 historical data show two important maintenance areas — valve problems and pipe repairs. Valve problems for the $3\frac{1}{2}$-year period may be categorized as follows:

Events	No. of events	Manhours
Body-to-bonnet leaks	5(11%)	128(7%)
Repacking	8(19%)	92(5%)
Reseating	11(26%)	300(17%)
Maintenance (inspection, cleaning, etc.)	9(21%)	1073(62%)
Miscellaneous	10(23%)	133(8%)

Additional information on valve maintenance in this system compared to valve maintenance in other systems is given in sections 4.4.6 and 4.4.7. Pipe repairs were mostly related to the inservice inspection and resultant pipe repairs during the 1977 refueling outages.

Duke operations personnel add the following comments to the data above:

- Seat, packing, and/or body-to-bonnet leaks in main steam valves can result in steam cutting of valve internals.
- Valve leaks can heat the surrounding area and activate the fire alarm system.
- Valve accessibility is poor and should be improved, especially on future plants.
- Setpoints on relief valves must be checked during the refueling outage. This requires 2 to 8 hours of critical path time.

Discussion

From the start of commercial operation in mid-1973 until late in 1976, each main steam stop valve at Oconee was tested weekly. In late 1976 it was suggested that system transients caused by these weekly tests could have been related to the steam generator tube failures that were then starting to occur at Oconee. To reduce this possibility, steam stop valve tests were changed from a weekly to a monthly basis; power changes needed to implement the tests were also limited to a 10% per hour rate. During the 1977 refueling outage, further changes were made so that the tests could be made at 95% power instead of 65% power.

Conclusions and Recommendations

- Most valves in this system (main steam relief, condenser bypass, throttle, and governor) were studied in more detail in our key valve study. See section 4.4 for conclusions and recommendations on these key valves.
- Because of high maintenance and test requirements on the main steam valves, new plant designs should ensure good accessibility and serviceability of these valves. Possible interaction of these valves with other components, etc., in accordance with the discussions above should be considered.
- Valve seat leakage is a problem with many valves and especially for main steam valves because of the high pressure drops and the cutting action of steam. See section 4.4 for our recommendations regarding seat leak problems.
- Pipe inspection and pipe repairs may have a significant direct generic impact on plant availability. Comparison of our findings and those in other studies may indicate that this problem needs further study and/or improvements.

4.2.3.3. Feedwater System (3C)

The feedwater system takes secondary water from the condensate system and routes it through two turbine-driven feedwater pumps, two stages of high-pressure heaters (B and A), and then to the OTSG. Appropriate pipes, valves, controls, and other equipment support these main components. Each pump and heater stage has two components in parallel.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	13	17
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	66	7
LFM (Oconee 1 hist. data, Table 4-2)	952	6

Four non-recurring events contributed to the Oconee 1 historical limiting factor for operation — a pump flange leak, a drain pump repair, a feedwater pump trip, and a pipe repair. These four events cost a total of 46 EFPH for the $3\frac{1}{2}$ years covered by the data. The 1977 operating data for the three Oconee units show that a total of nine events (three per unit) contributed to the calculated 1977 LFO. Five of the nine events were feedwater pump trips, two were feedwater nozzle repairs, and two were miscellaneous repairs.

The Oconee 1 maintenance data show two components (valves and feedwater heaters) to be the primary contributors to the calculated LFM for the feedwater system. Valve repairs took 1700 manhours in the 3½ years covered by the data. Valve maintenance was responsible for 72% of the work requests and made up 51% of the calculated LFM. For the 3½ years covered, the data show 85 repair events for 47 valves (1.8 events per valve). Feedwater valve problems may be categorized as follows:

Events	No. of events	Manhours
Body-to-bonnet leaks	24(28%)	148(23%)
Seat leaks	20(24%)	207(33%)
Hinge pin leaks	9(11%)	99(16%)
Operator and limit switch problems	12(14%)	82(13%)
Stem packing leaks	4(5%)	31(5%)
Flange leaks	2(2%)	20(3%)
Other	14(16%)	47(7%)

Sections 4.2.3.4, 4.4.5, 4.4.6, and 4.4.7 include additional information on valve failures in this system and compare valve failures in this system with those in other systems.

The second component that significantly contributed to the LFM was feedwater heaters. Eleven percent of the work requests and 37% of the calculated LFM were for heater repairs, which required 1200 manhours in 3½ years. About half the heater problems were due to tube leaks. The other common problems included sight glass and baffle plate failures, gasket leaks, and piping leaks (see Table E-1, page 17).

Discussion of Results

At the Oconee units, feedwater heater tube leaks are a continuing problem. During the 1977 Oconee 1 outage, some 24 feedwater heater tubes were plugged. An additional 18 tubes were plugged in May 1978. Many of these tubes were plugged as a precautionary measure; an average of 10 tubes are plugged for each leaking one. Generally, feedwater tube leaks do not require plant shutdown since moderately high feedwater leak rates can be tolerated, and redundancy of components allows for isolating defective components with only a slight power reduction (about 1% reduction for loss of one feedwater heater. About 8 hours is usually required to locate a feedwater tube leak and another 4 hours is required to plug the leak. Although feedwater heater tube repairs can be made with only a power reduction, shutdowns may be required because of personnel safety or instrument damage due to escaping steam. The failure rate of the high-pressure heater tubes in the feedwater system is considerably higher than the failure rate of the low-pressure heater tubes in the condensate system. The data do not show why, but interviews indicate that higher temperatures and pressures in the feedwater system may be contributing factors. At Oconee the Baldwin-Lima-Hamilton high-pressure feedwater heater tubes are carbon steel, whereas the low-pressure heater tubes in the condensate sytem are stainless steel. This may also contribute to the higher failure rate in the high-pressure feedwater heaters. Section 4.2.3.8 has further discussions of tube failures due to crud.

Interviews showed that at Oconcee, most feedwater pump trips were caused by feedwater pressure swings. These have been due in part to a defective control valve, but there are other as yet unidentified contributing factors. Studies by Duke are underway to identify these other factors.

The Rancho Seco plant has had several reactor trips due to feedwater pump trips (nine trips during the first four months of 1977). SMUD personnel indicate that these trips were caused by improperly installed O-rings, which allowed the feedwater system pressure to drop, causing a feedwater pump trip. SMUD believes that their feedwater pump problems have been solved by correcting this installation problem.

Conclusions and Recommendations

Improvements should be made in feedwater valves and in feedwater heaters. We recommend initiating a more detailed feedwater heater study program. Areas for study might include baffle designs, material selection, layup water chemistry, impact of steam and chemical erosion on heater tubes and heater drains, analysis of feedwater flow distribution, and the relationship between flow conditions and tube failures. Such a study should evaluate the economic benefits (due to improved plant availability and reduced plant maintenance) achievable through improved designs and/or designs using higher quality material, i.e., stainless steel, against the costs of achieving and implementing these improvements. Such a study should consider the changes in water chemistry that have occurred or may occur in the forseeable future.

4.2.3.4. Condensate System (3D)

At the Oconee plants the condensate system takes secondary water from the condenser/hotwell and circulates it through hotwell pumps, polishing demineralizers, condensate steam air ejectors, condensate booster pumps, four stages of low-pressure heaters (C, D, E, and F), drain coolers, and then to the feedwater system.

Heaters C, D, and E and the air ejectors each have two units in parallel. The other components each have three or more units in parallel. Appropriate pipes, valves, etc. supplement these main components.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	29	13
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	49	10
LFM (Oconee 1 hist. data, Table 4-2)	674	9

The Oconee 1 historical data show two non-recurring events that impacted plant availability: a hotwell pump repair and a hotwell pump motor bearing problem. These two events cost a total of 102 EFPH for the 3½ years covered by the data. The 1977 operating data for the three Oconee units show that five events contributed to the calculated LFO. The major contributions were the Unit 1 hotwell pump repair listed above (10 EFPH) and three Unit 2 delays in returning to power caused by polishing demineralizers and/or by problems getting the condensate water chemistry within specifications (130 EFPH).

The Oconee 1 maintenance data show two components (pump/motor combinations and valves) to be the primary contributors to the calculated LFM. A review of the pump/motor maintenance data shows it to be of a routine and/or non-repetitive nature. It involved 410 manhours for 3½ years of operation.

Valve maintenance work required about five times as many manhours (1900 manhours for the 3½-year period) as did all other components in this system. Valve problems may be classified as follows:

Events	No. of events	Manhours
Body-to-bonnet leaks	9(13%)	812(45%)
Seat, other internal repairs	21(31%)	426(24%)
Packing leaks	5(7%)	120(7%)
Flange leaks	2(3%)	92(5%)
Operator repairs	1(1%)	6(3%)
Other, including valve inspection and replacement	29(44%)	322(18%)

Discussion of Results

The data show 67 repair events for 43 valves (1.6 events per valve) for the 3½ years represented by these data. There are no indications of high failure rates on any particular valve(s). Relative to the total valve problems in each system, body-to-bonnet leaks of valves in the condensate system are half as frequent as those in the feedwater system (section 4.2.3.3). The relative frequency of packing leaks, however, is not appreciably different in the two systems. The higher incidence of body-to-bonnet leaks in the feedwater system may be due to the fact that both temperatures and pressures are about two times higher in the feedwater system. See sections 4.4.5, 4.4.6, and 4.4.7 for additional analysis of valves in this system.

Conclusions and Recommendations

Based on our data, we conclude that no components in the condensate system except valves need further attention. Any improvements made in feedwater system valves should be considered for applicability in the condensate system. (See also the additional generic valve analysis and conclusion given in section 4.4.)

4.2.3.5. Condenser Circulating Water (3E)

The function of the condenser circulating water system is to transfer heat from the condenser to the heat sink.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	35	30

The events that resulted in component maintenance are primarily preventive maintenance and are relatively unimportant.

Discussion, Conclusions, and Recommendations

At plants other than Oconee, condenser leaks have resulted in power reductions and plant shutdowns. This loss of plant availability could be reduced if quick and accurate methods were available for identifying leaking condenser tubes. A study and development program is recommended to develop methods that would reduce the impact of condenser leaks on plant availability.

4.2.3.6. Recirculated Cooling Water (3F)

The recirculated cooling water system removes heat from such subsystem coolers as the pump seal, oil, and bearing coolers; spent fuel coolers; water sample coolers; and the like.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconnee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	7	42

Conclusions

No events were identified wherein this system forced or extended a power reduction. The events that resulted in component maintenance are primarily preventive maintenance and are relatively unimportant. No improvement programs are suggested.

4.2.3.7. Auxiliary Steam (3G)

The auxiliary steam system receives steam from an auxiliary boiler or from operating units to drive feedwater pump turbines, turbine steam seals, condensate steam air ejectors, condenser water heaters, and the like during plant startup.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	0	--

Conclusions and Recommendations

No events involving loss of reactor power or component maintenance were identified. Discussion with Duke Power personnel confirmed that this system has not been and is not anticipated to become a plant availability problem. No recommendations are offered.

4.2.3.8. Moisture Separator and Reheater (3H)

Oconee 1 has four moisture separators, each of which contains two stages of reheaters. The reheaters are fed by extraction steam from the high-pressure turbine. Four drain pumps pump effluent from the moisture separator and reheaters.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	∿0	25
LFM (Oconee 1 hist. data, Table 4-2)	184	19

The data show that the General Electric moisture separator/reheaters have had essentially no impact on unit availability, but the system required 644 manhours of maintenance during the 3½ years of Oconee 1 data (see Table E-1, page 21). The moisture separator/reheater maintenance work may be classified as follows:

Events	No. of events	Manhours
Manway leaks	15(83%)	208(32%)
Impingement baffle replacement	2(11%)	432(67%)
Miscellaneous	1(6%)	4(1%)

During the 1977 refueling outage, leak detectors, erosion inspection equipment, and improved seal ring manway gaskets were added to the moisture separator/reheater.

Discussions with Duke personnel show that there have been numerous baffle plate repairs and tube leaks with loss of low-pressure turbine output and, hence, lowered electrical generation efficiency. Five (out of eight) of the GE moisture separator/reheater tube sections on Unit 2 are scheduled for replacement during the 1978 refueling outage. Except for two experimental sections, the original carbon steel tubes are being replaced with carbon steel. The experimental sections will contain one each of the following materials:

- 405 stainless steel
- 439 stainless steel
- 444 stainless steel (18% Cr, 2% Mo)
- E-Brite material (26% Cr, 1% Mo)
- Nickel-plated carbon steel

Manway leak problems were solved by welding the manway covers in place. Duke is now working with the vendor to develop improved manways.

Study Results (Rancho Seco)

SMUD interviews and reference 10 show that, during the construction phase, many plant components were either in site storage or in place and partially constructed.

During this period considerable oxidation occurred on carbon steel components, including the moisture separator/reheater and feedwater heaters. Subsequent chemical cleaning of the unit resulted in ∿9000 kg (∿20,000 lb) of iron being removed and an unknown amount of oxide being left in the system. Subsequent efforts to maintain the system under suitable chemical conditions were subverted due to system changes, repairs, etc.

Subsequent failures/repairs attributed to this experience occurred in moisture separator/reheater tubes and baffle plates and in feedwater heater tubes. Residual oxides also caused slow, difficult system cleanup; each power escalation step resulted in crud burst and consequent difficult removal by polishing demineralizers. This and many other similar experiences illustrate the importance and the difficulty of maintaining appropriate environmental control during plant construction and/or appropriate system cleanup after construction but before plant operation begins. Many control rod drive vent valve failures (section 4.2.4.1), for example, are known to be caused by primary system crud introduced during plant construction. General valve seat leakage (section 4.4.5) has been attributed primarily to foreign material in the valves. It is likely that much of this foreign material originated during plant construction. We believe that a study to determine ways to effectively maintain appropriate environmental control of components during plant construction and to effect system cleanup prior to plant operation would help to minimize this crud problem.

Recommendations

We recommend a more detailed study program to identify technical and economic considerations for improved moisture separator/reheater performance. Areas for possible study include the following:

- Improved manway gasket materials, manway design, and/or installation procedures to minimize leaks.
- Re-evaluation of baffle plate service requirements.
- Water/steam chemistry as it relates to crud buildup, heat exchanger material interaction, and release of crud into the RC system.
- Programs to determine ways to effectively maintain appropriate environmental control of components during plant construction and to effect system cleanup prior to plant operation.

As noted above, some of these improvement programs are currently underway.

4.2.3.9. Generator Stator Cooling (3I)

The function of the generator stator cooling system is to remove heat (by way of a stator cooling water and a hydrogen system) from the electrical generator stator windings.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	3	22
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	18	15
LFM (Oconee 1 hist. data, Table 4-2)	35	31

The limiting factors calculated from the Oconee 1 historical data (operational and maintenance) are negligible. The 1977 operational data for the three Oconee units show one event for each of the three units — low pump pressure, a cooling water controller problem, and a blown gasket for Units 1, 2, and 3, respectively. These three combined events cost 54 EFPH.

Conclusions

We conclude that improvements in the generator stator cooling water system can best be realized by improvements to minimize leaks in seals, gaskets, flanges, and the like.

4.2.3.10. Heater Drain System (3J)

The heater drain system includes the valves, tanks, pumps, and pipes that carry drain water from the feedwater heaters and moisture separator/reheaters to the main condenser.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	11	18
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	27	14
LFM (Oconee 1 hist. data, Table 4-2)	626	10

Oconee 1, 2, and 3 operational data show two events on Oconee 1 responsible for loss of plant availability. One was due to a malfunctioning valve operator; the other was for repair of a cracked drain line. (These two events cost 55 EFPH; see Figure D-1.) No events were identified for Unit 3 that impacted plant availability, and two events occurred on Unit 2. The latter were a minor valve problem and a minor power reduction, while a heater drain pump shaft and bearings were cleaned and repacked. Although the power reduction for the pump repairs was only 2%, it extended for 20 days and cost 25 EFPH (see Figure D-2). No additional availability limiting events were identified in the Oconee 1 historical data. More than two thirds of the work events in the Oconee 1 historical maintenance data were due to valve repairs (see Table E-1, page 22). Valve problems in this system may be classified as follows:

Events	No. of events	Manhours
Body-to-bonnet leaks	19(28%)	514(34%)
Seat and other internal repairs	17(25%)	460(30%)
Flange leaks	14(21%)	232(15%)
Packing leaks	6(9%)	102(7%)
Other problems, including valve operator and valve replacement	11(17%)	204(14%)
Total	67	1512

Section 4.4 gives additional information on valve failures. The following categories apply for non-valve components. (See Table 3-1, pages 23 and 24.)

- 71% of the pump events were on the oil system (changing oil, oil pump malfunction, and oil leaks).
- Tank problems are due to leaks (73%, primarily at flanges) and to malfunctioning level detectors (27%).
- No cooler events were reported.

Discussion

From interviews with plant personnel, the following opinions and additional information were obtained:

- Components (especially valves and valve operators) in this system undergo almost continuous service and are exposed to high temperatures and pressures; consequently, frequent breaks occur, and adjustments are needed frequently.
- Because of redundancy, a component failure in this system usually does not impact plant availability.
- During initial operation, the Bailey Meter level control system for the heater drain tank caused feedwater flow perturbation. Duke has changed to a Foxboro control system, which has a wider range and faster instrument response.

- Considerable problems have been encountered with sight glass level detectors. Duke is removing these sight glasses from service.
- Better level detectors, level control systems, and level control valves are needed.

Conclusions and Recommendations

Improvements in this system should be directed toward reducing maintenance. Because of the conditions imposed on components in this system, special consideration should be given to gaskets, valves, valve operators, and level detectors.

4.2.3.11. Instrument Air (3K)

Study Results

No Oconee 1 historical work requests identified with the instrument air system impacted plant availability or involved significant maintenance. A review of the 1977 data for Oconee 1, 2, and 3 shows one minor event that impacted plant availability; this involved a broken air line and cost 6 EFPH. No improvement programs for this system are suggested.

4.2.3.12. Turbine Lube Oil System (3L)

At each Oconee plant, a common lube oil purifier system serves both the main turbine and the feedwater pump turbine. The main turbine loop has an oil storage tank, oil circulating pumps, and oil lift pumps. The feedwater pump turbine loop has an oil storage tank and an oil circulating pump.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	45	10
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	75	6
LFM (Oconee 1 hist. data, Table 4-2)	175	20

Three events contributed to the calculating Oconee 1 historical LFO:

- Repair of a main turbine oil leak in June 1976.
- Inspection and repair of the main turbine turning gear oil pump in March 1977.
- An unexplained operator error in June 1977 caused a low oil pressure turbine trip.

The 1977 data showed two events on each of the three Oconee units which impacted plant availability. A study of these events did not indicate a generic failure trend. The two Oconee 1 events are listed above. The other four events (on Oconee 2 and 3) were attributed to varied component repairs and inspections as follows:

- Unit 2 (Figure D-2) — Replaced breaker on emergency bearing oil pump; replaced turning gear oil pump motor.
- Unit 3 (Figure D-3) — Lapped flange and replaced gasket on a valve for the oil tank; replaced valve because of body-to-bonnet O-ring leak.

The Oconee 1 historical data show that the turbine lube oil system has required a relatively large number of maintenance events (35). Most were during 1975. The calculated LFM, however, ranks low (20). Had the 1975 frequency of repair continued during other years, the ranking would have been much higher. The 35 maintenance work requests identified in our data search are categorized as follows:

Events	No. of events	Manhours
Clutch and bearing repairs	10(29%)	82(29%)
Sleeve bowl bars and bushings	5(14%)	38(13%)
Brake repair	3(9%)	48(17%)
Heater problems	3(9%)	15(5%)
Miscellaneous repairs	14(40%)	102(36%)

Discussion, Conclusions, and Recommendations

Interviews with Duke personnel indicate that the data above may not give a complete picture and that problems were indeed present during 1977. From these interviews and from a study of the data, we have the following comments:

1. The lube oil purifier has sustained usage with high maintenance. During refueling outages, the lubricating oil is pumped to an outside oil storage tank to permit removal of residue from the purifier. Seventy-two hours are required for this operation, and 72 additional hours are required to pump the oil back. The return pumping is on the secondary side critical path and is sometimes done near the end of the refueling outage; therefore, it could impact refueling outage time.

   Technology now exists for oil purifiers that may relieve these problems. A program for trial installation of an improved purifier that was developed for use with diesel and gas turbine fuels is now being considered by Duke for applicability to nuclear plants. This program holds promise of improving availability. The importance of oil purifiers warrants redundancy considerations for single-unit plants. Multiple-unit plants should incorporate cross-connect features.

2. The General Electric-supplied main turbine turning gear lube oil lift pumps, the emergency oil pump test solenoid, and the test valves have had a high frequency of repair. These components undergo severe service during plant shutdown. The high-pressure, low-volume oil flow results in considerable pump gear (impeller) wear. This conclusion should be reviewed for generic applicability. If confirmed, we recommend a development program for improved turning gear lift pumps.

3. Oil leaks occur at the flange gaskets, switches, instruments, etc., resulting in loss of oil pressure and possible loss of feedwater pump turbine control. We recommend a development program to minimize leakage at gaskets. This recommendation is more meaningful as applied to other components, such as valves, but as shown here, it also applies elsewhere.

4.2.3.13. Turbine Electro-Hydraulic Control System (3M)

The turbine electro-hydraulic control (EHC) system controls the turbine main steam stop valves, control valves, and reheat/intercept valves. This system is supplied and in some cases serviced by the turbine manufacturer.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	19	15
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	34	12
LFM (Oconee 1 hist. data, Table 4-2)	100	23

Of the 10 work events identified in the Oconee 1 historical data, two impacted plant availability: an oil leak and a reactor trip due to low oil pressure. These two events cost 33 EFPH for the 3½ years covered. Of the four 1977 events recorded for Oconee 1, 2, and 3, two were for Unit 2 — one to modify the General Electric system to obtain slower main steam stop valve closing time, and the other to repair a broken hydraulic line (39 EFPH total). Unit 3 had two unexplained losses of 120 V dc power to the system, which cost a total of 63 EFPH. No loss of plant availability attributable to this system was identified for Unit 1 during 1977. The Oconee 1 historical maintenance data show 10 work events which cost 35 manhours for the period covered. The 10 events may be categorized as follows:

Events	No. of events	Manhours
System modification	4(40%)	220(66%)
Equipment inspection	3(30%)	59(18%)
Oil pressure problems	2(20%)	38(11%)
Calibration	1(10%)	18(5%)

Study Results — Rancho Seco

Rancho Seco personnel report that there are cases where "foreign material" in the EHC oil lines has caused system malfunctions. There are also cases at Rancho Seco where EHC problems prevented the main steam stop valves from cycling properly, which resulted in turbine/reactor trip. SMUD noted that Westinghouse has recently addressed these control problems, and recurrence is not expected.

Conclusions and Recommendations

There are several cases of oil leaks, broken hydraulic lines, and low oil pressure due to oil leaks. This is another example of the need for improved (leak-proof/break resistant) lines, pipes, tubes, gaskets, etc. We recommend development programs to effect such improvements. We have no recommendations regarding the Rancho Seco cycling problems.

4.2.3.14. High-Pressure Service Water System (3N)

The high-pressure service water system supplies fire-fighting water to the auxiliary building and cooling water to various components, such as the component cooling water pumps.

No events that impacted plant availability were identified from either the Oconee 1 historical data or the Oconee 1, 2, and 3 data for 1977. Only two maintenance events were identified from Oconee 1 historical data. One is unimportant, and the other was for plugging of a ruptured cooler tube. The two events required 48 manhours of effort for the 3½ years of data.

4.2.3.15. Nitrogen Supply System (3P)

This system supplies a nitrogen blanket for water storage tanks. Nitrogen gas is also used to fill the pressurizer and steam generator during plant layup. No events that impacted plant availability were identified from either the Oconee 1 historical data or the Oconee 1, 2, and 3 data for 1977. Valves were the only components in this system identified as having a small impact on station maintenance. Valve maintenance involved 182 manhours for the 3½ years of data. See section 4.4 for a generic discussion of valve-related problems.

4.2.3.16. Steam Drain System (3Q)

The steam drain system collects steam and steam condensate from the steam seals and other main steam components and routes it back to the main condenser and/or to liquid waste. The steam drain system, therefore, performs a function similar to the heater drain system; the steam drain system has fewer components.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	194	18

The Oconee 1 historical data show that of 57 work requests identified, all were written on steam drain valves; the problems are classified as shown below.

Events	No. of events	Manhours
Body-to-bonnet leaks	9(16%)	70(10%)
Seat and other internal repairs	32(56%)	342(50%)
Flange leaks	0	0
Packing leaks	2(4%)	16(3%)
Other problems, including valve operator and valve replacement	14(25%)	254(37%)

Comparison of these statistics with data on the heater drain system shows a much higher percentage of seat leaks and a much lower percentage of flange and body-to-bonnet leaks in the steam drain system. We find no significant differences in pressure, temperature, or service for valves in the two systems. Heater drain valves carry water, while steam drain valves carry moist steam. This may be a contributing factor. Further statistical valve analysis on a non-system basis is given in section 4.4.

Conclusions and Recommendations

Conclusions and recommendations for valves in this system are similar to those for the heater drain system valves:

- Steam drain valves should be carefully selected because of the service seen. Generic application guidelines for valves, gaskets, valve operators, etc. in this system should be established to ensure maximum compatibility with the environment and the service seen.

4.2.3.17. Vacuum System (3R)

The vacuum system maintains condenser vacuum during plant operation.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	17	36

Although no events that impacted plant availability were identified from either the Oconee 1 historical data or the Oconee 1, 2, and 3 data for 1977, Duke reports an estimated one or two load reductions per year per unit because of vacuum leaks. Valves were the only components in this system identified as having an impact on station maintenance. The type of problems encountered with valves in this system do not appear to be different from valve problems in general. See section 4.4 for a generic discussion of valve-related problems.

Conclusions and Recommendations

In addition to impacting plant availability, vacuum leaks create chemistry problems due to excess oxygen in the secondary water. Vacuum leaks also impact turbine efficiency due to changes in turbine backpressure. We conclude that improved methods of identifying and correcting vacuum leaks are needed and recommend that such a program be established.

4.2.4. Auxiliary Mechanical Equipment Systems (4)

4.2.4.1. Control Rod Drive System (4A)

Study Results

The following tabulation lists the CRD system and subsystem limiting factors and rankings (based on Oconee 1 historical data):

Components	Limiting factors LFO[a]	LFM[b]	Rank LFO	LFM
Drives 4A1	43.3	50	--	--
Stators 4A2	45.3	103	--	--
Absolute position indicators 4A3	83.0	138	--	--
Power and TC cables 4A4	54.1	85	--	--
Closure/vent system 4A5	22.3	12	--	--
CRD control system 4A6	108.0	70	--	--
Total — CRD system	356.0	458	2	13

Other system limiting factors and rankings are given below.

Study results	Limiting factor	Rank
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	434	2
LFR (four-plant average, Table 4-4)	53	6 Secure/reinstall CRDM

Analysis of Oconee 1 Operational and Maintenance Historical Data

Between July 1, 1974, and December 31, 1977, there were 74 events requiring repair or maintenance on components of the CRD system. Thirty-three of these 74 events also forced or extended a power reduction. These events are discussed by component (subsystem) below.

Drives (4A1)

Five out of nine drive-related events impacted plant availability and occurred in a 6-month period between June and September 1976. These extended power reductions resulted after the mechanisms had been subjected to a "ratchet trip." Each extension involved jogging the drive to obtain proper engagement of roller nuts and leadscrew, and to overcome additional friction from debris caused by the ratchet trip. This trip action is defined under Discussions, Conclusions, and Recommendations below. This type of event did not recur through the end of 1977. Lost power generation due to these five events was 152 EFPH. Of the four additional events, three were repairs made during a scheduled refueling outage, and one event did not require power reduction to repair.

- - - - -

[a]Table 4-1. [b]Table 4-2.

Stators (4A2)

Sixteen events are recorded for the stators. Four events (three for stator replacement and one for a stuck rod) resulted in a loss of 158 EFPH. The other 12 maintenance events included 11 stator replacements and one rotor maintenance item. Stators supplied as original equipment on the Oconee units were of an epoxy-encapsulated bifilar design. Failures occurred during startup and operation, with the mode being predominantly a winding short in the stator end turns. The epoxy-impregnated stators were replaced with a varnish-impregnated stator design.

Operational experience with varnish-impregnated stators shows increased reliability. All the Oconee units have now converted to varnish stators. A comparison of the operational history between the epoxy and varnish types shows that:

- The epoxy stators had an average of about 7.7 operational failures per reactor operating year (1 reactor year is 69 stators at 80% availability).
- The varnish stators have had less than 1.9 operational failures per reactor operating year.

The stator improvement program included development of a varnish-impregnated monofilar stator in addition to the varnish-impregnated bifilar stator discussed above. The monofilar design provides a physical insulation barrier between phases and around the end turns in addition to the varnish impregnation. The monofilar stators have completed 3570 stator-days of laboratory tests and over 3260 stator-days of reactor operation with no failures.

Absolute Position Indicators (4A3)

The absolute position indicators (APIs) demonstrated relatively high LFs for both operation and maintenance, primarily as a result of a single occasion when the APIs and cables were sprayed by borated water from a leaking CRDM closure assembly. This one event caused 20 PI tubes and cables to require cleaning and/or replacement. Of the 18 events related to the APIs, 11 occurred in 1977, and 15 of the 18 involved replacement of PI tubes and API cards. Nine events caused a loss of 291 EFPH, six PI tube replacements, one reset reed switch, one cleaning and repairing 20 tubes, and one repairing 20 PI tube cables.

Power and Thermocouple Cables (4A4)

Three events involving replacement of power cables and thermocouple (TC) cables required power reduction to 0% with a loss of 189 EFPH. Two events required replacement of one and 10 power cables, respectively, and one event required replacement of all TC cables. The power and TC cables were all replaced in February 1977 and had been in service since the start of commercial operation in July 1973.

Closure/Vent System (4A5)

Four of five events relating to the CRDM closure/vent system occurred in the first four months of 1977. One event extended a plant shutdown, and one event required power reduction to 50%. All events were single occurrences. Total power loss was 75 EFPH.

CRD Control System (4A6)

Of 23 events relating to the control system, 12 occurred in the first six months of 1975, and six of these forced or extended plant shutdown. Nine of the 23 events were to replace switches, two for control rod repatch, and the remainder were single-occurrence events. Of the 23 maintenance events, 10 forced or extended a power reduction with a loss of 378 EFPH. Since the first half of 1975, eight maintenance events have occurred — two to replace switches and the remainder for non-recurring events.

Analysis of Oconee 1, 2, and 3 Current Data

The 1977 data for Oconee Units 1, 2, and 3 indicate that all three units lost significant generating capacity in 1977 due to problems with components of the CRD system. Oconee 1 lost 29 EFPH due to position indicators (two events) and 51 EFPH for stator work (one event). Oconee 1 also had an additional outage event of 246 EFPH when Duke replaced a stator, repaired a closure assembly, replaced a PI tube, cleaned 20 PI tubes, replaced 10 power cables, repaired 20 PI cables, and replaced all the thermocouple cables.

Oconee 2 lost 381 EFPH due to stator repair or replacement (seven events) and an additional 241 EFPH in a single event where 26 power cables were changed, 12 stators were repaired, and 10 stators were replaced.

Oconee 3 lost 124 EFPH for a single event on the drives and an additional 180 EFPH for three events on the stators.

Discussion, Conclusions, and Recommendations

Engineering information has been combined with CRD system service histories to understand problems and to provide recommendations for improvements. This information is presented for the various components below.

Drives (4A1)

The CRD is a non-rotating, translating leadscrew driven by a rotating roller nut. When power is lost to the CRD stator, the two halves of the roller nut separate, and the leadscrew with control rod attached is allowed to fall under the influence of gravity. An intermittent loss of power to the CRD stator will cause the roller nut to disengage, allowing the leadscrew to start to fall. If power is restored before the control rod is fully inserted, the roller nuts will attempt to re-engage with the falling leadscrew (called ratchet trip), and metal debris may be formed. CRDMs may have to be exercised and cleaned following a ratchet trip.

Since the problems with the drives occur because of an intermittent loss of power from the CRD control system, modifications to the CRD control system have been developed to prevent restoration of power to the stator while the control rod is falling.

Stators (4A2)

The CRDM utilizes a water-cooled stator to provide the driving power for control rod motion. The initial water jacket design was "unsealed," and any water spillage from the top of the CRDM could "wet" the stators and potentially cause problems. Changes to the water jacket design on operating units have been made to "seal" the jackets, thus preventing the entry of water into the stator cavity.

Stator failures have been due to shorts in the windings, which cause loss of magnetic field resulting in a control rod drop. Since the control rod cannot be withdrawn, the reactor is usually shut down for repairs to avoid introducing asymmetric core power distribution.

Another potential cause of CRD stator failure is energizing the stators without component cooling water supply. This condition resulted in extensive stator damage at Rancho Seco during July 1975. As a result, all 69 stators were damaged and had to be replaced. An interlock device has been designed to prevent stator damage by prohibiting stator energizing without cooling water.

The failure rate of the new varnish stators described above is higher at Oconee than at the other B&W plants. Failures at Oconee are primarily due to moisture. Some O-rings used to seal between the CRDM housing and the ID of the stator display a significant compression set, allowing water to enter the stator winding cavity. The O-ring set was due to exposure to higher than anticipated operating temperatures. An alternate silicone O-ring material exhibits superior high-temperature performance.

At present it is necessary to drain the entire cooling water manifold before changing even one stator. Quick disconnects on the stator connections to the component cooling water manifold are being investigated. These connections would automatically seal off the open ends of the water lines and prevent spillage of water when changing or maintaining stators.

Absolute Position Indicators (4A3)

The API consists of a tube mounted on the exterior of each CRDM with a series of magnetic reed switches inside each PI tube. These switches are activated by a magnet attached to the CRD leadscrew. During the time period of this study, most of the PI problems at the Oconee plant resulted from the inadvertent entry of reactor coolant, which caused improper electrical connector contact resistance.

A runback on a faulty indication of an asymmetric control rod is no longer required by the Technical Specifications since the system can be monitored and controlled with the API in-limit switch indication and the relative PI system that is part of the CRD control system.

An alternate PI system using dual-channel reed switches and system redundancy is being developed for new plants. Because this new PI design would be very expensive to retrofit into the older plants, the drive manufacturer is also developing a device for repair of faulty PI switches with provisions for installation in the CRDCS control cabinet and without shutdown of the reactor.

Power and Thermocouple Cables (4A4)

The CRD, PI, and thermocouple (TC) cable performance is being enhanced by improved handling and installation procedures and the use of a more moisture-resistant connector that is now available. Several utilities are replacing their old TC cables with new cabling and connectors.

Closure/Vent System (4A5)

The CRD closure is a pressure boundary for the RC system and is sealed with a metal O-ring and a backup ethylene propylene O-ring. A reinstalled closure is not pressure-tested until the reactor is heated and pressurized. CRD closure seal failures impact the startup critical path. One such event occurred on Oconee 1 in February 1977 when both the metal O-ring and the backup ethylene propylene O-rings failed, allowing hot pressurized water to spray over adjacent CRD components and cables.

Valves are used to vent the drives as the reactor system is filled with water. Studies indicate that the leakage sometimes occurs as a result of either improper valve operation or debris/crud on the valve seating surface. This problem has been reduced by the development of a vent valve design that is more tolerant of crud and valve misalignment (see also section 4.2.3.8 for a discussion of crud introduced during construction).

Radiation Contamination Control

Working on the CRDs is a significant contributor to radiation exposure. The activity is primarily due to radioactive crud buildup in the CRD area. Various flushing methods to wash the crud out of the CRDMs have been investigated, but some areas are inaccessible for flushing and thus limit this approach. Ultrasonic bath techniques have been investigated for cleaning heads and drives removed for maintenance. However, no devices of the size and shape needed for this work have been identified for in-place cleaning. We recommend finding ways to decontaminate heads and drives, including improved ultrasonic or other cleaning techniques.

Studies of particulate and radiological composition of the RC system crud and its effect and moving and wearing parts in the CRDM are now underway. The results of these studies should be evaluated for applicability to other equipment exposed to the reactor coolant and subject to high maintenance during refueling.

4.2.4.2. Fuel Handling Bridges (4B)

Stearns-Roger fuel handling bridges (and associated components) move fuel assemblies and control assemblies within, into, and out of the core during the refueling outage. Bridges on operating B&W plants use one telescoping mast and control board for fuel assemblies and a second mast and control system for control assemblies.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	269	6
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	--	--
LFM (Oconee 1 hist. data, Table 4-2)	428	14
LFR (four-plant average, Table 4-4)	106	3

Of 42 Oconee 1 historical maintenance events on the fuel handling bridges, 17 extended refueling outages. Delays in the fuel and fuel component movements during refueling cause this to be ranked number 3 in the LFRs. The 42 maintenance events on the fuel handling bridges in the 3½-year time span are classified below.

Events	No. of events	Manhours
Fuel handling mast	8(19%)	82(5%)
Control rod mast	18(45%)	1042(70%)
Controls and interlocks	9(21%)	282(19%)
Trolley/bridge drive	6(14%)	93(6%)

During the 1974 refueling outage, a problem with the control rod mast grapple not properly engaging the control components required 708 manhours and approximately 47% of the total maintenance time. This generic problem was corrected by making a modification to all control rod grapples.

Discussion and Conclusions

Fuel assembly spacer grids can interfere with spacer grids of adjacent fuel assemblies during vertical movement of the fuel, particularly when the bridge or trolley is misindexed and/or the fuel assembly is slightly bowed. Because of this possibility, load-limiting features have been incorporated into the fuel handling bridges. These load limitations have caused periodic hoist cutoffs and, thereby, slower refuelings. Improved spacer grids (not yet in operation) and improved fuel handling procedures are expected to eliminate most lost time due to spacer grid hangup. Hydraulic hoses have failed due to exposure to the operating environment and have delayed refueling operations.

Recommendations

This and other studies have identified a number of recommended changes and improvements in the present fuel handling equipment. Briefly, these are as follows:

1. An automatic indexing system for the fuel handling bridges and trolleys.

2. An improved multi-function mast and controls with pneumatic action to replace the present two-mast systems.

3. Consider upgrades to the currently installed fuel handling equipment as follows:
   a. Optimized fuel handling load limit settings.
   b. Revised grappling limits (establish elevation "band" to accommodate growth of fuel assemblies).
   c. Stepping motors to provide precise indexing alignment adjustments.
   d. Permanent identification tags for electrical connections to simplify electrical maintenance.
   e. A series of limit switch improvements.
   f. Selsyn chain restraints and protective cover to protect the indexing system from loss of index.
   g. Faster control rod hoisting operation.

Further in-depth analyses of problem areas are underway.

4.2.4.3. Fuel Transfer Equipment (4C)

The fuel transfer equipment includes the Stearns-Roger-supplied "upending hardware" and the transfer tube(s) through which fuel components are moved into and out of the reactor building. Most plants have two transfer tubes.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	25	14
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	--	--
LFM (Oconee 1 hist. data, Table 4-2)	28	33

Of 12 events requiring repair or maintenance on Oconee 1 between July 1974 and December 1977, four extended refueling outages. During the 1976 refueling outage, problems with the upenders required 3 days to correct. This single event accounts for 73% of the total lost capacity time and 43% of the total maintenance time. Problems with air motors, limit switches, and hydraulic hose were the principal contributors to maintenance time. All of the events extending the refueling outages were caused by problems with the upenders. The 12 fuel transfer system problems are classified as follows:

Events	No. of events	Manhours
Transfer tube	1(8%)	4(2%)
Transfer carriages	1(8%)	24(14%)
Upenders	6(50%)	107(64%)
Controls and interlocks	4(34%)	33(20%)

Section 4.3 contains additional discussions on problems with this system.

Conclusions and Recommendations

We recognize that the fuel transfer equipment presents significant problems. The problems are being addressed, but solutions are not now at hand.

Interviews with utility personnel indicate that differential settling of the reactor building and auxiliary building causes alignment problems in the transfer systems and tubes. In many cases alignment problems have caused the drive chain to disengage from the sprocket, disabling the fuel transfer basket in mid-tube. Based on an analysis of the availability limiting factors, we recommend the following:

- Future plants should be built with at least two transfer systems since a transfer mechanism failure on a single-system plant during refueling operations can directly impact the refueling outage time.
- A means should be provided to make a full functional equipment checkout, including system alignment, prior to the start of fuel handling operations. In some cases this may involve system changes so that the equipment may be operated dry (prior to filling the transfer canal). This is an improvement that could be backfit to existing plants. There are other cases where underwater components (switches, etc.) tested to be satisfactory when the pool was unflooded but which subsequently failed when the pool (and component) were flooded.
- On future plants, more space should be designed around the upender mechanism to permit easier access for maintenance.

4.2.4.4. CRDM Service Structure (4D)

The CRDM service structure mounts on top of the reactor vessel head and serves to support the CRDM. It is a cylindrical structure with an open top and a base closed by the reactor vessel head. Considerable heat is generated within the structure by the reactor vessel and the CRDM. At Oconee this heat is removed by eight cooling fans located near the base of the service structure. Some other plants use cooling fans, and dome cool the structure by routing building ventilating air to the base of the structure.

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	--
LFO (Oconee 1, 2, 3 1877 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	13	38

Conclusions and Recommendations

Cooling of the service structure is an important function, and failure to do so could result in damage to the CRDM. The present cooling criteria appear adequate, but further study of temperature distribution to verify criteria is needed.

4.2.4.5. Suppressors and Hangers (4E)

Suppressors (snubbers) are devices that allow essentially free thermal movement during normal plant heatup and cooldown but restrict component movement during dynamic events such as earthquakes or loss-of-coolant accidents. Hangers are component support devices (usually pipe support).

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	18	16
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	--
LFM (Oconee 1 hist. data, Table 4-2)	482	12

The Oconee 1 historical data show one event in 1976 where eight Itt-Grinnell suppressors were replaced with similar units and which resulted in plant shutdown. This one event cost 63 EFPH. One additional work event, reported in the Duke annual operating report, was done to modify the main steam line suppressor to prevent deformation during a turbine trip. Lack of information on manhours and the number of restraints modified precluded the use of these data in our data sheets. Most of the maintenance during this 3½-year period was done to suppressor inspection and repair, but some 6% was due to work on pipe hangers. A study of the maintenance data shows that most suppressor work occurred during 1977 when the plant was down for other reasons. (See Appendix E, system 4E.) A review of the 1977 data for Oconee 1, 2, and 3 shows no suppressor (or pipe hanger) events that caused or extended a power reduction.

Discussion

Historically, snubbers in operating plants have had problems with hydraulic fluid leaks but have had little effect on plant availability. With the increased use of snubbers in safety-related areas, the NRC suppressor inspection requirements have become more stringent, greatly increasing the potential for affecting plant availability. Currently, Oconee Technical Specifications require that 10 suppressors (or 10%, whichever is less) be inspected at each refueling outage. In addition, visual inspections must be made of all suppressors at 18-month intervals if no defects are found; the interval becomes more and more frequent if defects are found and increasing up to a frequency of once per month if eight defects are found.

Conclusions and Recommendations

Because of the increased NRC requirements for suppressor inspection and the problems resulting from this inspection, we recommend a study to:

- Establish guidelines for snubber design, qualification, production inspection, and installation to improve reliability.
- Minimize snubber inservice inspection requirements and facilitate the accomplishment of those required.

4.2.5. Electrical Systems (5)

The following tabulation lists the calculated limiting factors and ranks of the subsystems that make up the electrical systems:

System/subsystem	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)		
Generator (5A)	8	--
Switchgear (5B)	0	--
Controls (5C)	0	--
Exciter (5D)	217	--
Transformer (5E)	0	--
Substation (5F)	0	--
Isol. phase bar (5G)	0	--
Batteries (5H)	0	--
Battery chargers (5I)	0	--
Total electrical (5)	225	7

System/subsystem	Limiting factor	Rank
LFO (Oconee 1, 2, 3 1977 Data, Table 4-3)		
Generator (5A)	8	--
Switchgear (5B)	0	--
Controls (5C)	0	--
Exciter (5D)	0	--
Transformer (5E)	0	--
Substation (5F)	0	--
Isol. phase bar (5G)	0	--
Batteries (5H)	0	--
Battery chargers (5I)	0	--
Total electrical (5)	8	21
LFM (Oconee 1 hist. data, Table 4-2)		
Generator (5A)	12	--
Switchgear (5B)	0	--
Controls (5C)	0	--
Exciter (5D)	611	--
Transformer (5E)	56	--
Substation (5F)	0	--
Isol. phase bar (5G)	0	--
Batteries (5H)	0	--
Battery chargers (5I)	18	--
Total electrical (5)	697	7

The Oconee 1 operating data show three electrical system events that impacted plant availability. The GE-supplied field breakers were pulled and cleaned in 1977 because of dirty contacts. This cost 23 EFPH. Two events are attributed to the exciter: one to replace bearings and re-grout the exciter base plate, and one to realign the system to prevent mechanical vibrations. These two events cost 379 EFPH. The Oconee 1 historical maintenance data show that 81% of this maintenance time was due to the one bearing replacement/baseplate grouting event mentioned above. Except for this event, the repair events of the electrical systems appear non-generic.

Study Results — Rancho Seco

At Rancho Seco, two problems of significance were experienced with the Westinghouse electrical generator. About December 1, 1975, a malfunction in the generator seal oil system caused a quantity of generator seal oil to be spilled in the generator, resulting in a shutdown for cleanup. Before the unit was returned to power, moisture was detected in the generator stator windings. Dryout was attempted by running the generator at reduced speed in what was intended to be a controlled temperature dryout mode. However, due to an incorrect interpretation of the indicated stator temperature, the stators were overheated and the insulation degraded. All stator coils were replaced. Except for about one month of operation in March 1976, these two generator events caused a plant outage for a period of about 10 months.

Conclusions and Recommendations

We conclude that the Rancho Seco problems discussed above are one-of-a-kind events and that the necessary changes to avoid recurrence have been made. No design/development changes in this system are suggested.

4.2.6. Controls and Instrumentation (6)

4.2.6.1. Control and Monitoring Equipment Systems (6A)

Study Results

The following tabulation lists the system and subsystem limiting factors and rankings (based on Oconee 1 historical data):

	Limiting factors		Rank	
	LFO[a]	LFM[b]	LFO	LFM
Integrated control system (6A1)	31	25	--	--
Non-nuclear instrumentation (6A2)	38	116	--	--
Incore detectors (6A3)	0	154	--	--
Computers	0	0	--	--
Total — C&M equipment	69	295	9	15

[a]Table 4-1; [b]Table 4-2.

Other limiting factors and rankings for the control and monitoring equipment systems (6A) are as follows:

Study results	Limiting factor	Rank
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	10	18
LFR (four-plant average, Table 4-4)	28*	9*

*Remove and install incore detectors.

Analysis of Oconee 1 Operational and Maintenance Historical Data

Integrated Control System (6A1)

ICS problems are classified as follows:

ICS problem	No. of events	Manhours
Repairs	3(30%)	31(35%)
Recalibration	2(20%)	4(5%)
Startup delay (no cause reported)	4(40%)	45(51%)
Miscellaneous	1(10%)	8(9%)

Seven of the 10 reported work events (71%, 76 manhours) caused shutdowns or extended outages.

Non-Nuclear Instrumentation (6A2)

NNI problem	No. of events	Manhours
Repairs	4(21%)	54(14%)
Replacement	9(47%)	316(81%)
Recalibration	3(16%)	10(3%)
Miscellaneous	3(16%)	10(3%)

Of 19 work events, 21% (69 manhours) caused shutdown or extended outages.

Incore Detectors (6A3)

None of the incore detector work events caused shutdowns or extended outages. Incore detector problems are classified as follows:

Problem	No. of events	Manhours
Repair work	2(50%)	214(40%)
Replacement	1(25%)	292(54%)
Miscellaneous	1(25%)	32(6%)

Computers (6A4)

No work events were recorded for the computer system.

Analysis of Oconee 1, 2, and 3 Current Data

Three events that limited plant availability were reported for the integrated control system, and one was reported for the non-nuclear instrumentation. The three ICS events were due to a single failure on each of the three Oconee units and cost a total of 14 EFPH. The events were not similar or related. The single NNI event was due to a pressure transmitter hydraulic leak on Unit 1. The event cost 17 EFPH.

Discussion, Conclusions, and Recommendations

Within the control and monitoring equipment system, the integrated control system (ICS) and non-nuclear instrumentation (NNI) have occasionally caused some loss of generating capacity because of equipment failures. These failures have formed no pattern, and the frequency of such failures is as expected in terms of the reliability of the many individual electronic components in the system and subsystems. Also, the repair times have not been excessive, and most events that impacted plant availability occurred during 1975 and to a lesser degree during 1976, indicating that system performance may have improved with time.

Incore detectors have caused no downtime but have required manhours during scheduled outages. The majority of the work at Oconee 1 involved replacing 32 depleted detectors during the 1977 refueling outage. Interviews with engineers show that although detectors do not have a record of significantly impacting plant availability, they do have a history of requiring some maintenance, and this results in personnel radiation exposure (see Table 4-1). More importantly, the engineers note that incore detectors have not reached their full potential of reliably indicating core power distribution. Early incore detectors had problems with high background readings and sheathing failures. Improved lead wire designs, changes in insulating materials, and improvements in manufacturing techniques have reduced these problems somewhat, but further improvements are still needed. Suggested development/study areas include the following:

- Better signal-to-power conversion, including effects of rhodium depletion.
- Sheath failure studies.

4.2.6.2. Plant Protection Equipment System (6B)

Study Results

The following tabulation lists the system and subsystem limiting factors and rankings (based on Oconee 1 historical data):

Study results	Limiting factors LFO[a]	Limiting factors LFM[b]	Rank LFO	Rank LFM
Nuclear instrumentation/reactor protection system (6B1)	1	60	--	--
Safety-related controls and instrumentation (6B2)	0	0	--	--
Engineered safety features actuation system (6B3)	0	0	--	--
Total—plant protection system	1	60	23	27

[a]Table 4-1; [b]Table 4-2.

The Oconee 1, 2, and 3 limiting factor for operation and ranking are given below.

Study results	Limiting factor	Rank
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	1	24

The limiting factors for operation and maintenance rank near the bottom of each category.

Analysis of Oconee 1 Operational and Maintenance Historical Data

NI/RPS (6B1)

Event	No. of events	Manhours
Repairs	1(17%)	42(20%)
Replacement	2(33%)	160(77%)
Recalibration	3(50%)	6(3%)

Three NI (out-of-core detector) events reported in the 1977 Oconee data were power level "holds" to calibrate the NI to heat balance power that occurred twice on Unit 1 and once on Unit 3. Two NI recalibrations were performed on Unit 1 in 1977.

Safety-Related Controls and Instrumentation (6B2)

No work events reported.

Engineered Safety Features Actuation System (6B3)

No work events reported.

Discussion, Conclusions, and Recommendations

The only events in this category reported in either the 1977 data for the three Oconee units or the historical data on Oconee 1 were on the NIs and the RPS. Neither of these is a significant problem. The requirement to calibrate the NIs to heat balance power costs a small amount of potential power generation.

A system could be developed for continuous automatic calibration of excore detectors. Such a system would improve the present heat balance calibration method.

4.2.7. Waste Handling (7)

4.2.7.1. Liquid Waste Disposal System (7A)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	0
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	30	13
LFM (Oconee 1 hist. data, Table 4-2)	195	17

Analysis of Oconee 1 Historical Data

The Oconee 1 historical data show no events where equipment failures in the liquid waste disposal system forced or extended a power reduction. A total of 58 maintenance events were reported between mid-1974 and the end of 1977. Twenty-nine of these events were on valves, four on evaporators, and 25 on pumps. (See also Appendix E, system 7A.) The valve problems are classified as follows:

Event	No. of events	Manhours
Seat repair	8(28%)	78(36%)
Diaphragm replacement	8(28%)	64(29%)
Bonnet-to-body leaks	4(14%)	18(8%)
Miscellaneous	9(30%)	59(27%)

These 29 work events occurred on 23 different valves. See also sections 4.4.6 and 4.4.7 for comparison of valves in this system with those in other systems.

The four work events on the evaporators were uncorrelated, miscellaneous events.

The 25 work events on 10 pumps are classified as follows (see also section 4.6.3 and Table 4-13):

Events	No. of events	Manhours
Gaskets, seals, packing	10(40%)	262(59%)
Coupling	3(12%)	12(3%)
Bearings	1(4%)	8(2%)
Miscellaneous	11(44%)	162((36%)

Our data show six work events on the Intergoll-Rand spent resin transfer pump LWD-P7 in 1975 and none since. The problems with this pump were solved when the rigid line with several 90° elbows was replaced by rubber hose having turns of larger radius to allow freer flow of the resins without clogging.

The data also show seven events (three seat events and four coupling events) on the miscellaneous waste evaporator resin pump WD-P42. This pump is in continuous use, and no special significance is attached to these failures.

Analysis of Oconee 1, 2, and 3 Operational Data

In 1977 three events were reported on Oconee units wherein power escalation was limited because no remaining liquid waste storage capacity was available to permit further boron dilution. Two of these events occurred on Unit 3, while the third was on Unit 1.

Discussion, Conclusions, and Recommendations

Units 1 and 2 were built with a shared liquid waste disposal system, while Unit 3 has its own system. The liquid waste disposal systems originally built at many early nuclear plants were underdesigned. Records show that on Oconee 1 during 1977, the liquid waste disposal system was overloaded on 96 days. Duke recognized this problem several years ago and built a supplemental "interim" liquid waste processing facility. The permanent facility is in the design stage and has not yet been built.

Oconee 1 historical data show no events that forced or extended a plant shutdown or power reduction due to equipment problems on the liquid waste disposal system. The maintenance work on the valves, evaporators, and pumps is to be expected, especially in view of the heavy load placed on the system (see Appendix E, system 7A).

We conclude that packless valves throughout much of the plant would reduce the radwaste by reducing packing leaks. We also suggest that drain, vent, and relief valves should have a means of detecting leakage and that drain and vent lines should be double-valved to minimize radwaste. Duke is backfitting to double-valve the drain and vent valves.

Duke reports that the existing Aqua-Chem evaporators are shut down for repair approximately 25% of the time. This does not show up in our documented data. They report both foaming and carryover problems, and evaporators designed to handle 15 gpm actually only handle 9 gpm. We conclude that better evaporator technology is needed, and this may be an area that needs further study and development.

Environmental requirements on liquid waste disposal are very stringent. Such requirements impact plant initial and operating costs, increase exposure to operational personnel and in some cases impact plant availability. We conclude that a study of the system and component design to identify more effective ways to meet these requirements would be desirable.

4.2.7.2. Gaseous Waste Disposal System (7B)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	0
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	0
LFM (Oconee 1 hist. data, Table 4-2)	143	22

Analysis of Oconee 1 Historical Data

From a study of the data, the following conclusions are reached:

1. The data show no failures that contributed to plant shutdown or power reduction.

2. Components:

   a. Valves — Valve problems are classified as follows:

Event	No. of events	Manhours
Electrical/adjust limit switch	9(35%)	38(20%)
Replace diaphragm	6(23%)	64(33%)
Repair operator	2(7%)	10(5%)
Miscellaneous	9(35%)	82(42%)

See Appendix E, system 7B and sections 4.4.6 and 4.4.7 for additional information.

b. Compressors — Eight of the 21 events on two Nash compressors involved rebuilding or repairing the pumps and compressors (132 manhours). The remaining miscellaneous items consisted of repairs or adjustments to control components and operational checks (110 manhours).

c. Transmitters — Four events to repair or recalibrate flow transmitters occurred in 1976 (29 manhours).

d. Gas Analyzer — One event to reset a circuit breaker was the only problem reported (2 manhours).

e. GWD Vent Header — Three of five events consisted of pressure gage checks (35 manhours).

Discussion, Conclusions, and Recommendations

Plant availability is not directly affected by this system. However, maintenance manhours for this system are fairly high and thus exert a peripheral effect on plant availability by utilizing personnel and tools that could be applied to correction of availability-limiting problems. System improvements could be effected by using stainless steel components in lieu of carbon steel, including an upgrade of the valves and compressors to stainless steel.

4.2.7.3. Solid Waste Disposal System (7C)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	0
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	0
LFM (Oconee 1 hist. data, Table 4-2)	0	0

Analysis of Data

No work events are recorded relating to solid waste during the period from July 1, 1974, through December 31, 1977. Solid waste disposal is a subcontract function at Oconee. No records were found relating to problems with solid waste handling or shipping.

4.2.7.4. Coolant Storage (7D)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	9	20
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	10	17
LFM (Oconee 1 hist. data, Table 4-2)	32	32

Analysis of Oconee 1 Historical Data

Only one event occurred that limited plant availability, and this is reported under Oconee 1, 2, and 3 operational data below. The historical data on the coolant storage system are classified as follows (also see Appendix E, system 7D, and sections 4.4.6 and 4.4.7 for additional information):

1. Valves

   a. Repairing one valve required that the plant be brought to hot shutdown conditions (valve CS-66).

   b. Seventeen events occurred on 14 valves without requiring power reduction. The valve problems are classified as follows:

Event	No. of events	Manhours
Replace bonnet and diaphragm	1(6%)	4(5%)
Replace diaphragm	10(56%)	45(61%)
Miscellaneous repairs, repacking, electrical, and cleaning	6(38%)	25(34%)

2. Pumps — Three problems (all in 1977) with the Ingersoll-Rand pump in this system required two seal replacements and one wiring correction (32 manhours).

Analysis of Oconee 1, 2, and 3 Operational Data

The 1977 Oconee power histories show only a single event on the coolant storage system that affected power operation. Unit 1 lost 31 EFPH when a shutdown was required to replace a bonnet and diaphragm in a Grinnell valve CS-66 (previously mentioned). No events were reported for Units 2 or 3.

Discussion, Conclusions, and Recommendations

No specific changes in this system are suggested. Valves and pumps that require more than routine maintenance should be upgraded.

4.2.7.5. Coolant Treatment System (7E)

Study results	Limiting factor	Rank
LFO (Oconee 1 hist. data, Table 4-1)	0	0
LFO (Oconee 1, 2, 3 1977 data, Table 4-3)	0	0
LFM (Oconee 1 hist. data, Table 4-2)	292	16

Analysis of Oconee 1 Historical Data

No events were reported on the coolant treatment system either on Oconee 1 historical data or for Oconee Units 1, 2, or 3 in 1977, that resulted in a loss of plant availability. The historical data show 83 maintenance events, which are discussed below (see also Appendix E, system 7E).

1. Valves

   Of the 61 reported events on 16 valves, 28 were reported on one Fisher governor valve (CT-28), and 21 of these were to clean or "unclog" the valve. System modification in late 1975 corrected this problem (see Discussion, Conclusions, and Recommendations), and no problems have been reported with this valve since the correction. Valve problems are classified as follows:

Event	No. of events	Manhours
Replace diaphragm	13(31%)	121(33%)
Clean	28(46%)	180(49%)
Miscellaneous repair	12(20%)	57(15%)
Replace valve	2(3%)	10(3%)

See also sections 4.4.6 and 4.4.7 for additional valve analysis.

2. Piping

   Two events required "unclogging" pipe lines (26 manhours).

3. Evaporator

   Of six events reported for the evaporator, one to clean the unit required 88% of the total maintenance manhours (360 of 408 manhours).

4. Pumps

   Of the 14 events reported for four pumps, one item — replacing a shaft, seals, bearings, and coupling on one pump — required 36% of the total pump outage time. The breakdown of pump problems is as follows:

Event	No. of events	Manhours
Couplings	4(29%)	32(15%)
Replace motor and pump	3(21%)	40(18%)
Misc. rebuilding	2(14%)	96(44%)
Seals	1(7%)	8(4%)
Miscellaneous	4(29%)	44(20%)

   See also section 4.6.3 and Table 4-13 for analyses of pump problems.

Discussion, Conclusions, and Recommendations

Plant availability was not affected by this system. Valve repair events have been reduced by redesigning the coolant system. Of the 61 valve events reported in our historical data, only five occurred in 1976 or 1977; all the other 56 events occurred in 1974 or 1975.

The system was modified to eliminate dumping floor and laundry drains in with the reactor coolant for waste processing. The revised system contains strainers and filters to trap mop strings from the floor drains and trash from the laundry drains, which were previously clogging the valves and pipes.

4.2.8. Other (8)

4.2.8.1. Polar Crane (8A)

Study Results

During the 1977 Oconee 1 refueling outage, polar crane repairs caused 17 hours' delay on August 8. During the 1977 Rancho Seco refueling outage, the following delays were attributed to polar crane repairs: 16 hours on August 21, 19 hours on August 22, 11 hours on August 25, and 6 hours on August 26 (see Appendix F). B&W

refueling engineers reported that several parts on the polar crane failed during the start of the 1978 TMI-1 refueling outage, causing significant delays.

From the Oconee 1 historical data, we have calculated a limiting factor for operation of 6 and a limiting factor for maintenance of 52. These values and their ranking with other limiting systems are given in Tables 4-1 and 4-2. We consider the best measure of the importance of polar crane problems is the number of hours of critical path delay time encountered, not the number of manhours or clock hours involved in correcting the problem.

Discussion

The reactor building polar crane is an important component during refueling outages since it is the only crane available to lift heavy components such as the reactor vessel head, reactor internals, and equipment being moved into the containment. During the start of the refueling outage, polar crane activities are often on the critical path. Thus, a breakdown in the polar crane during this critical period often results in direct delays in the refueling outage. The data show that such delays are not uncommon.

Conclusions

From interviews and the data described above, the following conclusions are reached:

- Polar crane checkout and preventive maintenance cannot be started until after the refueling outage shutdown. When unexpected problems arise during checkout, refueling outage critical path delays follow.
- Polar crane equipment, like fuel handling equipment, stays unused in a hostile environment for one year. A common problem is electrical failures due to switch contact oxidation. To a lesser degree, there are problems with gear box leaks and brake malfunctions (Rancho Seco, 1977 refueling outage).
- Demands on polar cranes should be minimized by increased use of specialty jib cranes.

Recommendations

Based on the results and conclusions above, we recommend the following:

- Establishing, in conjunction with utilities and crane manufacturers, improved preventive maintenance and spare parts inventory guidelines, including identifying, if possible, ways to ensure polar crane operability prior to the critical path need time. Redundance of critical components should be considered.
- Establishing generic guidelines for jib crane installation. These guidelines should address economic advantages of having specialty cranes available when needed to minimize demands on the polar crane against the economic penalty of buying, installing, and maintaining additional equipment. These guidelines should be made applicable to new plants and operating plant retrofits.
- Identifying, through a special study, ways to minimize the effects of the hostile environment on polar cranes and fuel handling equipment. Suggested areas of study are
  - Special switch/relay contacts which can withstand this environment,
  - Hermetically sealed contacts and/or controlled environment,
  - Use of removable plug-in control modules.

4.3. REFUELING

4.3.1. Refueling Outages — Major Considerations

Availability Problem

The prime factor that limits availability of a nuclear unit is the refueling outage, including associated maintenance, inspection, and test activities. A review of past refueling outages of B&W plants indicates that the average plant loses 65 days, or about 18% of the yearly availability. Our analysis of four refuelings at three plants indicates 18.5% average lost availability; individual unit values range from 12 to 23.3%. It is recognized that unexpected and unplanned events can have major effects on many of the sequentially performed events in each refueling.

Work Force Productivity

Many of the refueling tasks are repetitive; therefore, the utility crews are expected to perform more efficiently with experience. Severe environmental conditions of high temperature, humidity, noise, and radiation exert a negative effect on the efficiency of workers. These conditions, coupled with multi-layer, anti-contamination garments and breathing apparatus, make working conditions difficult. Utilities that have only one nuclear unit only perform the refueling task once per year, which does not allow the opportunity to maintain experience without other training periods. Despite these handicaps, the refueling crews are improving their performance as demonstrated at Duke's Oconee Station, where Unit 1 required 74 days to bring the generator on line, and Unit 3, which followed, required only 42 days to complete the outage. These two comparisons are not directly relatable in that abnormal occurrences affected the total time required for each unit, but a review of the Refueling Work Activities (Table 4-4 and Appendix F) shows a 35% average performance improvement between individual tasks. These observed clock times provide support for the learning curve in progress.

Planning

Planning for refuelings has improved, but time is still lost due to insufficient work detail planning, unavailable spare parts, coordination of subtask support, equipment unavailable or out of place, and missing tools. A simple missing wrench could delay a crew several hours due to (1) searching for the tool, (2) leaving the containment, (3) crossing radiation check points, (4) changing to street clothing, and/or (5) securing the tool and then reversing the process.

Training

Manpower availability impacts a refueling outage. At most nuclear plants, trained personnel are scarce. Since each station usually has one refueling per year, little opportunity exists to maintain a trained crew. A station with multiple units can mantain a trained staff. However, the adverse psychological effect on worker productivity is higher for single units than for multiple-unit stations. Also, the possibility of longer fuel cycles means longer intervals between refuelings, which further emphasizes the problem.

4.3.1.1. Observations at Duke Power, Oconee 1

Duke refueled all three Oconee reactors back-to-back starting on May 29, 1977. The refueling for Oconee 1 began on August 5, 1977, using crews that had just completed the Oconee 2 refueling. (See Appendix F for details.)

Environmental Limitations

During this period temperatures exceeded 90F and were accompanied by high humidity in the closed containment. These conditions remained until late in the evenings. High temperature and high humidity probably contributed to a decrease in worker performance.

In the early stages of refueling, several delays were encountered because of airborne radioactivity levels as major components were opened. Much of the release was xenon gas, but the monitoring equipment does not distinguish between xenon and the more restrictive iodine gas. Therefore, precautionary containment evacuation was often executed while a time-consuming sample analysis was made. While the total direct delay due to containment evacuation was only 21 hours, each of the five delays broke the systematic work patterns, which extended tasks longer than the data indicate.

Equipment Problems

The RC pump blind flanges were misplaced, which caused a delay of 2.5 days as new flanges were machined. The overhead crane malfunctioned, causing another 17-hour delay. Both of these delays were on the critical path. The fuel handling equipment was checked out prior to the refueling operation, but once the pool was filled, problems developed: mechanical, electrical, and hydraulic malfunctions in the transfer upender, the control rod telescopic mast, the grapple, and hydraulic hose. Such equipment as the fuel handling equipment and the overhead crane, which remain inactive in the containment except during refueling outage, have high potential failure rates. Improper indexing of the handling equipment caused binding between fuel elements and/or spacer grid hangup, especially in those cases where there was fuel element bowing.

In the spent fuel pool several cases of indicated high levels of gaseous radioactivity required evacuation of the pool area, which in turn slowed the refueling operation in the containment.

Inservice Inspection

The automatic reactor inservice inspection (ARIS) equipment was used for inspection of the reactor vessel welds at Oconee 1. Since this was the initial test of the ARIS equipment, it was expected that between 5 and 7 days total time would be required for this inspection; however, the inspection was performed in less than 60 hours critical path time. During the outage, 32 incore detector assemblies were replaced due to rhodium depletion. This operation was performed according to schedule because of the experience gained from Oconee 2 assembly removal.

Due to the inservice inspection of steam generator tubes (about 24% of the total tubes in both steam generators), the outage was prolonged by several weeks. Original planning was based on 3% of the tuber per generator being inspected. The additional tubes were inspected to comply with NRC guidelines.

Secondary System Tasks — Non-Critical Path

While turbine maintenance has been a concern with regard to controlling the critical path, the Oconee Station performance indicates that the secondary system tasks did not extend the outage critical path. The outage included the following BOP tasks:

- Shut the unit down and perform the necessary turbine tests.
- Secure water systems and place turbine on turning gear for 24 hours.
- Take turbine off turning gear and remove lube oil system from service.
- Disassemble, inspect, and reassemble low-pressure turbine B.
- Perform moisture separator-reheater modifications.
- Disassemble, inspect, and reassemble the feedwater and emergency feedwater pumps.
- Inspect the feedwater pump turbines through manways only.
- Recoat the upper surge tank inner wall.

- Inspect the A and C low-pressure turbines through the manway.
- Inspect (a) the hotwell, (b) the condenser tube and water box, and (c) the steam generator (secondary side).
- Inspect feedwater heaters 1A1, 1B2, and 1C1, including pulling the shell on the 1A1 heater.

These tasks were completed in 28 working days with two crews on an 8- or 10-hour daytime shift.

Duke sustained some turbine maintenance delays due to lack of spare parts. To avoid future problems of this nature, Duke ordered spare turbine and generator bearings and turbine diaphragms.

Duke maintains a specialty maintenance crew which has developed a high level of expertise. This crew rotates from plant to plant, performing maintenance as required. This expertise plus a large laydown working area in the triple bay turbine building, good spare parts inventory and tooling allow them to complete maintenance within the reactor critical path.

Thus, as a result of proper planning of these BOP tasks, the critical path was controlled by the reactor. We conclude that in the forseeable future and except for unusual circumstances, other units may be refueled with the primary side work activities controlling the outage critical path.

Startup Activities

Once plant heatup was started, RC pump balancing and pump seal leakages caused additional delays. Plant inspection and corrections for leak-tightness of the BOP were completed, and the approach to criticality was started.

Upon bringing the reactor critical, zero power tests, including the control rod drop test, were performed. Power escalation was interrupted to verify proper control rod drive connections. Further checks between 40 and 70% full power included verification of flux tilt, verification of new incore detector accuracy, and xenon stability prior to recording core data.

While it is recognized that the core tilt conditions prolonged the approach to power because of the additional tests, the physics test time could be improved by optimizing methods and the development of advanced techniques.

4.3.1.2. Observations at Rancho Seco

The Rancho Seco 1977 outage included the normal refueling-maintenance outage and a complete turbine-generator warranty inspection. The turbine controlled the 1977 critical path events, but since this was a one-of-a-kind activity, the NSS was considered to be controlling relative to this refueling study. (Refer to Appendix F for details.)

The refueling schedule was extended by unanticipated delays, i.e., overhead crane repairs, lack of proper compressed air hoses, problems associated with installation of a new design control rod mast, a crud burst in the RC system, inability of the jib crane to position CRDM leadscrews, fuel assembly spacer grid hangups, airborne radiation levels, and shortage of breathing masks. Additional information is given in the following paragraphs.

The shield blocks were removed quickly, but the overhead crane required repairs that delayed moving equipment into the reactor building and delayed placing the RV insulation racks into the pool, thus delaying removal of the insulation. Lack of proper compressed air hoses delayed the detensioning of the RV head, but once underway the operation went smoothly. A newly designed control rod mast for the fuel handling bridge was readied for installation but eventually turned out to be a problem in the refueling schedule. A delay of over two weeks occurred because of improper machining of new parts, damaged hydraulic lines during installation, leaking hydraulic lines, limit switch failures, replacing valves, and failure of the mast to grapple properly. After this delay, the mast was abandoned and a manual device was installed.

Securing of the CRDMs also took place during this interval. This task was slowed due to high radiation (750 mRem) caused by a crud burst in the RC system shortly before shutdown and by the inability of the jib crane to help lift the CRDM leadscrews high enough, which caused the workers to make the lift manually in order to park the leadscrews. Fuel assembly spacer grid hangups caused some minor delays, but these were handled by the refueling crew.

After emptying the fuel transfer canal, several attempts were made to clean it, but the airborne radiation levels were too high to permit workers to enter without using breathing apparatus. A shortage of compressed air lines and breathing masks caused additional delays, which prevented the reactor head from being tensioned before the reactor building was closed up to perform the containment leak tests.

The seals on RC pump A were replaced. While draining the reactor coolant system, a misaligned valve permitted this flow to be routed back to the fuel transfer canal, requiring the canal to be recleaned. The Rancho Seco crew kept the equipment hatch open (except for fuel movement periods) during the outage, which provided a more workable environment in the reactor building and expedited refueling.

4.3.1.3. Conclusions

Four refueling activities of B&W reactors have been catalogued in Appendix F, and the calculated LFRs are listed in Table 4-4. The standard projected schedule is developed in section 4.3.2 and used in Appendix F and Table 4-4.

Table 4-4 gives system rankings of "Rank 1 — OTSG tube inspection" and "Rank 17 — Install/remove stud hole plugs, remove/install shield blocks and ARIS work." Rank 1 is the furthest from standard and rank 17 the nearest. Recommendations for reducing the LFRs are given in section 4.3.3.

4.3.2. Refueling Performance Standards

Projected Standard Schedule

The projected standard schedule for refueling has been prepared as a basis for comparison with actual station refueling histories. This schedule has been developed from standard projected performance times, which are discussed in the following paragraphs, including the rationale for seleqtion. Further, this schedule is included in Appendix F along with the actual schedules for the four plants studied. Unusual events and/or unexpected delays can prevent a unit from reaching these projected standards, but the comparisons are offered to provide a refueling goal. Unexpected delays usually occur during refueling/maintenance periods. During a refueling operation, hundreds of details must be performed, quite often in sequence; a delay in any one of these events would likely impact the performance critical path.

Standard Projected Performance Time

The "standard projected performance time" is an estimated "normal" time to complete the individual task. The projected times were derived by studying B&W Nuclear Service estimates, utility estimates, and reviewing actual refueling activities using existing equipment now at operating plants. Times for most of the activities have been reduced in the field on an individual task basis, but the overall actual refueling activities have not yet matched the overall projected standard schedule. As the unit matures, worker efficiency should improve, but unexpected equipment failures may be more frequent based on a yearly inspection of much of the equipment. The standard time is not as good as the best time, but it is much better than the average. The standard times are primarily "critical path" (c) clock times. Non-critical (nc) times are given where problems and/or performance could cause entry into the critical path events. Abnormal work must be added to this projected standard schedule.

These times are given in Table 4-15 and are used in the calculation of the LFRs (refer to Table 4-4 and Appendix C) and preparation of the projected standard schedule (Appendix F). The projected standard schedule is a total of 25 days and assumes that abnormal or unexpected events do not control event performance. It also assumes that existing equipment is used. No credit is taken for expected future design changes. The bases for selecting the standard projected performance times are as follows:

1. Shutdown/Startup — 94 hours (c)

   Shutdown only — 30 hours (c)

   The 30 hours for shutdown begins with the unit being removed from the utility grid and continues until the nuclear system is placed on decay heat cooling. During this interval most of the turbine tests are performed. Studies performed at two plants indicate that with improved coordination and no unexpected events, this shutdown period can be completed within 24 hours.

   Startup — 64 hours (c)

   Sixty-four hours are allotted for system startup, which is further subdivided as follows: System heatup and initial deboration — 42 hours
   Power escalation to 40% FP — 16 hours
   Power escalation to 75% FP — 6 hours

   These startup hours do not include physics testing, including zero power tests, which are identified later in a separate allocation. The initial heatup period assumes that the BOP is ready and that water chemistry is within specified limits.

2. Reactor Building Purge — 9 hours (nc)

   As a result of site location, some plants purge the reactor building prior to shutdown and do not need this time allocation. However, 9 hours are assigned for this operation in the projected standard schedule. This operation is normally performed in conjunction with the plant cooldown and is not a critical path event.

3. Health Physics Survey — 2 hours (c)

Two hours are given for a final radiation survey before clearance is issued to allow refueling personnel to enter the reactor building to perform work. Immediately following this clearance, preventive maintenance is performed on the polar crane to prepare it for heavy work during the next few days.

4. Moving Equipment Into/Out of Reactor Building — 24 hours (c)

The projected schecule allows 24 critical path hours to move equipment and heavy tools into the reactor building in preparation for the refueling operation. Sufficient tools and materials are assumed to be in place within 8 hours after the equipment hatch is opened. This 8-hour time is for the initial movement of equipment into the building and includes lowering RC pump parts/tools and the reactor head insulation racks into place. It does not represent total transit time in that the equipment hatch is often opened/closed to move in other equipment during the refueling-maintenance operation. Time for removal of equipment and tools from the reactor building is included in other activities.

5. Reactor Coolant Pump Seal Removal — 96 hours (16 c, 80 nc)

Ninety-six hours are assigned for removal and replacement of the RC pump seals and for balancing the pump motor if required when bringing the pumps back to service. All of this time, with the exception of balancing the pumps, covers non-critical time work events. This time is based on actual performance data at one plant minus the observed delays that occurred. Actual total manhours expended on pump maintenance on all plants for which we have data averaged 740 in a 188-hour span. Based on more recent refueling data from three stations, 96 hours could be reduced to 76 assuming no unexpected events.

6. Removal / Installation of Shield Blocks — 11 hours (c)

Eleven hours are given to remove the shield blocks over the reactor vessel. Best times have been experienced at one station at which 7- and 8-hour schedules have been recorded during recent refuelings.

7. Cleaning of Transfer Canal — 8 hours (c)

The projected plant schedule allows 8 hours to clean the reactor pool after the refueling operation. No time has been assigned for cleaning the pool before refueling since it is assumed the pool is still clean from the last refueling. One station completed this pool cleaning in 6 hours. Average cleanup lapsed time is 20.4 hours, and average total manhours expended is slightly over 80.

8. Removal/Installation of Incore Detectors — 51 hours (nc)

Time allocations for withdrawing, cutting, and reinstalling incore detectors:

Withdrawal	17 hours
Cutting, replacing old detector assemblies	15
Reinstallation	17
Removing old detectors from tank	2
Total	51 hours

These times are based on work covering three detectors per hour for detector withdrawal and reinstallation and one hour for cutting up each replaced detector. If more than 15 assemblies are replaced during the outage, extra time must be allotted, and those plants that have movable incore probes would require an addition 6 to 8 hours to remove this equipment. None of this allocated period should impact the critical path events.

9. Removal/Installation of RV Head Insulation — 9 hours (c)

Three hours are given to remove the reactor vessel head insulation, and 6 hours are assigned to reposition the insulation. Time allocations are based on the assumptions that the reactor pool is clean and that the insulation racks have been placed in the pool during the time allotted for moving equipment into the reactor building.

Work crews at two stations have removed the insulation within 1.5 hours, and reinstallation was accomplished in less than 3 hours at one of these units. Average performance time has been about 10.5 hours with about 42 manhours being expended.

10. Installation/Removal of Canal Seal Plate — 12 hours (c)

Six hours are given for each of the installation and removal operations. Time studies of this operation have indicated that the tasks could be performed in as little as 8 hours total, but care must be exercised in placing the seal plate to prevent leaks. Thus, extra time is allocated.

The best performance time in the field is 14 hours at one station where only a partial crew was used for the installation. Average field performance clock time is 17 hours with an average total of 76 manhours. We estimate that this 17-hour time can be improved; thus, we have shown a 12-hour projected period.

11. Detension/Retension RV Head — 24 hours (c)

Twelve hours are allocated for each of the tensioning and detensioning operations. This allows for time to set up equipment (two tensioners) on the reactor vessel structure and to perform the operation.

The actual tensioning-detensioning operation at one station was accomplished in less than 12 hours clock time, but the average time has been about 30-34 hours with the average total manhours expended being 344. If physical limitations permit, the use of three tensioners instead of two could reduce this standard 24-hour time by 25%. Care should be taken to ensure that tensioners are properly calibrated or they can give the appearance that a stud is binding and thus could cause an unexpected delay.

12. Secure/Reinstall CRDMs — 32 hours (nc)

Eighteen hours are assigned to vent and secure the CRDMs for removal of the RV head. Another 14 hours time is allowed to reinstall the mechanisms after refueling the core. Considerable variation exists in times to perform this task. Difficult working environments are a major factor. If breathing masks are required, work proceeds slowly and could increase working time for the task by 50%. Experience at one plant indicated that mockup training could reduce the projected 32 hours to as little as 10.

13. RV Head Removal/Reinstallation — 18 hours (c)

This task involves removal of the upper CRDM service structure and lifting of the RV head and the reversal of these operations when refueling is complete. Actual removal and the resetting of the head only require 1 hour each, but several additional hours are given for rigging and structure removal. Two stations accomplished this task within 12 and 18 hours. Total manhours averaged about 90.

14. Plenum Removal/Reinstallation — 18 hours (c)

This task involves setting the indexing fixture and movement of the plenum from the reactor vessel to the deep end of the transfer canal. Workers at one station performed this task in 17 hours, and the average is less than 20 hours of elapsed time. Average total manhours for this task: 120.

15. Installation/Removal of Stud Hole Plugs

This task involves removing studs and pouring a rust inhibitor into the stud holes and, after the refueling, removing the agent from the holes and cleaning the hole plug threads.

16. Fill/Drain Transfer Canal — 21 hours (c)

Eight hours to fill and 13 hours to drain the transfer canal are allotted based on the recorded average time of 20.4 hours at three units. The actual time is fixed for a particular station based on its design capacity. This time is a critical path event. The 21-hour period assumes that the canal water will be usable water from the BWST and that the method of filling the canal will avoid crud pickup. Equipment capacities vary at different plants, and additional equipment and water cleanup methods may be required in this area.

17. Checkout of Fuel Handling Equipment — 40 hours (nc)

Forty hours are allowed to perform maintenance on the fuel transfer mechanisms and fuel handling equipment. Quite often this work has been accomplished in 15 hours, but 40 hours are assigned in case faulty equipment is discovered and needs repair.

All of this time should be non-critical path time unless major repair work is required. Average manhours expended for this task is less than 30, but over 300 manhours have been expended when a major problem occurred, and this task has caused critical path delays.

All transfer pool equipment should be checked out when the pool is empty (nc). If the pool is filled, then this item is on the critical path. Changes to equipment may be required to allow complete dry checkout.

18. Refueling Operations — 139 hours (c)

Seven days have been assigned to accomplish the movement of fuel and the associated control components. This task assumes that refueling equipment has been checked and is working properly. The task ends with core reshuffle verified as being correctly installed. Most refuelings have taken considerably longer, but recent refueling operations at three stations have been completed within 3.75, 5.5, and 7 days. Average time for refuelings is about 11 days with about 1500 total manhours expended.

Recent excellent utility refueling times of 3.75 to 7 days have been possible because of efficient use of installed available equipment and detailed planning, wherein fuel and component travel movements were minimized. Improvements included the use of the main bridge for new and old fuel movements in and out of the reactor vessel, swapping of control elements in the upender and/or the spent fuel pool, and the use of the auxiliary bridge to arrange fuel elements in the core.

19. Steam Generator Tube Inspection — 140 hours (nc)

The standard time is based on 120 hours to eddy-current test 3% of the steam generator tubes. Twenty additional hours are allocated for tube repairs if required. Only a few hours of repair work should be critical path time. Initial testing should be performed early in the event that additional testing of tubes is needed. It is assumed that equipment and manpower are available to test tubes in both steam generators simultaneously.

20. Exercise Vent Valves — 1 hour (c)

This projected time is based on site observations.

21. Containment Leak Tests (every 3rd year) — 108 hours (80 c, 28 nc)

These tests are performed only every third year, but they do consume critical path time, not only by the tests themselves but in the preparation and securing from the tests, which dilute manpower from other concurrent tasks. These are basically 3-day tests in which the leak rate of the containment is checked. Since this is an infrequent test, data from only two plants were available to make this judgment as to time allocation.

22. Physics Tests — 72 hours (c)

Seventy-two hours are allotted for physics testing after the refueling outage. Time to escalate power is not included in the physics test allotment but is carried as part of the startup time allowance. This escalation time includes normal power increases and normal deborations. Physics testing for the purpose of this study is divided into 0, 40, and 70% full power. Additional tests are performed at 100%, but since no power is lost, they are not considered a limiting factor.

Thirty-six hours are allowed for zero power testing, which includes determination of the "all-rods-out" boron concentration, temperature coefficients, rod worths, and rod swap results. These tests have been completed in about 36 hours at three stations. Sixteen hours are allocated in the startup operation to escalate power and bring the turbine on line. Upon reaching 40% FP, 14 hours are allowed to check out the incore detectors, check the power imbalance of the out-of-core detectors, and check out core power distribution. If incore current leakage measurements are taken, an additional 24 to 30 hours are necessary to perform this checkout.

Eighteen hours are allotted for 75% FP tests. Incore measurements are cross-checked, and value changes (if needed) are placed in the computer.

NI detector calibrations are checked, and corrections to the power range and imbalance electronics are made if required.

Xenon equilibrium at 40 and 75% FP levels is not included in the 72-hour standard.

Recorded time spans range from 60 to 189 hours for physics testing. It is assumed that the longer times recorded include delay times during startup that were improperly identified. Some plants have waited more than 30 hours at both the 40 and 75% FP levels for xenon stability to build into the core before tests can be run, all of which contributes to the longer recorded times.

23. System Alignment and Checkout — 72 hours (c)

Three days are assigned to align the systems for power operation and to complete the check lists after the RC system is closed and ready for pressurization. Many of these tests run in parallel with each other; any task is capable of entering the critical path during this period. Typical tasks to be performed are:

- Refill and vent the RC system and obtain proper water chemistry.
- Align and check LPI system.
- Align and check HPI system.
- Align and check core flood tanks.
- Pre-heatup check lists:
  - Electrical distribution systems
  - Valve alignments
  - Various cooling water systems
  - Waste disposal systems
- Instrumentation checks.
- Start up condensate and feedwater systems.
- Check out steam seal system and establish turbine vacuum.
- Establish OTSG level and hydrogen blanket.
- Verify reactor cooling system.
- Establish reactor building integrity.
- Actuate penetration room fans.
- Establish pressurizer steam bubble and vent.
- Cross-check CRDM wiring hookups.
- Start heatup.
- Turbine warmup.

24. <u>ARIS Inspection</u> (periodic) — 130 hours (50 c, 80 nc)

Automatic reactor inservice inspection time allocation is 50 hours for the inspection and 80 hours for assembly and disassembly of equipment. These times have been achieved at two stations. The test is performed every third year and should be added to the schedule if this test is to be run during the refueling outage.

4.3.3. Projected Standard Schedule Vs DOE Report Conclusions

The projected standard schedule incorporates the standard project performance times and is given in Appendix F. Note that this gives a total of 25 days, which is based on using existing equipment and the normal work force productivity. It does not include containment leak tests or ARIS inspection. This estimate may be improved by "best" productivity of the work force.

Better total times are estimated in three DOE refueling outage availability reports prepared by B&W, Westinghouse, and Combustion Engineering, respectively (2,3,4). Each contractor estimates an optimum critical path schedule based on incorporating recommendations made in the reports. For B&W and Westinghouse the estimated optimum critical path schedules are 19 and 21 days, respectively. Our calculation of Combustion Engineering's best time is 19.25 days.

The 4- to 6-day differential between the B&W/EPRI standard schedule and the B&W, CE, and Westinghouse DOE values may be explained on the basis that the projected standard used in this report utilizes existing equipment and "normal" work force productivity versus the equipment improvements and "best" productivity of the DOE studies.

An EPRI report (5) also supports the optimum times given in the DOE reports. This report projects 18 days needed to complete refueling if interference items, i.e., testing, maintenance, and inspections, could be removed based on data from the shortest PWR refueling studied.

Our refueling outage studies are performed primarily for identification of limiting factors on plant availability. In the discussion of LFs, the project team has developed suggested courses of action to reduce the LFs. The suggestions should be considered supplemental to other suggestions included in the three DOE refueling outage reports.

4.3.4. Limiting Factors and Recommendations

During the observations at Oconee 1 and Rancho Seco, several factors were noted which could improve refueling operations. The data suggest that equipment improvements can yield faster operations, but one of the major benefits would be from better in-depth planning and coordination.

In general, the main tasks are well planned, and procedures are available. However, planning and coordination of support personnel is often incomplete.

4.3.4.1. Planning and Coordination

During the refueling outages, we observed activities where critical path time could have been saved by more coordination. While a numerical analysis similar to the limiting factor analysis was not performed, a review of the refueling activity charts indicates delays that detailed planning and coordination could have avoided. A refueling-maintenance outage is complex, with many components and specialty work crews that must be coordinated. A refueling outage is relatively new; in-depth expertise has not been fully developed, and additional training is warranted in many areas. Thus, the planning and coordination of activity has the greatest potential for improvement.

Therefore, the project team recommends that utilities maintain refueling staffs that would be responsible for the following:

1. Early in-depth planning to include subtask requirements.
2. Continuous surveillance of operations during the outage to seek possible improvements. As LF operations are improved, other operations will need further improvement.
3. Maintaining availability of spare parts and consumables.
4. Maintaining trained personnel.
5. Performing pre-activity checkout of equipment availability before work commences.
6. Maintaining coordinators both in and out of the containment.
7. Securing prompt engineering approvals that are required during the refueling.
8. Continuous awareness on the part of refueling outage personnel of the actual critical path and adjustment of manpower, equipment, and tools as necessary to maintain the schedule.

4.3.4.2. Balance-of-Plant Critical Path

As increased efficiency is realized, the primary side outage time will eventually decrease to a level below the BOP schedule. As this occurs, many of the BOP maintenance items may become critical path events.

In this study the projected standard schedule (critical path) and the following work activity analysis do not include turbine work. The three DOE refueling outage reports (2,3,4) indicate that turbine work could become a critical path item as primary side refueling outage schedules improve. Our limiting factor studies support this DOE conclusion.

4.3.4.3. Limiting Factor for Refueling

Refueling work activities are listed below and include related refueling and maintenance (where applicable) limiting factors. LF rank is also given; rank 1 indicates that the work activity has the greatest potential for improvement.

The LFR represents a four-plant average as given in Table 4-4. The LFM represents Oconee 1 historical data as given in Table 4-2. The work activities are as listed in Table 4-15 and Appendix F.

- Steam Generator Tube Inspection (LFR 268, rank 1; LFM 1606, rank 4) Steam generator tube inspection has been primarily limited to the minimum 3% of the tubes as required by NRC regulations, until recently when inspections have exceeded 20% at two units. This inspection work should be a non-critical path activity, but at two plants it has impacted the critical path.

  Conclusions and Recommendations: In order to minimize radiation exposure and the possible impact on the critical path, the following conclusions and recommendations are made regarding OTSG inspection:

  1. Improve equipment so that the inspection rate is increased. Inspections should be scheduled early in the refueling and run concurrently on both generators.
  2. Control of the equipment should be automated from a more remote location so that exposure to workers is limited. This recommendation requires further study and development.

  These recommendations would ensure that eddy-current tests are performed without endangering the critical path activities except in very abnormal cases. Additional equipment recommendations are included in section 4.2.1.6.

- RC Pump Seal Removal/Replacement (LFR 124, rank 2; LFM 1603, rank 5) While most of the maintenance and seal inspection-replacement activities are removed from the critical path and central work areas, certain activities, such as balancing the pump motors and leakage corrections, can only be performed during startup, which is on the critical path. Activities in radiation-control areas also limit the working time of an individual.

  Annual maintenance during refueling should be limited to removal and inspection of the third seal. However, due to design revisions, maintenance is more extensive, but this additional work is expected to decline as the service time is extended.

  Conclusions and Recommendations:

  1. A tool rack and portable layout area should be installed to aid in faster removal and assembly of seals. This will require an engineering evaluation and arrangement drawings.
  2. Additional recommendations regarding equipment improvements are included in section 4.2.1.3.

- Refueling Operations (LFR 106, rank 3)
  During this study, refueling averaged 388 hours. Additional data indicate that refueling times ranged from 144 to 384 hours during 1977 after refueling equipment maintenance had been performed. Delays due to equipment failures contributed greatly to the longer times. Recent 1978 performance at three stations, where average refueling time was 5 days (120 hours), indicates that worker skill is approaching a peak efficiency if the equipment performs properly. The equipment durability also appears to be emerging as the limiting factor for this task.

  Failures are not limited to a specific area but are a mixture of mechanical, electrical, and hydraulic problems. The combined effects of heat, humidity, and prolonged inactivity in this environment contribute to the high maintenance requirements. Underwater limit switches and leaky hydraulic lines are components that require large amounts of maintenance attention. For more information on the refueling equipment problem, refer to sections 4.2.4.2 and 4.2.4.3.

  Conclusions and Recommendations:

  Operations —

  1. Utilities that have achieved the best refueling performance have had efficient equipment utilization. Examples are the use of the main bridge primarily to move fuel elements into and out

of the reactor core, use of the auxiliary bridge to rearrange fuel elements in the core, use of the upender to contain fuel assemblies while control elements are being transferred, and the use of the mast in the refueling pool to transfer the control elements.

2. Most units do not have control rod masts in their spent fuel pools. The refueling operation should be evaluated to determine whether it would be cost-effective to add such a feature. As an alternative, several utilities now have manual tools which permit rod handling in the spent fuel pool.

Equipment —

1. Design changes are needed to upgrade critical components for higher humidity and temperature service conditions.
2. Present refueling operations should be reviewed for general improvements; examples are as follows:
   a. Some of the systems have only one fuel transfer tube. If a failure occurred in the tube, the entire refueling operation would be halted. This area should be investigated for alternative methods of transferring the fuel to the spent fuel pool, e.g., two transfer tubes, or for storing fuel in the containment. This study would include both existing design limitations and proposed changes.
   b. The main bridge now transfers single fuel elements from the core to the transfer tube area. Therefore, for each element moved from the core, the main bridge must take the trip and return.

      An idea worth consideration involves transferring several fuel elements with each fuel bridge trip. Since normal cores are loaded symmetrically, fuel assemblies might be transferred together. However, problem areas to be investigated must include accident analysis, licensing, critical mass, and economics.

Additional equipment recommendations are included in sections 4.2.4.2 and 4.2.4.3.

- <u>Check Out Fuel Handling Equipment</u> (LFR 78, rank 4; LFM 428, rank 14, FH bridges; LFM 28, rank 33, fuel transfer system)

Maintenance and checkout of fuel handling equipment take from 15 to more than 300 hours. The project team can make no correlation between time histories and work crews/equipment status.

Conclusions and Recommendations:

1. Bridges and transfer system equipment that exhibit more than routine maintenance requirements should be upgraded. For additional recommendations, see section 4.2.4.2 and 4.2.4.3.
2. Personnel should be trained in repair and maintenance of fuel handling equipment.

- Shutdown/Startup (LFR 77, rank 5)

  Shutdown operations have generally been orderly, but cooldown has exceeded the scheduled period mainly because of other activities, e.g., secondary side water chemistry, slowing the operation. Startup operations also include many delays from other activities which slow the operation more than shutdown (refer to Appendix F charts). Many of these delays were difficult to fully identify by the observation team, but inspections during the startup, such as checking valve leaks, checking out support system and rod drive wiring, and the like, contribute to these delays.

  Conclusions and Recommendations: No specific conclusions and recommendations are offered at this time. It is assumed that as additional experience is gained and improvements are made to other operations, a resultant improvement will also be reflected during these operations.

- Physics Tests (LFR 70, rank 6)

  Physics testing after refueling has taken from 144 to 240 hours of critical path time. Actual physics tests have been accomplished at several plants within 60 to 80 hours when delays and additional tests are deleted. Zero power tests have frequently been completed within 36 to 38 hours. Extreme time variations occur in performing 40 and 75% full power physics testing.

  Several plants have waited for xenon stability to occur before test results are recorded. Other plants proceed upward without awaiting xenon stability. Frequent delays have occurred at higher power levels when incore power profiles indicate abnormal core power distributions.

  Conclusions and Recommendations:

  1. For recommendations regarding core physics testing, including reduction of xenon "hold" periods, refer to section 4.2.1.8.
  2. Incore calibration probe(s) capable of traversing selected core positions would help to resolve questions of whether an observed abnormal peak is real or is due to a faulty detector indication. This would require a development program and detail design activity. (See also section 4.2.6.1 for recommendations on incore detector improvement.)

- Secure/Reinstall CRDMs (LFR 53, rank 7; LFM 458, rank 13)

Securing and reinstalling the CRDs has a high LFM based on the average elapsed time of 85 hours during our study. However, recent improvements (listed below) could lower this time to about 16 hours total.

1. A leadscrew lifting device called a "jumping-jack tool" has been built which enabled field crews to uncouple the leadscrews in 4 or 5 hours; previously this task required upwards of two work shifts using overhead cranes. The new device is available.
2. A new, quick-release valving arrangement to the vent manifold permits venting operations to be completed within one hour. The engineering has been done on this system.
3. A wiring plate is being designed which should allow a quick hookup of all control wiring and will lower wiring confirmation checks. Time savings are estimated at 50 to 75%.

- Detension/Retension Reactor Vessel Head (LFR 38, rank 8)

The tensioning operation on the reactor vessel has proved to be an extremely hot and fatiguing task, which requires up to 80 hours. The projected standard is 24 hours. Time to perform this task is directly related to the workers' experience and the number of tensioners employed. At one plant the detensioning operation was recently completed within 5 hours using three tensioners; the tensioning task was performed within 8 hours. Malfunctioning equipment can cause serious delays on both the tensioning and detensioning operations.

Conclusions and Recommendations:

1. Three detensioners rather than the conventional two should be used to speed up this operation. An additional tensioner should be used as a backup. The stud handling tool should be redesigned so that the reversing drive quickly resets. Compressed air connections should be located near the vessel.

The conclusions and recommendations below also apply for installing the canal seal plate (LFR 7, rank 12), insulation installations (LFR 3, rank 16), and stud hole cleaning (LFR 2, rank 17).

2. Install single-man elevators to raise and lower personnel and tools into the fuel transfer canal. The present method is time-consuming, contributes to worker fatigue, and could become a safety factor. The basic elevator design is commercially available.
3. Improve the capacity of the containment cooling systems or install a cool room (or tent) where workers can obtain temporary relief from the environmental conditions without having to leave the containment. This would require an engineering evaluation and detail design.

- Clean Transfer Canal (LFR 28, rank 9)

  Transfer canal cleaning times have ranged from 8 to 90 hours. If the canal is not adequately cleaned, the reactor vessel closing is either delayed or prolonged. Cleanness must adhere to a low enough radiation level to allow workers to enter the canal floor without wearing breathing apparatus. A hydrolyzer pumping system is used in the cleaning operation, but this system tends to transfer the contamination from one part of the pool to another rather than remove the particles.

  Conclusions and Recommendations: Design and build a "car-wash" fuel transfer canal cleaner to contain the contamination. The idea consists of rotating brushes, controlled flushing water, and a "squeegee" that could be mounted on tracks on top of pool walls. This equipment would require a developmental program and detail engineering.

- Remove/Reinstall Incore Detectors
  (LFR 28, rank 9; LFM 154, rank with 15)
  The average time required to partially withdraw and reinsert the incore detectors has been about 80 hours. At Oconee 1 this time was extended because of removing 32 detector assemblies as a result of sensitivity depletion. Up to 1977 no other utility has had to remove a large number of detectors due to their long life. Therefore, only a small historical data base is available. At least one hour per assembly is required to chop up the tips of the depleted detectors. It should be possible to perform all the incore detector operations away from the critical path events.

  Conclusions and Recommendations: This task is not a problem area.

- Containment Leak Tests (every third year) (LFR 28, rank 9)

  Containment leak tests are a severe limiting factor but are based on an NRC requirement that containment leakage be checked periodically. This is performed every third year. Four or 5 days are required in preparation and performing the tests at Rancho Seco. Considerable manpower was drained from other tasks, which slowed the outage both before and after the tests. The net result is to put these other activities on the critical path.

Conclusions and Recommendations: All past containment integrity results should be studied to determine whether a longer interval between tests (up to 10 years) is warranted. Each time the CLT is performed, the unit penalty is a 1.5 to 2% availability loss.

- Reactor Building Purge (LFR 11, rank 10)

  Purging the containment is normally a non-critical path activity which is performed concurrently with the reactor shutdown activities. Time required is a function of purge system capacity, containment gaseous radioactivity levels, site location, and health physics access criteria. Oconee 1 purge time noted from our data is 20 hours, which reflects site boundary restriction problems and high gaseous radioactivity levels in the containment.

  Conclusions and Recommendations: Gaseous activity level reductions of 50% appear to be possible by isolation and correction of gas leaks and traps. Onsite data must be obtained to evaluate the release problems.

- RV Head Removal/Reinstallation (LFR 10, rank 11)

  Most of this task time is used rigging the head for lift; the actual lift is performed quickly. A device to permit quick rigging has been designed and is now available.

  Conclusions and Recommendations: As workers gain more experience, this task time will be reduced.

- Fill/Drain Transfer Canal (LFR 10, rank 11)

  This task, which usually requires 20 to 25 hours to complete, is limited by the design capacities of the respective plants. During cooldown for refueling, crud bursts occur in the RC system. The radioactive crud is dispersed into the transfer canal when the canal is flooded with water and creates visibility problems that affect the ability to perform refueling operations. The radioactivity of the crud produces radiation exposure problems for personnel involved with the refueling operations.

  It has been reported that when the transfer canal is filled rapidly with high-capacity (low-pressure injection) pumps, pool turbidity is worse than when low-capacity (spent fuel cooling) pumps are used. This fast-fill problem may be related to crud burst dispersal. The present cleanup method is to use the purification demineralizers and filters with crossover connections to the decay heat removal system.

Conclusions and Recommendations: A study and development program is recommended to relieve the impact of filling/draining the transfer canal on plant availability. Suggested areas for study/development are the following:

- A study of crud (sources, radioactivity, turbidity, etc.) as it relates to transfer canal water problems.
- The cost effectiveness of increasing filling and draining capacities to reduce the fill/drain times. The study should include existing pumps and alternative pumps/flow paths.
- The cost effectiveness of improved/additional canal water cleanup system(s). The relative merits of a special shutdown cleanup system should be included.

• Install/Remove Canal Seal Plate (LFR 7, rank 12)

The prime concern of these operations is placing the plate properly so that a leakless seal is made when the canal is filled. Utilities are spending extra time (31-hour average) to ensure that leaks do not occur.

Conclusions and Recommendations: Basic designs for backfitting and new construction have been completed for an inflatable seal to replace the canal plate. Use of this seal should be considered to reduce installation time and lower total exposure to workers.

• Plenum Removal/Reinstallation (LFR 6, rank 14)

This task is similar to head removal (work activity 13, LFR 10, rank 12).

Conclusions and Recommendations: As workers gain experience, times will be reduced.

• Health Physics Survey (LFR 5, rank 15)

Normally this task has only a minimal impact on the outage. Oconee units have experienced high gaseous radioactivity releases that have delayed initial entry into the containment and also caused additional evacuations. The primary source of this activity is believed to be reactor coolant leakage from packings, gaskets, and seals. Thus, additional HP surveys are performed, increasing the time for this function.

Conclusions and Recommendations: A program to define and understand this phenomenon more fully is recommended. The extent of the problem at other nuclear plants needs to be defined before recommendations are made for a specific development program.

- Move Equipment In/Out of Reactor Building (LFR 4, rank 15)

This operation is straightforward and progresses well unless the polar crane fails. If the crane does break down, most containment work stops, and critical path delays occur.

Conclusions and Recommendations:

1. Polar crane design provisions need to be re-evaluated versus nuclear service requirements and recommendations made for equipment improvement. Additional detailed recommendations are included in section 4.2.8.
2. Monorails and other conveyance devices, e.g., conveyor belts, are recommended. These items are commercially available.

- Remove/Reinstall RV Head Insulation (LFR 3, rank 16)

Head insulation removal and reinstallation operations have consumed an average of about 12 hours per operation. With suitable racks and procedures to minimize the number of polar crane lifts, this operation can be performed much more quickly. At Rancho Seco, where a total of three racks and one lift per rack were used, the insulation was removed in 1.5 hours.

Conclusions and Recommendations: Utilities should review their rack design and handling procedure to ensure that this task is being performed in the shortest practicable time. A portable one-man elevator would be beneficial for workers to enable them to get out of the canal more quickly for relief from the hot environment. The basic elevator design is commercially available.

- Install/Remove Stud Hole Plugs (LFR 2, rank 17)

This operation is a routine critical path item of short duration. The polar crane must be available on schedule, and the stud hole plug racks must be in place. The studs are removed, rust inhibitor added to the holes, and the plugs installed for thread protection. After refueling, the plugs are removed and stored, and the holes are cleaned by rotary brushes and a vacuum system.

Conclusions and Recommendations: Estimated time savings of 10 to 15% may be obtained by increasing the availability and efficiency of stud hole cleaning equipment and increased work force efficiency.

- Remove/Install Shield Blocks (LFR 2, rank 17)

  This operation is similar to the stud hole plug activity noted above. The polar crane must be available on schedule, and adequate laydown space must be provided adjacent to the reactor. Data from Oconee 1 and 3 and Rancho Seco give elapsed times of 17, 16, and 5 hours, respectively, for an average of 13 hours for the operation.

  Conclusions and Recommendations: Improvements in this area are based on work force efficiency and an available working crane. Estimated time savings of 10 to 15% are possible.

- Automatic Reactor Inservice Inspection (Optional) (LFR 2, rank 17)

  The ARIS equipment has only been used at the Oconee site. At Oconee 1 the actual test time required was less than 60 hours; at Oconee 3 the test time was reduced to 44 hours.

  Conclusions and Recommendations: As experience is developed, an additional reduction in required time can be expected.

4.4. STUDY OF KEY VALVES

4.4.1. Introduction

The 1976 EPRI study, "Assessment of Industry Valve Problems" (6), was conducted to define specific problem areas and identify technical areas that could improve the performance of valves in nuclear power plants through further research and development. One of the many recommendations from that study was to identify and list "key valves" according to an established selection criterion.

As an amendment to this prime contract, EPRI authorized a special study of "key valves" in three nuclear power plants equipped with B&W nuclear steam systems. Key valves are defined as valves and valve operators, with associated instrumentation, that have had a negative impact on plant availability and/or valves that have required excessively high maintenance. The scope of this special study included identifying the key valves, the causes of the valve problems, and analysis of the failures, and appropriate recommendations for improving the availability/reliability of the valves. This section describes the work performed and generic conclusions drawn, summarizes the information obtained from the participating utilities and valve manufacturers, discusses the data for each key valve studied, and recommends courses of action. For additional information in support of the discussion, refer to the appendices and the tables found at the end of this section.

4.4.2. Key Valve Identification

Key valves in B&W nuclear power plants were identified by the following methods:

1. The participating utilities were requested to supply a list of valves that had caused or extended power reductions. The following utilities participated:
   - Duke Power Co. — Oconee Units 1, 2, and 3
   - Sacramento Municipal Utility District — Rancho Seco Unit 1
   - General Public Utilities — Three Mile Island Unit 1
2. A list of key valves was generated from our availability limiting factor data. Data sources for this list included the following:
   - Oconee 1 historical work requests
   - Oconee Units 1, 2, and 3 power history/work events data for 1977
   - NRC Gray Book
   - EEI Outage Reports
   - MPR-241 (EPRI NP-241, reference 6)

3. A list was prepared from data and conclusions from an internal report of a study of B&W-supplied valves and valve operators on all B&W operating plants. An initial list of key valves was prepared from the sources above and subsequently modified slightly as more information was obtained under the study; the final list is given in Table 4-6. It includes all valves that meet either of two criteria:

   (a) The valve has caused or extended a power reduction on at least two plants.

   (b) The valve has caused or extended an outage at one plant and has been identified as a high-maintenance item.

   Flow diagrams showing the functions of these key valves with respect to other components are given in Figures 4-2 through 4-5.

4.4.3. Data Collection and Study Results

The initial effort was to review (in-house) the initial key valve list to confirm that it was as complete and accurate as possible based on feedback from day-to-day field operations and/or valve vendors. All work requests from the Rancho Seco and Oconee 1 plants were identified to show valve repair and failure data for these valves. Other information on manufacturer, type, size, and inservice conditions was also obtained; these data are shown in Appendix G. Interviews were held at Rancho Seco and Oconee. At Rancho Seco six persons from a wide cross section of disciplines — including plant operation, engineering, and maintenance — were interviewed to obtain additional information relative to their experience with the key valves. In addition, considerable effort was spent in reviewing work request files and operational and shift supervisors' logs. Eight persons in operations and maintenance work at Oconee 1 were later interviewed with respect to their valve operating experience. In addition to individual discussions at Oconee the site work request files and operational logs were reviewed for significant input associated with outages due to valve problems.

In lieu of a visit to the Three Mile Island plant site, telephone communications were established to discuss their valve operating problems to see whether a "common thread" existed for valves used in the same or similar applications as at Rancho Seco and Oconee 1.

One other visit was made — to the Limitorque Engineering and Manufacturing facility in Lynchburg, Virginia. There discussions with the product manager centered around the comments and findings from the Rancho Seco and Oconee 1 site visits, which are reflected in sections 4.4.4 and 4.4.5.

We also had several discussions with valve manufacturers to determine the current practices among manufacturers to improve valve reliability/availability. For example, the Velan Vàlve Company and other manufacturers are now reviewing maximum torque values on operators to be mounted on their valves. They are taking into account inertia forces and torque switch dropout time in determining maximum thrusts applied to the valves. They are then examining the allowable stresses on the stems and valve seats to determine where damage could occur. Velan has established limits on sizing operators to prevent this damage where operability of the valve is required with voltage conditions; example: 70% voltage. There are cases where the valve could be damaged.

A live loading technique is used on the packing glands, which results in long packing life. Torque arrangements are specified to maintain packing tightness. Various valve manufacturers have developed nuclear valve concepts that prevent wear on the valve seats by a design change which allows the disk to be opened and closed from a parallel movement with the seat.

Code case 1621 requires special NDE requirements for stems and packing glands on safety-related valves. Several valve manufacturers have performed operability tests to determine valve frequency in adherence to Regulatory Guide 1.48 at Wyle Laboratories. More sophisticated techniques are used to determine valve natural frequencies. Our conversations with valve manufacturers have indicated that tests have come within 7% of the calculated values on natural frequencies.

4.4.4. Discussion of Data and Information Received

4.4.4.1. Pressurizer Spray Control Valve

The pressurizer spray control valve (Figure 4-2) was identified by B&W some 5 years ago as a valve application wherein design upgrading was indicated. After an extensive study and laboratory test period, a new valve design by Target Rock was selected and field-tested at Oconee and recommended for backfit on all B&W operating plants.

Reports from Duke confirm that the Target Rock valves work well. Note that Figure 4-2 shows two pressurizer spray control valves at Rancho Seco; most B&W plants have only one. This may explain why this valve has caused less availability problems at Rancho Seco than at other B&W plants. The shutdowns on May 21, 1977, and May 9, 1978, for repair of these valves indicate that more problems may be expected in the future. Discussions with Rancho Seco personnel indicate that the temperature above the pressurizer causes a breakdown of the grease in the operator, which either leaks out or cakes up in the operator. Although Limitorque said that even if this happened, the operator would not malfunction, we consider this to be an unacceptable situation. The use of a high-temperature grease (lithium base) in the valve operator was suggested by Rancho Seco, but again this is considered an unacceptable long-range solution. Another suggestion is to build a heat shield/deflector under the operator. No actual failures of operators have occurred so far, although this has been a high maintenance item at Rancho Seco.

4.4.4.2. Pressurizer Spray Control Bypass Valve

The pressurizer spray control bypass valve (Figure 4-2) is used to supply approximately 1.5 gpm of warming water through the spray line to the pressurizer spray nozzles to prevent thermal shock to the nozzle during the spraying operation. The original valve for this application was a Velan ½-inch globe valve, which experienced excessive problems with excessive packing and body-to-bonnet leakage. The original valve at Rancho Seco was replaced with a Control Components valve, and the one at Oconee was replaced with a Kerotest metal diaphragm valve. Both plants report that the replacement valves have presented no significant problems. At TMI, the bypass valve is an Auto-Clave valve which has experienced some packing leaks, but insufficient information was available to determine to what extent. Experience has shown that a packless type valve is best for this application. A possible alternative to using this bypass valve is to install a small fixed orifice in its place since a small constant flow is the desired result.

4.4.4.3. Pressurizer Spray Control Block Valve

This is an isolation valve between the spray control valve and the pressurizer (Figure 4-2). At Rancho Seco it is a Velan 2½-inch motor-operated gate valve, and at Oconee it is a Rockwell 2½-inch motor-operated globe valve. No work requests for repair of the valve were found at Rancho Seco, but at Oconee they have experienced problems with the torque and limit switch adjustments, and Oconee personnel indicated that packing leakage has been a problem. TMI also reports packing leaks but could not provide further details.

The limit switch problems are attributed to high-temperature effects on the grease in the geared switches. Limitorque advised us that they supply Nebula EP-I grease as the standard nuclear grade grease, which meets the qualification tests required by IEEE 382-1972. Rearrangement of the valves to a location away from the intense heat of the pressurizer or installing heat shields around the operator are recommended solutions. Duke has placed heat shields around other valves with similar hot environment problems and alleviated the problem. Relocation of the valves would entail a systems analysis and trade-off study. This has been done on one plant not involved in this study.

The packing leaks are attributed to frequent cycling of the block valve. It is recommended that a packless valve be employed for this service in all plants.

4.4.4.4. Pressurizer Power Relief Valve

The pressurizer power relief valve (Figure 4-2) is an electrically actuated pressure relief device that may be operated remotely or may be set up to relieve pressure automatically to reduce RC system pressure spikes before the "Code Safety Valves" relief point is reached. In the automatic mode when steam pressure reaches the setpoint, a pressure switch is actuated and completes the relay circuit that energizes the valve solenoid. This solenoid actuates the pilot valve, which then lifts the main valve disc from its seat.

At all three plants where this 2½-inch Dresser power-actuated relief valve was investigated, seat leakage of the main valve was identified as the primary problem. This creates a difficult operating condition since the leakage discharges to the reactor coolant drain tank, thereby raising the tank temperature. Common practice among the operationl personnel has been to isolate the power relief valve with the block valve when seat leakage occurred. This defeats the purpose of the relief valve but is an acceptable mode of operation to avoid plant shutdown.

The primary cause attributed to seat leakage of the pressurizer power relief valve is crud accumulating in the pilot valve, which does not allow the main valve to reseat. Another factor believed to contribute to the seat leakage problem is orientation and movement during maintenance, adjustment, and reinstallation. Following standard maintenance procedures, Rancho Seco reports cases of seat leakage.

A corrective measure that we recommend is to continue to improve RC system water quality to minimize the crud that could reach the relief valve. Another recommendation is to perform an in-depth study to determine whether this valve and the block valve can be removed from the system. The Rancho Seco plant has operated for about a year with this valve isolated with no problems. We also concluded that the plants that have had the least trouble with the pressurizer power relief valve are those that have contracted outside specialty vendors and/or the manufacturer to perform maintenance and adjustments. Generally, the contracted vendors can provide more experience than the utility's own maintenance department. The utilities that perform their own maintenance on these valves must provide expert training and suitable test facilities that include hot lines to simulate actual operating conditions.

4.4.4.5. Pressurizer Power Relief Block Valve

This valve (Figure 4-2) was originally supplied to Oconee and TMI-1 by Dresser and to Rancho Seco by Velan. Its function is to provide shutoff when the pressurizer power relief valve needs maintenance. The Dresser valves exhibited thermal growth problems that made them difficult to open after plant heatup. These valves have since been replaced at Oconee with Westinghouse valves and at TMI-1 with Velan valves. At Rancho Seco, the Velan valve has had packing leakage problems, but there is no evidence that it has caused a shutdown or delayed a startup.

Another reported problem has been with the high-temperature effects on the valve operator. The grease within the operator breaks down from the heat and drips out through the grease seals and "cakes" up inside the operator. Oconee has partially relieved this heat problem by raising the valve and valve operator about 3 feet above the pressurizer. Limitorque states that this heat would not cause a malfunction. We conclude, however, that suitable permanent fixes are needed. Possible fixes include the following:

- Move the valve and valve operator away from the pressurizer.
- Install "effective" heat shields.
- Remove the valve completely (see discussion above on the pressurizer power relief valve).

In future plants, we recommend that these and other pressurizer valves be relocated away from the top of the pressurizer to a less hostile environment.

4.4.4.6. Pressurizer Code Safety Valves (Figure 4-2)

These two spring loaded valves are designed to open automatically when steam pressure reaches a predetermined setpoint. Their purpose is to protect the RC system from overpressure. It has been observed that these valves will often leak after lifting. After a period of leaking steam, the seats are damaged by the cutting action of the steam. Relapping and other maintenance is then required. It was also found that after performing maintenance and making setpoint adjustments that the orientation and movement of the valve are very important. If not properly handled, the valve may leak after being reinstalled on the pressurizer. Loads imposed by the closed system's discharge piping must not exceed the limits specified by the valve manufacturer.

It was found that these valves have caused the most trouble during the initial years of plant operation. Representatives from Oconee and TMI said that they do not feel that this valve is a big problem now. These two plants also contract the maintenance and adjustments for the valve to the manufacturer and/or outside laboratories. However, Rancho Seco indicated that the valve was still a problem. This plant had the least operating experience of the three (several months of operation on its second core) and was the only one that performed its own maintenance and setpoint adjustments.

Since proper operation and frequent inspections of these valves are essential for plant operation, the following recommendations are made:

- Each plant should purchase at least one and preferably a complete spare set of code safety valves plus pertinent replacement parts. When one set is in operation, the other can be in for maintenance and repair. Then, should a valve leak, a spare will be available for quick replacement. This arrangement does not solve the causes of valve problems, but, since the valves being used are state of the art, it provides an alternative until advanced designs are available.
- For the utilities that perform their own repair and maintenance of the pressurizer code safety valves, they must provide training to develop expert mechanics to do the work and have adequate equipment for testing. Otherwise, they should utilize the expertise of the manufacturer, who has the proper test fixtures and procedures for handling, maintenance, and setting and adjusting to compensate for temperature effects and the like.

4.4.4.7. Pressurizer Sample Block Valves

These are the first-off valves in the pressurizer sampling lines that sample the steam and pressurizer liquid spaces (Figure 4-2). At Oconee the steam sample block valve is a ½-inch Velan manual globe valve, and the liquid sample line has a 1-inch Velan manual gate valve. These have caused problems due to packing leaks. The packing was originally a braided asbestos impregnated with graphite. Oconee now recommends John Crane 187-I for non-rotating steam applications and John Crane 1625 GF for rotating steam applications. The liquid sample valve at the lower portion of the pressurizer has caused the most problems. We plan to replace both of these valves during the next outage with packless metal diaphragm valves.

Both valves at Rancho Seco were initially 1-inch Velan globe valves with Limitorque operators. They have replaced the liquid sample valve with a Weston 1-inch globe valve. Only one work request was listed for this valve since its installation. The steam space sample valve has a history of packing leaks. SMUD plans to replace this valve with a packless metal diaphragm valve.

For TMI the only information available was that they have a Hoke valve and an Auto-Clave valve for pressurizer sampling lines. They have experienced problems with these valves, including packing leaks. Because of the high pressures and temperatures in which these valves operate, a special high-quality valve is recommended for this service, such as a manual metal diaphragm packless valve.

4.4.4.8. Letdown Line Relief Valve

The design for the makeup and purification system includes one relief valve located just upstream of the letdown prefilters (see Figure 4-3). A 2½-inch Dresser valve is used at Ranch Seco, and a 2½-inch Lonegran valve is used at Oconee.

At Oconee it was reported that this valve has worked quite well, recalling only a few times it had not completely closed after opening. The records show that on August 27, 1977, internal parts were replaced in addition to two other work requests for repacking. They attributed the seating problems to damage caused by foreign material in the lines from the RC system getting into the valve internals.

The Rancho Seco arrangement has two relief valves, one upstream of the prefilters set at 225 psi and one downstream of the prefilters set at 150 psi. During the 1977 refueling outage, startup was delayed due to the upstream valve opening. It was discovered that the setpoints for the two valves had been interchanged, thereby causing the valve to open prematurely. This is believed to be the primary cause of problems with these relief valves although Rancho Seco also reports some problems with this valve failing to close properly due to foreign material in the valve internals.

4.4.4.9. Makeup Flow Control Valve

The makeup flow control valves (Figure 4-3) are 2½-inch valves with Bailey Meter positioners. At the Oconee 1 and TMI-1 plants a Leslie valve was used. At Rancho Seco a Fisher valve is used. Oconee reports seat leakage, body-to-bonnet leaks, and worn stem guides. These problems are attributed to severe service and vibrations due to high flow and high pressure drop (ΔP), which affected the positioner.

Consequently, this valve had to be operated in the manual mode much of the time. Out of 10 work requests found written for this valve, four were for positioner problems; the remainder were for repair of E/P converters and broken air lines and for loose linkage.

At all plants this valve sees almost continuous service from startup to shutdown and has extremely difficult operating conditions due to high pressure and low flow. Repair of this valve involves considerable radiation exposure.

Because of the high demands on this valve, it should receive special attention and be designed to withstand the low flow associated with high pressure drop. System design modifications to relieve the demands on this valve should be considered. The use of a second flow valve in parallel would provide an arrangement to increase service life. Special pipe and valve component supports are needed to minimize damage due to vibrations.

4.4.4.10. Letdown Flow Control Valve

On all the plants studied the letdown flow control valve (Figure 4-3) is a $2\frac{1}{2}$-inch Leslie valve with a Bailey Meter positioner. The valve is in parallel with the letdown block orifice. When the required letdown flow is greater than the designed orifice capacity, this valve is opened. The valve was identified by Rancho Seco as requiring frequent maintenance due to internal valve damage from high pressure drops. Valve repair involved large man-rem exposure. There were 23 work requests against this valve at Rancho Seco and 10 at Oconee. The higher maintenance rate at Rancho Seco is attributed to operation of the valve almost continuously, whereas Oconee usually uses the valve only during startup and shutdown.

Different modes of operation to control reactivity impose different requirements on the operation of the letdown flow control valve. Considering the severe service, Rancho Seco's valve has performed quite well. For the mode of operation at the Oconee plant, the maintenance rate on this valve is not considered abnormally high.

The most common maintenance item from all of the plants is packing leaks, which make up approximately 50% of the work items. Approximately 30% of the problems were attributed to the positioner. In addition to correlating a high maintenance rate to the amount of service, this valve is also subjected to high ΔP operating conditions for valves.

A recent modification to the plug and cage design has been made by Leslie, which has increased the service life because of full guiding at all positions of the plug. Based on recent studies and current design improvements, Control Components is offering their "Self-Drag Velocity Control Elements" to eliminate many of the present problems encountered with severe service control valves.

4.4.4.11. RC Pump Seal Injection Throttle Valves

These valves are located downstream of the seal injection control valve (see Figure 4-3). There is one throttle valve in each of the four lines to the RC pump seals. At Oconee 1 it was determined that after approximately 4 years of operation, the Velan throttle valves were beginning to erode; one has been replaced. During startup and shutdown, these valves are subject to high ΔP conditions, which is believed to be the primary cause for erosion.

Rancho Seco has experienced fewer problems with these valves. They reported that Rockwell angle globe valves were used rather than needle valves, and these have seen approximately 2 years' service. The valves were used to balance seal flow in the beginning of plant life and have essentially been left alone since then. However, their valves are now harder to adjust, which may be a sign of internal erosion. They believe that possibly the single-seal flow control valve is now taking all the pressure drop. Future plans at Rancho Seco are to replace these four valves during the 1980 refueling (which would be after approximately 3½ years of use).

TMI-1 reported that they have had no problems with these valves. Their valves now have approximately 3½ years of operating life. A recommendation for present systems is to perform an annual inspection of the valve internals for possible refurbishing prior to excessive wear. For replacement, a system modification is recommended to include individual flow control valves to each RC pump seal to supplement the main modulating control valve, or employ other valves, such as Valen's "Dragon-Tooth" type, the CCI self drag velocity control elements, or Leslie severe service control valves.

4.4.4.12. LPI Pump-BWST Isolation Valve

Isolation valves are located between the LPI pump and the BWST (Figure 4-4). At Rancho Seco there are two 16-inch Aloyco gate valves with Limitorque operators. At Oconee there are 14-inch Wm. Powell gate valves with Limitorque operators.

Rancho Seco has experienced problems of excessive wear on the valve wedge guides. They attribute this wear to the undue weight of the steam and plug on the lower guide because of the non-vertical mounting angle of the valve. Although work requests do not confirm this as a serious problem, nor was it found to be a problem at the other plants, several instances of bent stems were reported at Rancho Seco and one at TMI-1. In both plants it was attributed to the plant personnel not being able to judge when the valve was completely closed and manually closing the valve too tight, beyond the torque setting specified for the valve operator. These problems support the conclusions of an earlier EPRI valve study (6) that valve orientation and valve operator sizing are important contributions to plant availability.

In accordance with NRC regulations, Technical Specifications require the valve to be cycled periodically to confirm that it will operate. It is recommended that further study be undertaken to evaluate the optimum frequency of testing and compare this with the manufacturer's design life cycles. The Gray Book reported an outage at Rancho Seco on October 1, 1977, due to failure of the valve operator to engage the valve during a scheduled test. Subsequent investigation revealed that the engaging lugs had "rounded over" to the point that it was impossible to engage. Inspection of other valves in the plant with the same type of operator revealed similar abnormal wear. In the plant engineer's opinion, they had a generic problem and contacted Limitorque. Limitorque performed a test problem, but in our discussions the study team was told the cause of the failure could not be positively identified. Limitorque offered two possible causes: a material hardness problem in the lugs, and improper declutching operations. The vendor does not believe this to be a generic problem since it is not a frequent mode of failure. Limitorque and SMUD are continuing to pursue the solution to this problem.

We recommend a study to evaluate the mismatch of valve torque requirement to operator capability to reduce the chances of valve/operator damage. The vendor recommends the addition of a spring compensator on the operator to prevent overtorquing the valve stems as pointed out previously in their assessment of industry valve problems. This addition is more expensive, but it does compensate for the inertia of the operator after the torque or limit switch trips. Criteria for sizing are discussed in 4.4.4.5.

4.4.4.13. Decay Heat Letdown Isolation Valves

These valves provide isolation of the suction line to the decay heat pumps for decay heat removal during refueling or shutdown conditions. They are 12-inch motor-operated gate valves manufactured by Walworth at Oconee and TMI-1 and by Velan at Rancho Seco. Each plant has two valves in series, as shown in Figure 4-4.

Although these valves have yet caused any outages, their important functions and vulnerable location to the RC system make them a potential serious limiting factor component if they should fail. The primary problem encountered thus far with these valves has been packing leaks. The original packing material was John Crane 187-I. A recent occurrence at Oconee involved the second valve in series from the RC system, which was stuck shut and could not be opened in the normal manner. This caused the decay heat system to be inoperative until the valve was reopened.

If the first valve off the RC system needed internal repair, it would require unloading the core and draining the RC system to a point below the intersection of the suction line and the RC piping. We recommend that a system change be incorporated to add a redundant parallel decay heat line with similar valves parallel to the two under study. Then, should either of these valves not open for any reason, the decay heat system would still be functional. It would also alleviate difficult testing and maintenance schedules, which must be coordinated before and after refueling.

4.4.4.14. Main Steam Throttle (Stop) Valves

Problems with these valves were found to be less serious than were originally reported (see Figure 4-5). These valves do have the potential of causing lengthy delays, but no major problems have been found during this study.

At Rancho Seco the two main areas of concern are (1) operation of the "electro hydraulic control" (EHC) system (which controls the valves) and (2) the frequency with which the throttle valve is tested. The EHC was reported to have experienced sequencing problems, causing the throttle valves to be opened or closed in one particular sequence so that, if any other sequence were used, they would not open or close properly. This is a Westinghouse system supplied with the turbine. Rancho Seco personnel advised us that Westinghouse is working with them to correct the problem. With regard to the frequent testing of the throttle valves, the turbine/throttle valve manufacturer recommended a daily test on these valves; however, the rate has been reduced at all plants because of the severe transient produced in the steam system. Another reason for reduction of the testing frequency was the development of cracked seats in the governor valves. Additional information was not available to the study team.

Oconee reported that they have had no major problems with the throttle valves. They have only experienced some minor trouble with debris in the EHC oil system and minor instrumentation and electronic problems with the EHC system.

4.4.4.15. Main Steam Code Safety Valves

Eight of these valves are located in each steam line between the steam generator and steam stop valves (see Figure 4-5). They are Dresser valves at Rancho Seco and TMI-1 and Crosby valves at Oconee 1.

Both Rancho Seco and TMI identified these valves as causing loss of availability. At Rancho Seco the most common problem has been with seat leakage of the valve after having been popped. At least one delay in return to power was caused by one of these 16 valves being inadvertently left off an adjusting mechanism. Vibrations later apparently caused the setpoint to drift. There were 44 work requests for these 16 valves, 34 of which were for lapping the seat and plug, 7 were setpoint adjustments, and 3 were miscellaneous.

The main steam code valves on Unit 1 at TMI are maintained and adjusted by Dresser. TMI reports no problems with these valves on Unit 1.

One case was reported at Oconee where the main spring broke during actuation and required a shutdown to repair. All the work requests found were for lapping the disc and seats.

4.4.4.16. Turbine Governor Valves

The turbine governor valves (Figure 4-5) were supplied by the turbine supplier, as were the throttle (stop) valves. At Oconee 1 and TMI-1 it was found that the governor valves have caused very little trouble. At the Rancho Seco plant, the turbine governors have caused at least four outages (October 1975, December 1977, and two in March 1975). Typical problems have been shearing of the anti-rotation stem pin and cracks in the valve seats in areas around staging points. Rancho Seco reported that these problems are being worked on by the vendor, but details of the corrective action were not available.

4.4.4.17. Main Steam Bypass to Condenser Valve

These valves, also known as turbine bypass valves (Figure 4-5) were reported by all three plants as having caused loss of availability. At Rancho Seco there are four Fisher 6-inch control valves with a Bailey Meter positioner used for bypass valves. TMI-1 uses six Fisher 8-inch control valves with Bailey Meter positioners that are similar in design to the Rancho Seco valves. Oconee has four Atwood-Morrill 8-inch control valves with a Bailey Meter positioner.

At Rancho Seco the valves originally experienced problems related to the internal pilot design and with the Belleville springs, which did not reseat the valve after operation. When the proper spring was used, the valve operated successfully. Stroking problems caused by breaking of linkage between operator and valve were also corrected. Rancho Seco maintenance personnel feel that this valve is no longer a problem.

Discussions with TMI personnel revealed that their bypass valves were still experiencing seat leakage problems. Several modifications had been made to correct seat and cage leakage and blow body-to-bonnet gaskets. It had not been determined at this report writing whether or not the modifications had corrected the problem.

It was reported at Oconee that their bypass valves have had seat leakage due to scarring and wearing of the seats and stem. The valve has a high pressure drop across the seat with main steam pressure on one side and a vacuum on the other, which has caused erosion of the valve internals.

4.4.5. General Conclusions and Recommendations

EPRI report NP-241, "Assessment of Industry Valve Programs," stated that valve applications most prevalent in causing plant shutdowns involved the main steam isolation valves (MSIVs), feedwater control valves, pressurizer spray valve, and turbine valves with their related controls (6). Our study showed that the problems with valves associated with the pressurizer agree with the NP-241 conclusion. More important, the study showed a positive trend of valve performance in today's nuclear power plants. The first three general recommendations below pertain to key valves (in this study), and the last three pertain to valves in general.

4.4.5.1. Main Steam Stop and Governor Valves

The main steam stop and governor valves were found to be a problem only at Rancho Seco. Oconee 1 and TMI-1 have had no significant problems with their steam system valves. In these three plants, the turbine manufacturer supplied the steam turbine-related valves. At Rancho Seco the valves were serviced by the turbine vendor; we were unable to obtain enough information to evaluate the problems.

4.4.5.2. Pressurizer Valves

All of the valves associated with the pressurizer have been identified as having either caused a loss of plant availability or are considered to be high-maintenance items. Four problem areas were identified. The first problem area of primary concern is reactor coolant leakage to the reactor building atmosphere. This usually occurs via packing leaks or body-to-bonnet leaks and, if the leakage is greater than allowable Tech Spec limits, a shutdown is required. Another problem created by this leakage source is the cost associated with the storage, treatment, and disposal of the resultant wastes.

Seat leakage is the second problem that not only is detrimental to the valve internals but causes difficult operating conditions. It usually occurs through the power-actuated relief valve or safety valves and is vented to the RC drain tank, in some cases creating a cooling problem for that tank.

The third problem is the identification of leakage. When any of the pressurizer safety/relief valves develop seat leakage, it is generally difficult to determine which valve is leaking. Instrumentation is needed to determine which valve is leaking, thereby avoiding the possibility of removing the wrong valve for repair.

The fourth problem identified with valves associated with the pressurizer is their location in hostile environments. The hostile environment in the pressurizer area not only affects the operators, but it makes maintenance and repairs more difficult because of high temperature, radiation, and accessibility.

We recommend the following to increase the availability of valves associated with the pressurizer.

1. Employ packless metal diaphragm valves and hermetically sealed solenoid valves wherever possible. This could be done more easily on new plants under design than on existing plants where seismic analysis, economic conditions, and licensing requirements must be considered.

2. Seat leakage has been attributed primarily to foreign material in the valves that prevent proper seating. Continued and concentrated efforts are necessary to minimize this crud or foreign material and keep it from entering the valve seats as discussed further in section 4.2.3.8.

3. Develop a more sensitive method of monitoring safety valve seat leakage. This would be especially helpful for the safety/relief valves and two block valves. Perhaps acoustic detection devices could be employed, or a camera to monitor atmospheric relief.

4. In the design of future plants, locate valves (where feasible) away from the pressurizer and behind shielded walls. On operating plants, to the extent practicable, either move valves away from the pressurizer and/or install suitable heat shields.

5. When the utility maintains the pressurizer safety/relief valves, they must also provide suitable maintenance fixtures and maintain a mechanics' training program to retain expertise. For instance, Dresser Valve Company is preparing an *Applications Manual* for their "consolidated" safety relief valves. information in this manual is based on the results of an extensive study by Dresser, B&W, and several participating utilities. The intent of the manual is to describe the working parts of relief valves and their functions. It will also prescribe recommended techniques for testing, adjusting, inspecting, repairing, maintaining, and handling safety/relief valves.

4.4.5.3. Valve Motor Operators

The reported major problem with valve operators was due to the effect of prolonged exposure at high temperature (above 300F). This is reflected in the lubricant used in the mechanical gears and the limit switches. Limitorque, the major supplier of valve operators, made a design change in the limit switch lubricants to withstand the hostile environment. We recommend moving the valves affected to less hostile locations relative to the pressurizer. Relocation can help solve another major problem, that of high radiation exposure for maintenance personnel.

The valve operators are oversized. Requirements for operators to provide design torque at 80 and 70% of rated voltage has created a mismatch of valve to operator. The inertia of the operator after tripping the torque or limit switches is not easily absorbed in the valve. Therefore, some compensating feature to reduce the effects of mismatch is recommended for operators that, by regulation, must be oversized to ensure operation at reduced voltage.

4.4.5.4. Body-to-Bonnet Leaks

Body-to-bonnet leaks are among the most common valve problems identified, as illustrated in Table 4-7. We believe that a better understanding of body-to-bonnet leaks could lead to design and operating improvements that would result in improved plant availability and less plant maintenance. Based on these conclusions, we recommend that a program be initiated to do an in-depth investigation of the parameters that could affect body-to-bonnet leaks. Suggested parameters to be investigated are the following:

- Temperature and pressure effects
- Thermal cycling effects
- Stud (or bolting) parameters
  - Torquing
  - Material (carbon Vs SS in hot borated water fluids)
  - Spacing
- Gaskets
  - Material
  - Design
  - Other
- Bonnet design
  - Flange weight
  - Distortion Vs temperature
  - Surfacing

There are numerous cases documented and discussed in previous sections where flanges on tanks, heat exchangers, valves, etc. have leaked and caused maintenance and/or plant availability problems.

We believe that the information gained from such a parametric study would also be directly applicable to flange leakage problems.

4.4.5.5. Information Feedback

In this key valve study as well as in the plant availability study and in our discussions with B&W engineers, we find many cases wherein designers and suppliers are unable to obtain sufficient information to even identify or understand a problem, much less resolve it. This was made especially clear in our contacts with valve and valve operator vendors.

This study team fully understands and appreciates the difficulty of obtaining good failure data even when a man is assigned to the site as an observer/data collector as in this study. Although utility management generally supports dissemination of information, the workers' and operators' primary concern is to correct a problem as quickly as possible using as little manpower as is practicable, and then to resume their main function — generation of electricity. In most cases, little effort is given to apprising the supplier of the information needed for long-range product improvement. The maintenance and operating personnel do not necessarily have access to specification and engineering data, environmental limits, and other ordering data. The manufacturer and specification writer may be further removed.

4.4.5.6. Valve Application Guidelines

In this study as in reference 6, we find cases where the valve performance can be improved by using specialized valve types which are best suited for a particular application.

We believe that the nuclear industry would benefit significantly from a set of standardized valve "application guidelines" for valves in key applications, locations, etc. Although informative, practical general guides already exist (see references 7 and 8, for example), we recommend developing specific valve guidelines (recommendations) for specific applications. For example, the pressurizer sample block valve guidelines might recommend, among other things, that the valve be a manually operated, packless-metal, diaphragm gate valve and that it be located outside the secondary shield wall. The guidelines for the letdown flow control valve might be that it has demonstrated that it will operate for X number of cycles with 50 and 200 gpm flow, at pressure drops across the valve of 1000, 1500, and 2000 psi, etc. Even though each NSS supplier has BOP criteria for placement of valves, industry guidelines on valve location should be included to minimize worker radiation exposure, ease of repair, etc. (The use of scaled plant models can also be an extremely useful tool in equipment arrangement.)

These guidelines should cover the various types of valves that impact plant availability directly or indirectly. It should also cover general valve applications. For example, the guildeline might recommend that all RC system first-off valves be hermetically sealed — perhaps a special type of hermetically sealed valve. We recommend that these valves be identified and standards be established by representatives from the utilitites, NSS suppliers, architect-engineers, valve vendors, and valve vendor operator.

4.4.6. Valve Categorization by Vendor, Type, Size

4.4.6.1. Discussion of Results

In an attempt to identify particular generic valve problems for key and non-key valves, we have categorized valve and valve operator problems by vendor, valve type (globe, gate, etc.), size, and failure mode (body-to-bonnet leak, seat leak, etc.). The analysis is based on Oconee 1 work requests for the period July 1, 1974, through December 31, 1977. The results are summarized in Table 4-7.

This table categorizes valve repair events by valve vendor and valve size. In all cases, repair event data are presented on a unit valve basis. Although the valve operator vendors are not identified, operators (where applicable) are either manual or motor-driven. Manual operators generally do not present problems. Motor-driven operators are almost without exception supplied by Limitorque.

In compiling these data, we listed all work request repair events identified, but we did not infer repairs not specifically listed on the work requests. For example, if the work request showed that a valve underwent both seat repair and repacking repair, both events are listed. If the work request showed only seat repairs, we did not assume that the body-bonnet gasket was replaced nor that the valve was repacked although both were probably done.

4.4.6.2. Conclusions

Conclusions are difficult to draw from an analysis such as this because of statistical uncertainties due to the relatively small sampling, the diverse valve applications, and the influence of factors not included in this analysis, such as frequency of use, pressure, temperature, water/steam conditions, etc. These factors are known to be important and are discussed as appropriate in section 4.2 and in other parts of section 4.4. A study of Table 4-7 reveals no trends of significance by vendor nor by failure mode. The few cases where failure rate appears to be high (for example, 1.3 seat leaks per valve on Dresser relief valves and 1.6 packing leaks per valve on Rockwell-Edwards globe valve) are attributed to the small data sampling (three valves in both cases) or severe operating environment. This conclusion is illustrated by the following summary, which shows the total failure rate by vendor for all events where five or more valves are involved.

	Failure rates, events/valve		
Vendor	Globe	Gate	Relief
Rockwell	1.0	NA	--
Crosby	--	--	0.3
Fischer	1.8	NA	--
Lanergan	--	--	1.2
Walworth	--	0.5	--
Velan	1.3	1.1	--
Crane	1.4	1.4	--
Crane-Chapman	--	1.4	--
Grinnell	0.1	--	--

The "total valve repairs" (categorized by valve size) in Table 4-7 show that for gate valves, body-to-bonnet repairs occur about twice as often as do seat repairs and packing leaks. The same trend is not evident, however, for globe valves. The data in Table 4-7 also show a noticeable increase in failure rate as the size of the valve increases and a noticeably higher failure rate of globe valves than the failure rate of gate valves. The following tabulation illustrates these conclusions:

	Failure rate, all valves	
Valve size, in.	Globe	Gate
$\frac{1}{4}$ to $1\frac{1}{2}$	0.9	0.6
2 to 3	1.4	0.9
$3\frac{1}{2}$ and up	2.0	1.1
Total	4.3	2.6

Relief valve data are not included in this tabulation because of the small data sampling.

4.4.7. Valve Analysis by System

To give more information on valve performance in one system compared to that in another system, we have ranked the LFOs and LFMs for valves by system and present this in Tables 4-8 and 4-9. The source of the data is the Oconee 1 historical data (July 1, 1974, through December 31, 1977).

Table 4-8 shows the cases identified where valves impacted plant availability. A total of five events in four systems were identified. A comparison of the limiting factor for operation for valves with the limiting factor for operation for other components (Table 4-1) shows that, except for pressurizer valves, the valves are relatively less important as they relate to plant availability.

Table 4-9, however, shows that valves are quite important as they relate to plant maintenance. Comparison of these data with the LFM of other components (Table 4-2) shows that, except for a few notable exceptions (turbine, steam generator, RC pumps, etc.), valve maintenance is among the highest maintenance noted. Table 4-9 shows that on the average there are 0.4 repairs per valve per year. Valves in the main steam system low-pressure service water and the condenser circulating water show a noticeably lower frequency of repair. Valves in the coolant treatment and penetration room vent systems show a noticeably higher repair rate.

4.5. RADIATION EXPOSURE DATA

4.5.1. Introduction

Radiation exposure data for Oconee Units 1, 2, and 3 were reviewed. The major source of these data was the 1977 quarterly Personnel Radiation Dose Reports to the Oconee station manager. Duke collected and reported these data in compliance with U. S. NRC Regulatory Guides 8.8 and 8.10. Personnel exposure data are recorded for routine work and special shutdown work for the Oconee Station. This routine work includes such functions as normal station maintenance, changing of filter cartridges, and other planned or expected tasks. These data are compiled for the site, not for individual units. Special shutdown work, including refueling/maintenance outage work, is compiled by unit; radiation exposure data are given for all significant work categories.

4.5.2. Study Results

Table 4-10 gives total man-rem exposures for nine categories of routine work by quarter for Oconee 1, 2, and 3 for 1977. Exposure for each work category is summed and the percentage of total man-rem exposure for each work category is shown. The totals in Table 4-10 show that station maintenance (41.4%) and station surveillance (29.1%) account for the majority of personnel exposure from routine work.

Table 4-11 gives the man-rem exposure for each of 17 special shutdown work categories at Units 1, 2, and 3 for 1977. The Oconee 1 OTSG tube leak test and plugging resulted in the highest man-rem exposure, accounting for 25.6% of the total special shutdown work dose. Each unit completed a refueling outage during 1977. Exposure at Oconee 1 was higher than at Units 2 and 3, primarily because of OTSG leak testing and tube plugging. Exposure levels for Unit 1 were also higher for RV head removal and replacement, primary valve repair or replacement, inservice inspection, and incore instrument work. Installation of instrumentation in the 2B steam generator of Oconee 2 accounted for 36.5 man-rem. This installation, made on the 2B steam generator only during 1977, contributed over 13% of the special shutdown man-rem exposure at Oconee 2.

Table 4-12 is a bar graph summary of the total exposure associated with routine and special shutdown work. Total man-rem exposure is shown for each calendar quarter of 1974, 1975, 1976, and 1977. The high quarterly cumulative dose for the third quarter of 1977 is caused by refueling Oconee 1 and 2 and OTSG tube inspection and plugging on Oconee 1, 2, and 3. The bar graph shows fairly uniform totals for exposures from routine work over the 4-year period except for the third quarter of 1977, when the back-to-back refuelings occurred.

4.5.3. Conclusions and Recommendations

These radiation exposure data are presented to show the relative importance (from a personnel exposure viewpoint) of the various work activities and to show the change in personnel exposure with time. Table 4-12 reveals no significant change with time in the cumulate exposure for routine work. Cumulative exposure due to special shutdown work appears to have increased with time; this is attributed primarily to increased steam generator tube leak testing and plugging and eddy-current testing during 1977. Other than this steam generator work, we find no special work category where improvements are suggested. Rather, we conclude that most, perhaps all, work involving radiation exposure would be worthy of additional studies to identify ways to reduce exposures. These studies should be comprehensive and should include such areas as the following:

- Increased use of remote tooling.
- Improved "quick disconnects" in areas such as the reactor vessel head, control rod drives, and manway covers, etc.
- Reduction/elimination of crud traps and crud bursts.
- Improved flushing and decontamination techniques.
- Improved shielding techniques.
- Improved procedures.
- Selected tool kits for specific tasks based on work activity evaluation.

4.6. ADDITIONAL ANALYSIS AND AVAILABILITY PROBLEM AREAS

Sections 4.6.1 and 4.6.2 describe problem areas that we identified during this study although not directly from the documented outage data. The majority of the information is based on the experience of operators and engineers. All of these problems have either caused some loss of plant availability or have potential for resulting in loss of availability. Each problem is described briefly and, whenever possible, an approximate impact time is given. Recommendations for programs to alleviate the impact of each problem are also given. Sections 4.6.3 and 4.6.4 give further analysis of pumps/motors and heat exchangers and compare the performance of these components in one system to their performance in another.

4.6.1. Secondary System Chemistry

4.6.1.1. Feedwater Iron Control

Control of the iron content in the steam generator (OTSG) feedwater during startup has been a concern at several plants. Operating specifications require that the iron content be less than 100 ppb prior to feeding the OTSGs and less than the normal operating limit of 10 ppb within 8 hours. Recently, startups at both Davis Besse Unit 1 (DB-1) and Crystal River Unit 3 (CR-3) have been delayed as a result of this problem. The delays were about 5 and 3 days at DB-1 and CR-3, respectively. An in-depth study is recommended to determine the sources of the iron and the reasons for its entry into the system. Methods should be defined to eliminate the sources, to reduce the amount of iron entering the system to acceptable levels, or to otherwise reduce the impact on plant availability.

4.6.1.2. Disposition of Cycle Drains on Shutdown

Operating plant experience has shown that OTSG water chemistry excursions can occur during and following planned and unplanned plant shutdowns. The source of the chemicals appears to be turbine steam cycle component washing, the effluent from which is subsequently transported to the feedwater via the high pressure heater drain tank. We recommend that the various methods of handling these effluents should be reviewed and standard methods developed to minimize the impact of the chemical excursion on the subsequent startup. The present routing methods and their effects on the system should be included in the study. Procedures that do not permit altering drain routing during shutdown from reactor trips should also be included.

4.6.1.3. Computerized Chemistry Monitoring

Deviations from the secondary chemistry specification requirements have caused power reductions and plant shutdowns. An on-line computerized system to monitor the water chemistry and display the information to the plant operators would minimize the impact of chemistry deviations on plant availability by permitting corrective actions to be taken before the out-of-specification situation occurs. We recommend that a study and development program be implemented to develop such a system. The system should include in-line monitors, sample analyzers, and other equipment and software to provide the operators with data and corrective actions to be taken.

4.6.1.4. Chemical Addition System

Malfunctions in chemical addition systems have led to chemistry conditions that delayed plant startup, especially in the secondary plant. Improvements in the reliability of these systems will reduce the time lost during plant startups. We recommend that a study program be implemented to identify specific causes for malfunctions in these systems and possible solutions.

4.6.2. Reactor Coolant System

4.6.2.1. High Chloride Levels in RC System

High chloride levels in the RC system have been a problem at PWR plants during plant shutdown and refueling periods. The chloride levels have exceeded the Technical Specifications for the plants and have caused losses of plant availability of up to several days. The high chlorides have been attributed primarily to chloride elution from the anion resin in the purification demineralizers, but other sources also have an impact on the problem. A study and development program is under consideration by EPRI. The highlights of the proposed work scope being discussed are as follows: (1) identify source(s) and pathways of chlorides into the RC system, (2) study design and operating procedures to eliminate and reduce the impact of the problem, and (3) develop recommendations for programs that could relieve the problem. (See section 4.2.2.1 for additional discussion on this subject.)

4.6.2.2. High Temperature Filtration

High radiation levels in the RC system have had a major impact on plant maintenance and refueling operations and have thus affected plant availability. One important cause of high radiation levels is the radioactive corrosion products deposited on the surfaces of RC system components and in the coolant itself. One feasible method to limit the radiation levels is to remove the activated corrosion products with high-temperature, high-flow filtration techniques connected directly to the RC system. We recommend a program for the design, installation, and testing of a demonstration filtration system. The program should require determination of the effectiveness of the filter in an RC pump bypass loop with flows in the range of 1000 to 2000 gpm and the effectiveness of the flush techniques for removing the corrosion products from the filter.

4.6.2.3. Chemical Contamination of RC System

Sodium hydroxide (caustic) is used in reactor building spray systems to enhance the removal of radioiodine during post-LOCA operations. Three instances are noted wherein the caustic from this system has contaminated the RC system and the auxiliary systems, resulting in total losses of about 25 EFPD in plant availability. The initial caustic entry causes the contamination of many areas. The RC system can be cleaned up, but the potential remains for further contamination resulting from residual caustic retention in the auxiliary systems. This condition exists during resumed power operation. We propose a study program to establish administrative controls and design and operating modifications to minimize the potential for such contamination.

4.6.2.4. Hydrogen Control in RC System

Following refueling, the control of dissolved hydrogen in the RC system during startup has been a problem. The primary reason is that the amount of hydrogen addition is limited by the method of getting hydrogen into the coolant — currently by dissolving hydrogen in the purification letdown water flowing through the hydrogen gas space in the makeup tank. The problem is compounded by the fact that some of the dissolved hydrogen is lost in the expansion water bled out of the RC system during heatup and in the water bled out of the system to reduce the boric acid concentration. The problem can be resolved by providing a more efficient method of dissolving the hydrogen in the purification letdown water. We recommend a program to define and evaluate methods, equipment, and/or procedures to accomplish this objective.

4.6.3. Analysis of Pumps/Motors by System

The LFMs for pumps/motors are given in Table 4-13. These values, taken from Table 4-2, were calculated from Oconee 1 historical data. The LFMs for pumps and motors for the RC system are shown for comparison with other pumps/motors. Table 4-13 gives the number of components, the number of maintenance events per component, and a calculated LFM per component.

A special study was completed to determine whether the pump/motor maintenance data indicated either a generic pump/motor problems or a system-related problem. As seen in Table 4-13, the RC pumps/motors are clearly the largest pump/motor LFM by system and by component.

The makeup and purification pumps/motors have the next most important LFM by component, and feedwater pumps are third. No explanation for this is offered; however, these pumps work against a high head and generally have high service demands. Other pumps/motors have not been major causes for maintenance as indicated by the LFM values. The data show that, except for the RC pumps/motors, the average number of repairs has been 0.8 repair per component per year. This compares to 0.4 repair per component per year for valves. The average LFM for pumps/motors is 16.7 per component per year. Maintenance data indicate that only the RC pumps/motors are subject to generic, i.e., design class, problems.

LFOs for pumps and motors have also been studied (Table 4-1). In addition to the RC pumps/motors, only the pumps and motors in the condensate and feedwater systems show impact on operation. The relative size of these LFOs compared to that for the RC pumps and motors indicates no generic pump/motor problem. Similarly, comparison of the components and feedwater pump/motor LFOs with the LFOs with those from other systems and components indicates no significant problem.

4.6.4. Analysis of Heat Exchangers* by System

Heat exchangers were investigated as parts of systems in earlier sections of 4.2; LFOs and LFMs were calculated (Tables 4-1 and 4-2). Further analyses of heat exchangers as a class were completed to determine whether a generic problem is indicated.

The LFOs for Oconee 1 indicate that, with the exception of the steam generators, heat exchangers have not impacted the availability of the plant (Table 4-1). The LFMs, however, indicate that significant maintenance has been required on the heat exchangers. Table 4-14 lists the LFMs for heat exchangers, the number of components, the number of events per component, and the calculated LFMs per system and per component. Heat exchanger maintenance in the feedwater and moisture

- - - - -

*Includes heaters, reheaters, and coolers.

separator/reheater systems has been primarily to repair tube leaks, gasket leaks, and baffle plates, as discussed in sections 4.2.3.3 and 4.2.3.8. The single event for the makeup and purification system was to replace the letdown cooler, as discussed in section 4.2.2.1. Table 4-14 shows that the average number of repairs has been 0.44 repair per component per year, about half that of pumps/motors. The average LFM per component per year is 35.5—about twice that for pumps/motors, which says that heat exchangers require less frequent maintenance than pumps/motors, but that maintenance requires more time.

Table 4-1. Oconee 1 Limiting Factors for Operation, July 1, 1974 - December 31, 1977

Rank		System/component	No. of events	Power loss factor	Mean time to repair, h	Additional loss per event, EFPH	Average loss per event, EFPH	LFO (normalized)
1	1F	Steam generators	9	1	154	131	285	733
2	4A	Control rod drives	32					356
		4A1 Drives	5	1	3	27.3	30.3	43
		4A2 Stators	4	1	7.4	32.3	39.7	45
		4A3 Position indicator	8	0.94	12.4	23.9	36.3	83
		4A4 Power & T/C cables	3	1	25.3	37.8	63.1	54
		4A5 Closure/vent system	2	0.75	5.3	35.2	39.2	22
		4A6 CRD control system	10	0.95	12.8	25.5	37.7	108
3	1D	RC pump motors	23	0.97	27.8	24	51	335
4	1H	Core physics & Rx safety	57					307
		1H1 Power ascension	5	0.29	8.4	0	2.44	4
		1H2 Core tilt	3	0.25	165	0	41.2	35
		1H3 Xenon hold	46	0.11	27.7	0	2.5	33
		1H4 Startup physics tests	3	0.69	396.7	0	275	235
5	1A	Reactor and internals	1					295
		1A1 Reactor	0					0
		1A2 Internals	1	1	1034	0	1034	295
6	4B	Fuel handling bridges	17	1	55.4	0	55.4	269
7	5	Electrical systems	3					225
		5A Generator	1	0.85	4	27.3	30.7	9
		5B Switchgear	0					0
		5C Controls	0					0
		5D Exciter	2	1	248	131	379	217
		5E Transformers	0					0
		5F Substation	0					0
		5G ISO phase bar	0					0
		5H Batteries	0					0
		5I Chargers	0					0
8	1C	Reactor coolant pumps	3	1	62	131	193	165
9	6A	Control & monit. equip.	11					69
		6A1 Integr. Cont. System	7	0.87	8.9	7.8	15.5	31
		6A2 Non-nucl instr'n	4	0.9	9.6	24.8	33.4	38
		6A3 Incore detectors	0					0
		6A4 Computers	0					0
10	3L	Turbine lube oil system	3	0.96	26.2	27.3	52.5	45
11	3A	Main turbine	2	0.63	9	65.5	71.2	41
12	1G	Pressurizer	1					40
		Valves	1					40
		Heaters						
13	3D	Condensate	2					29
		Heaters	0					0
		Valves	0					0
		Pumps and motors	2	0.33	30	41.5	51.4	29
		Air ejectors	0					0
		Demineralizers	0					0
14	4C	Fuel transfer system	4					25
15	3M	Turbine EHC system	2	0.9	9.7	24.5	33.2	19
16	4E	Suppressors & hangers	1					18
		Hydraulic suppressors	1	1	37	27.3	64.3	18
		Pipe hangers	0					0
17	3C	Feedwater system	5					13
		Valves	0					0
		Heaters	0					0
		Pumps	2	0.6	7.3	7.5	11.9	10
		Turbines	1	0.4	2	3	3.8	1
		Miscellaneous	1	0.4	4	5	6.6	2

Table 4-1. (Cont'd)

Rank		System/component	No. of events	Power loss factor	Mean time to repair, EFPH	Additional loss per event, EFPH	Average loss per event, EFPH	LFO (normalized)
18	3J	Heater drain system	2					11
		Valves	1	1	6	27	33	9
		Tanks	0					0
		Pumps	0					0
		Coolers	0					0
		Pipes	1	0.2	10	2.5	4.5	2
19	3B	Main steam	2					10
		Valves/valve tests	2	0.73	5.8	13.7	17.9	10
		Pipe weld repair	0					0
20	7D	Coolant storage	1					9
		Valves	1					9
		Pumps	0					0
21	8A	Polar crane	1	1	20	0	20	6
22	3I	Gen stator cooling	1	1	2	27.2	29.3	3
23	6B	Plant protection equip.	2					1
		6B1 Nucl. inst/Rx prot. sys	2	0.58	2	0	1.16	1
		6B2 Saf-rel'd C&I	0					0
		6B3 Engrg saf feat act sys	0					0
--	1B	Fuel and rods	0					0
--	2A	Makeup and purification	0					0
--	2B	Decay heat/LPI	0					0
--	2C	Chem add'n and sampling	0					0
--	2D	Spent fuel cooling	0					0
--	2E	Reactor building spray	0					0
--	2F	Core flooding system	0					0
--	2G	Low-pressure serv. water	0					0
--	2H	Demineralized water	0					0
--	2I	Component cooling system	0					0
--	2J	Penetr. room vent	0					0
--	3E	Condenser circ. water	0					0
--	3F	Recirc. cooling water	0					0
--	3G	Auxiliary steam system	0					0
--	3H	Moist. sep/reheater	0					0
--	3K	Instrument air	0					0
--	3N	HP service water	0					0
--	3P	Nitrogen supply	0					0
--	3R	Vacuum	0					0
--	4D	Service structure	0					0
--	7A	Liquid waste disposal	0					0
--	7B	Gaseous waste disposal	0					0
--	7C	Solid waste disposal	0					0
--	7E	Coolant treatment	0					0
--	3Q	Steam drain system	0					0

Table 4-2. Oconee 1 Limiting Factors for Maintenance, July 1, 1974 - December 31, 1977

Rank		System/component	No. of events	Mean time to repair, manhours	LFM (normalized)
1	3A	Main turbines	1	926	3241
2	1D	RC pump motors	105	69	2061
3	1A	Reactor and internals	1	6202	1772
		Reactor	0		0
		Internals	1	6202	1772
4	1F	Steam Generator	12	468	1606
5	1C	Reactor coolant pumps	17	330	1603
6	3C	Feedwater system	118		952
		Valves	85	20	486
		Heaters	13	94	351
		Pumps	11	24	74
		Turbines	6	8	14
		Miscellaneous	3	32	27
7	5	Electrical Systems	23		697
		5A Generator	4	10	12
		5B Switchgear	0		0
		5C Controls	0		0
		5D Exciter	3	713.3	611
		5E Transformers	7	28	56
		5F Substations	0		0
		5G Isolated phase bar	0		0
		5H Batteries	0		0
		5I Chargers	9	7	18
8	3B	Main steam system	60		679
		Valves/valve tests	43	40	493
		Piping	17	38	186
		Air ejectors	0		0
9	3D	Condensate system	80		674
		Valves	67	28	544
		Pumps and motors	10	41	118
		Air ejectors	3	14	12
10	3J	Heater drain system	112		626
		Valves	67	23	431
		Tanks	30	12	103
		Pumps	14	21	84
		Coolers	0		0
		Pipe	1	30	8
11	2A	Makeup and purif'n system	95		613
		Valves	75	7.2	154
		Pump motors	8	10.6	24
		Pumps	7	74.8	150
		Filters	4	7	8
		Coolers	1	970	277
12	4E	Suppressors and hangers	45		482
		Hydraulic suppressors	43	37.1	456
		Pipe hangers	2	44	26
13	4A	Control rod drives	74		458
		4A1 Drives	9	19.6	50
		4A2 Stators	16	22.4	103
		4A3 Position indicators	18	26.8	138
		4A4 Power and T/C cables	3	98.7	85
		4A5 Closure/vent system	5	8.2	12
		4A6 CRD control system	23	10.7	70

Table 4-2. (Cont'd)

Rank		System/component	No. of events	Mean time to repair, manhours	LFM (normalized)
14	4B	Fuel handling bridges	42	35.7	428
15	6A	Control & monit. equip.	33		295
		6A1 Integr. cont. system	10	8.7	25
		6A2 Non-nuclear instr'n	19	21.4	116
		6A3 Incore detectors	4	134.5	154
		6A4 Computers	0		0
16	7E	Coolant treatment system	83		292
		Valves	61	6	105
		Piping	2	13	7
		Evaporator	6	68	117
		Pumps	14	15.7	63
17	7A	Liquid waste disposal	58		195
		Valves	29	7.6	63
		Evaporator	4	5.5	6
		Pumps	25	17.8	126
18	3Q	Steam drain system	57	12	194
		Valves	57	12	194
19	3H	Moisture separator/reheater	18	36	184
20	3L	Turbine lube oil system	35	17.5	175
21	1G	Pressurizer	24		156
		Valves	23	23.6	155
		Heaters	1	1	1
22	7B	Gaseous waste disposal	58		143
		Valves	26	7.5	55
		Compressor	21	11.5	69
		Transmitters	4	7.3	8
		Gas analyzer	1	1	1
		Vent header	5	7	10
23	3M	Turbine EHC system	10	35	100
24	2I	Component cooling system	11		97
		Valves	7	38.3	77
		Coolers	2	24	14
		Pumps/motors	1	8	2
		Pressure switch	1	15	4
25	2C	Chem Add'n & boron sampling	23		73
		Pumps	19	11.3	61
		Tanks	1	2	1
		Mixers	3	13.7	11
26	2B	Decay heat/LPI	32		65
		Valves	27	6	46
		Pumps	3	18	15
		Coolers	0		0
		Tanks	2	4.7	4
27	6B	Plant protection equipment	6		60
		6B1 Nucl inst/Rx prot system	6	34.7	60
		6B2 Safety-rel. cont & instr.	0		0
		6B3 Engr safety feat act sys	0		0
28	8A	Polar crane	3	61.3	52
29	3P	Nitrogen supply system	7		51
		Valves	7	25.7	51
30	3E	Condenser circ water	7		35
		Valves	2	18	11
		Pumps	5	17	24

Table 4-2. (Cont'd)

Rank		System/component	No. of events	Mean time to repair, manhours	LFM (normalized)
31	3I	Generator stator cooling	6		35
		3I1 Stator cooling water			
		Pumps	2	22.5	13
		Coolers	2	30	17
		Valves	0		0
		3I2 Hydrogen			
		Valves	2	9	5
32	7D	Coolant storage	20		32
		Valves	17	8.9	23
		Pumps	3	10.7	9
33	4C	Fuel transfer system	12	14	28
34	2J	Penetration room vent sys	10		26
		Valves	10	9.2	26
35	2G	LP service water system	5		17
		Valves	4	14.5	16.6
		Pump motor	1	2	1
36	3R	Vacuum system	6		17
		Valves	6	10	17
37	3N	HP service water system	2		14
		Pump motor	1	12	4
		Cooler	1	36	10
38	4D	Service structure	2		13
		Ductwork	1	8	2
		Fans	1	40	11
39	2E	Reactor building spray	6		12
		Valves	3	3	3
		Pumps	3	10.7	9
40	2D	Spent fuel cooling system	2		8
		Pumps	1	4	1
		Valves	1	24	7
41	2F	Core flooding system	6		8
		Valves	4	5.5	6
		Flow transmitters	2	4	2
		Tanks			
42	3F	Recirculated cooling water	2		7
		Pumps	2	13	7
		Valves	0		0
		Coolers	0		0
--	1H	Core physics and Rx safety	0		0
--	1B	Fuel and rods	0		0
--	2H	Demineralized water system	0		0
--	3G	Auxiliary steam systems	0		0
--	3K	Instrument air	0		0
--	7C	Solid waste disposal	0		0

Table 4-3. System-Related Limiting Factors — Oconee Units 1, 2, and 3 (1977)

Rank	System/component	No. of events	No. of units affected	Average limiting factor
1	1F Steam generator	9	3	1119
2	4A Control rod drive	22	3	434
3	1H Core physics and RX safety	60	3	313
4	1D RC pump motors	2	1	109
5	1C RC pumps	2	1	94
6	3L Turbine lubricating oil	6	3	75
7	3C Feedwater	9	3	66
8	1G Pressurizer	2	2	65
9	2A Makeup and purification/HPI	4	3	59
10	3D Condensate	5	2	49
11	2B Decay heat/LPI	3	2	42
12	3M Turbine EHC system	4	2	34
13	7A Liquid waste	3	2	30
14	3J Heater drains	4	2	27
15	3I Generator stator cooling	3	3	18
16	2C Chem add'n and sampling	1	1	16
17	7D Coolant storage	1	1	10
18	6A Control and monitoring equip.	4	3	10
19	2E Reactor building spray	1	1	10
20	3B Main steam	6	3	8
21	5A Generator (electrical)	1	1	8
22	3A Main turbine	2	2	5
23	3K Instrument air	1	1	2
24	6B Plant protection equipment	3	2	1
25	3H Moisture separator/reheaters	1	1	0

Table 4-4. LFR Analysis Where LFR = (Performance - Standard) × Critical Path Adjustments

		LFR = $(P - S)F_p$ Performance time (P) - critical/near-critical, h						
Rank	System/work activity	Oconee 1	Rancho Seco	Oconee 3	TMI-1	P — average refueling EFPH loss, h	Projected standard, hours — S	LFR[a]
1	OTSG tube inspection	1110	29	96	316	388	120 NC[b]	268
2	Remove, reinstall RC pump seals	256	244	208	174	220	96 NC	124
3	Refueling operations	385	2445	204	144	245	139	106
4	Check out fuel handling equipment	131	296	15	30	118	40 NC	78
5	Shutdown and startup	338	205	200	--	248	94	77[c]
6	Physics tests	253	232	--	149	211	72	70[c]
7	Secure/reinstall CRDMs	103	122	62	52	85	32	53
8	Detension/retension RV head	88	63	64	34	62	24	38
9	Clean transfer canal	38	94	--	28	36	8	28
9	Remove/reinstall incore detectors	142	24	90	60	79	51 NC	28
9	Containment leak tests	--	191	--	102	136	108	28
10	Reactor building purge	20	NA	NA	--	20	9 NC	11
11	Remove/reinstall RV head	31	30	28	18	28	18	10
11	Fill/drain transfer canal	21	24	28	50	31	21	10
12	Install/remove canal seal plate	23	14	18	20	19	12	7
13	Remove/reinstall plenum	31	17	20	28	24	18	6
14	Health physics survey	12	2	--	--	7	2	5
15	Move equip in/out of reactor bldg	32	0	24	28	28	24	4
16	Remove/reinstall RV head insul'n	14	12	10	12	12	9	3
17	Install/remove stud hole plugs	15	8	11	18	13	11	2
17	Remove/install shield blocks	17	5	16	--	13	11	2
17	ARIS work	59	--	44	--	52	50[d]	2

[a]All LFRs include a value of F_p = 1.0 except as noted.

[b]Does not include time for plugging tubes.

[c]F_p = 0.5.

[d]Does not include 80 hours assembly and disassembly time.

Legend

NC Non-critical path

NA Not applicable

-- No data

Table 4-5. Oconee 1 Clock Hours to Complete Selected Tube Plugging Events

Description	No. of men req'd	1976 Gen A 10/31	1976 Gen B 12/8	1977 Gen B 1/15	1977 Gen B 2/28	1977 Gen B 3/22	1977 Gen B 5/7	1977 refueling outage Gen A — 9/18 Crit'l path	1977 refueling outage Gen A — 9/18 Total	1977 refueling outage Gen B — 9/18 Crit'l path	1977 refueling outage Gen B — 9/18 Total	Δ Avg hours	Δ Avg manhours
Delays		(15)	(15)	17	12	12	13	2.5	2.5	2.5	2.5	11	--
Hydro test		(45)	(31)	17	24	45	56	NA	NA	NA	NA	36	--
Eddy curr., fl op test	8	(60)	(28)	26	60	20	74	176	240	86	578	136	1086
Tube plugging	4	(35)	(13)	8	42	16	20	34	34	90	90	32	129
Cut and pull tubes	4	NA	NA	NA	12	NA	7	NA	NA	68	113	44	176
Weld rep. on tube plug	2	NA	NA	35	NA	NA	NA	NA	NA	NA	NA	35	70
Leak acceptance test		(13)	(18)	11	12	18	10	NA	NA	46	46	18	--
Total clock hours		168	105	114	162	111	180	215	277	293	830	312	--
Total manhours		693	340	355	744	299	779	1563	2059	1369	5485	--	1461
No. of tubes plugged		2	4	3*	6	6	3	5	5	35	37	--	--

Legend: (xx) Estimated.

xx* Includes rewelding of one tube.

Δ Averages are calculated on total hours.

Table 4-6. List of Key Valves

Item	Valve name	Selection critera	Reference figure
1	Pressurizer spray control	A	4-2
2	Pressurizer spray control bypass	A	4-2
3	Pressurizer spray control block	B	4-2
4	Pressurizer power relief	A	4-2
5	Pressurizer power relief block	B	4-2
6	Pressurizer code relief	A	4-2
7	Pressurizer sample block	A	4-2
8	Letdown line relief	A	4-3
9	Makeup flow control	B	4-3
10	Letdown flow control		4-3
11	RC pump seal injection throttle		4-3
12	LPI pump-BWST isolation	A	4-4
13	Decay heat letdown isolation	B	4-4
14	Main steam throttle (stop)	A	4-5
15	Main steam code safety	A	4-5
16	Turbine governor	B	4-5
17	Mian steam bypass to condenser		4-5

Selection criteria: A — Caused or extended a power reduction on at least two plants.

B — Caused or extended a power reduction on at least one plant and was identified as a high maintenance valve.

Table 4-7. Valve Failure Categorization

Notes

BF: butterfly.

CK: check.

GLOBE VALVE — Failure Category, Repairs/Valve

Vendor	Size	No. of Valves	Valve: Body-to-Bonnet	Valve: Seat	Valve: Packing	Valve: Other	Operator: Limit/Torque Sw.	Operator: Motor/Solenoid	Operator: Gearbox	Operator: Other	Replaced
Rockwell		11		.1	.7	.3	.2	.4		.4	.2
Dresser											
Crosby											
Fisher		21	.7	.4	.1	.6	0	0	.1	.2	
Lonergan											
Walworth											
Velan		30	.3	.4	.5	.1				0	.2
Alovco											
Crane		8	.5	.5	.4						.1
Crane-Chapman											
Norris											
Graver											
Grinnell		11				.1	.5	1.0			
Rockwell-Edward		3		.3	1.6		1.0		.3		
Valve Repairs Categorized by Valve Size											
0.24 to 1.5 inch		39	.2	.3	.3	.1	0	.1	0	.2	.2
2.0 to 3.0 inches		27	.4	.3	.5	.2	.3	.4	0	0	.1
3.5 inches and up		18	.6	.4	.5	.5	.2	.1	.1	.3	.1
TOTAL		84	1.2	1.0	1.3	0.8	0.5	0.6	0.1	0.5	0.4

GATE VALVE — Failure Category, Repairs/Valve

Vendor	No. of Valves	Valve: Body-to-Bonnet	Valve: Seat	Valve: Packing	Valve: Other	Operator: Limit/Torque Sw.	Operator: Motor/Solenoid	Operator: Gearbox	Operator: Other	Replaced
Rockwell	4	.1			.1	.1	.1			
Dresser	1		1		4					
Crosby										
Fisher	1									
Lonergan										
Walworth	10			.5		.8	.1	.1	.3	
Velan	18	.4	.5	.3						
Alovco	4					.8				
Crane	40	.4	.2	.3	.3	,2				
Crane-Chapman	8	.5	.3	.3	.3					
Norris										
Graver										
Grinnell										
Rockwell-Edward										
Valve Repairs Categorized by Valve Size										
0.24 to 1.5 inch	14	.4	0	.1	.1	.1	0	0	0	0
2.0 to 3.0 inches	11	.2	.2	.1	.4	.3	0	0	.2	.1
3.5 inches and up	61	.3	.2	.3	.3	.3	0	0	0	0
TOTAL	86	0.9	0.4	0.5	0.8	0.7	0	0	0.2	0.1

RELIEF VALVE — Failure Categ, Repairs/Valve

Vendor	No. of Valves	Seat	Packing	Other	Replaced
Rockwell					
Dresser	3	1.3		1	
Crosby	16	.3			
Fisher					
Lonergan	5	1		2	
Walworth					
Velan					
Alovco					
Crane					
Crane-Chapman					
Norris					
Graver					
Grinnell					
Rockwell-Edward					
Valve Repairs Categorized by Valve Size					
0.24 to 1.5 inch	1	0	0		
2.0 to 3.0 inches	5		1	0	.8
3.5 inches and up	18		.4	0	0
TOTAL	24		2.4	0	0.8

OTHERS — Failure Categ, Repairs/Valve

Vendor	No. of Valves	Type	Seat	Packing	Other
Rockwell	2	BF	.1		.1
Dresser					
Crosby					
Fisher	5	BF	.8		1.4
Lonergan					
Walworth					
Velan	7	CK	.7		.3
Alovco					
Crane	8	CK	.3	.1	.8
Crane-Chapman					
Norris	7	BF	.9		.4
Graver	18	BF	.1	.1	1.5
Grinnell	39	BF	1.0		.7
Rockwell-Edward					
Valve Repairs Categorized by Valve Size					
0.24 to 1.5 inch	22		1.3	0	1.7
2.0 to 3.0 inches	28		.6	0	.5
3.5 inches and up	36		.3	.1	1.2
TOTAL	86		2.2	0.1	3.4

Table 4-8. Limiting Factor for Operation for Valves Based on Oconee 1 Historical Data

Rank	System	No. of key valves	No. of events	LFO annualized
1	1G Pressurizer	12	1	40
2	3B Main steam	51	2	10
3	3J Heater drain	42	1	9
4	7D Coolant storage	14	1	9
5	Others	--	0	0
	Total	119	5	68

$$\text{Avg events/component-year} = \frac{5}{119 \times 3.5} = 0.012$$

$$\text{Avg LFO/component-year} = \frac{68}{119} = 0.57$$

Table 4-9. Limiting Factor for Maintenance for Valves Based on Oconee 1 Historical Data

Rank	System	No. of key valves	No. of events	Events per component	LFM[a]	LFM per component[a]
1	3D Condensate	43	67	1.6	544	13
2	3B Main steam	51	43	0.8	493	10
3	3C Feedwater	47	85	1.8	486	10
4	3J Heater drain	42	67	1.6	431	10
5	3Q Steam drain	40	57	1.4	194	5
6	1G Pressurizer	12	23	1.9	155	13
7	2A MU&P	36	75	2.1	154	4
8	7E Coolant treatment	16	61	3.8	105	7
9	2I Component cooling	5	7	1.4	77	15
10	7A Liquid waste	29	29	1.0	63	2
11	7B Gaseous waste	18	26	1.4	55	3
12	3P Nitrogen supply	4	7	1.8	51	13
13	2B Decay heat/LPI	20	27	1.4	46	2
14	2J Penetr room vent	2	10	5.0	26	13
15	7D Coolant storage	14	17	1.2	23	2
16	3R Vacuum	6	6	1.0	17	3
17	2G Low Pressure Serv.	13	4	0.3	17	1
18	3E Cond Circ Water	10	2	0.2	11	1
19	Others	$\leq$10	<5	~0.5	<10	1
	Total	418	618	30.2	2958	128

$$\text{Avg events/component-year} = \frac{618}{418 \times 3.5} = 0.42$$

$$\text{Avg LFM/component-year} = \frac{2958}{418} = 7.1$$

[a]Annualized.

Table 4-10. 1977 Routine Work Dose Summary — Oconee Units 1, 2, and 3

Work category	Dose, man-rems					
	1st Q	2nd Q	3rd Q	4th Q	total	%
Station maintenance	31.25	55.43	50.36	32.04	169.08	41.4
Station surveillance (Insp, oper.)	21.38	31.40	43.93	22.08	118.79	29.1
Filter change operation, disposal	4.33	7.57	14.69	10.36	36.95	9.1
Radioactive waste handling, disposal	12.43	6.16	10.67	5.19	34.45	8.5
General cleanup, decontamination	3.49	4.74	8.28	4.80	21.31	5.2
Resin sluice flush modification	--	--	13.45	--	13.45	3.3
Spent fuel handling, shipping	6.07	3.91	1.52	1.33	12.83	3.1
Fuel sipping—Units 1 & 2 SF bldg	--	--	--	0.49	0.49	0.1
Miscellaneous	--	0.79	--	--	0.79	0.2
Total dose, man-rems	79.74	109.21	142.9	76.29	408.14	--
Percentage	19.5	26.8	35	18.7	--	100

Table 4-11. 1977 Special Shutdown Work Dose Summary — Oconee 1, 2, and 3

Work category	Dose, man-rems				
	Unit 1	Unit 2	Unit 3	Total	%
OTSG tube leak test, plugging	171.96	15.05	33.48	220.49	25.6
General entry, misc. work	79.86	98.26	31.38	209.50	24.3
SG tube eddy-current testing	26.05	13.54	21.30	60.89	7.1
RV head removal, replacement	26.77	14.25	15.34	56.36	6.5
Primary valve repair or replacement	26.20	11.51	5.14	42.85	5.0
Defueling/refueling operations	16.24	15.99	4.62	36.85	4.3
Inservice inspection	20.69	7.58	8.35	36.62	4.2
2nd-of-a-kind instr install. in 2B OTSG		36.5		36.5	4.2
RC pump motor repair	13.54	15.46	3.91	32.91	3.8
Nuclear station modification work	13.97	9.56	5.16	28.69	3.3
Incore instrumentation work	18.19	2.82	2.68	23.69	2.7
RC pump seal inspection, repair	10.97	8.01	1.99	20.94	2.4
Letdown cooler replacement	10.07	8.97		19.04	2.2
RV head work on storage stand	3.30	11.61	3.55	18.46	2.1
General cleanup, decontamination	7.37	5.06	3.18	15.61	1.8
Pipe hanger inspection			2.76	2.76	0.3
Repair upender Unit 1 & 2 SF pool	0.17			0.17	0.2
Total dose, man-rems	445.32	274.17	142.84	862.33	--
Percent	51.6	31.8	16.6		100

Table 4-12. Oconee Units 1, 2, and 3 — Radiation Exposure in 1974-1977

TOTAL ACCUMULATED DOSE, MAN REM	1974	1975	1976	1977
ROUTINE WORK	162	342	240	408
SHUTDOWN WORK.	350	113	754	862
TOTAL	512	455	994	1270

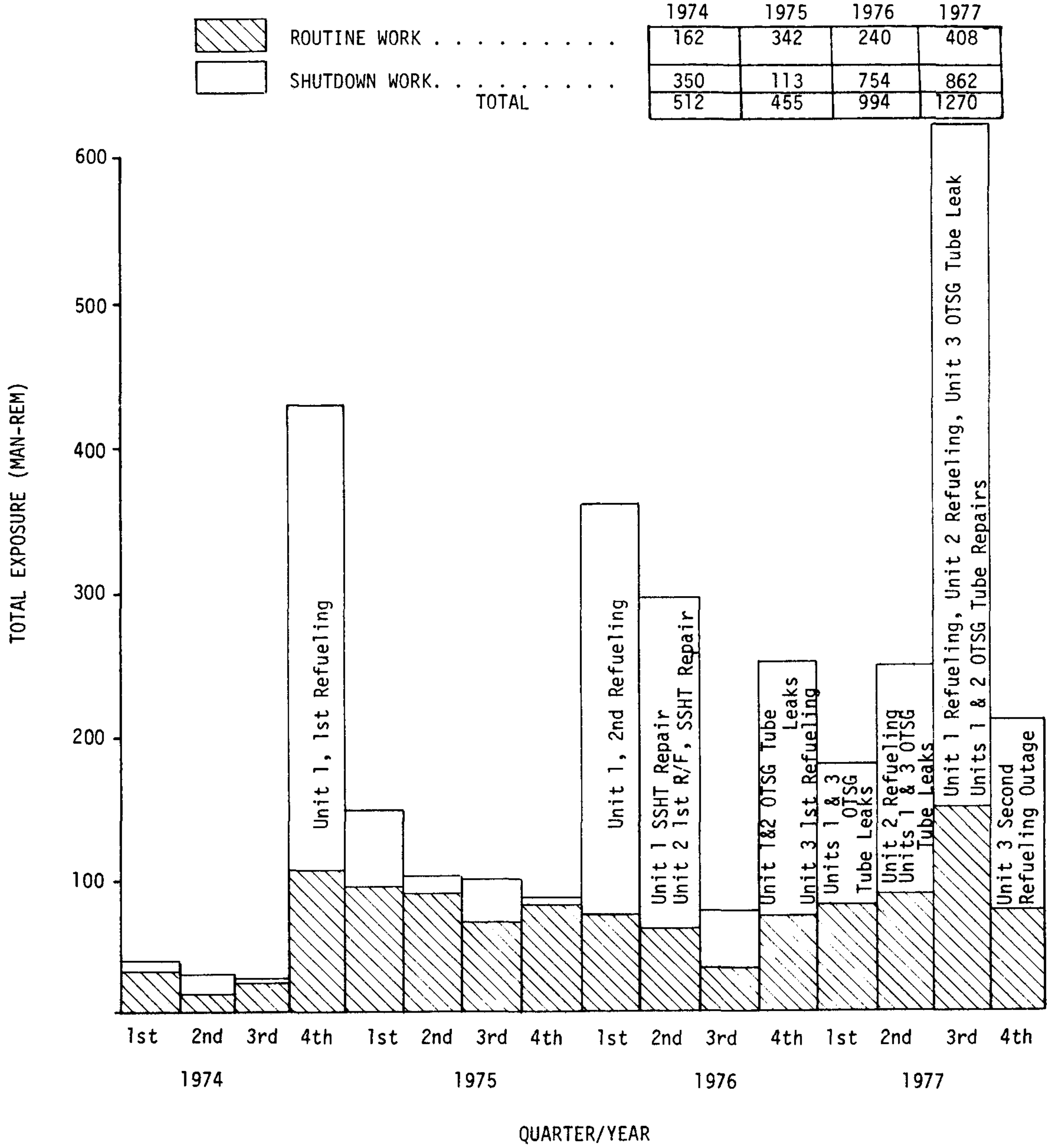


Table 4-13. Limiting Factors for Maintenance for Pumps/Motors Based on Oconee 1 Historical Data

Rank	System	No. of components	No. of events	Events/ component	LFM[a]	LFM/ component
1	1C, 1C pumps/motors	4	122	31.0	3664	916
2	2A Makeup and purification	3	15	5.0	174	58
3	7A Liquid waste disposal	10	25	2.5	126	13
4	3D Condensate	6	10	1.7	118	20
5	3J Heater drain	8	14	1.8	84	10
6	3C Feedwater	3	11	3.7	74	25
7	7E Coolant treatment	4	14	3.5	63	16
8	2C Chem add'n/boron recovery	3	19	6.3	61	20
9	3E Condensate circ. water	3	5	1.7	24	8
10	2B Decay heat/LPI	3	3	1.0	15	5
11	3I Generator stator cool.	2	2	1.0	13	6
	Others	≤3	≤3	--	<10	--
	Total (exclusive of RC pumps/motors and "others")	45	118	28.2	752	181

$$\text{Avg events/component-year} = \frac{118}{45 \times 3.5} = 0.75$$

$$\text{Avg LFM/component-year} = \frac{752}{45} = 16.7$$

[a]Annualized.

Table 4-14. Limiting Factors for Maintenance for Heat Exchangers[a] Based on Oconee 1 Historical Data

Rank	System	No. of components	No. of events	Events/ component	LFM[b]	LFM/ component[b]
1	3C Feedwater	13	13	1.0	351	27
2	2A Makeup and purification	2	1	0.5	277	138
3	3H Moisture sep/reheater	4	18	4.6	184	46
4	3I Generator stator cooler	2	2	1.0	17	8
5	2I Component cooling	2	2	1.0	14	7
6	3N HP service water	1	1	1.0	10	10
	Total	24	37	9.1	853	236

$$\text{Avg events/component-year} = \frac{37}{24 \times 3.5} = 0.44$$

$$\text{Avg LFM/component-year} = \frac{853}{24} = 35.3$$

[a]Includes heaters, reheaters, and coolers.

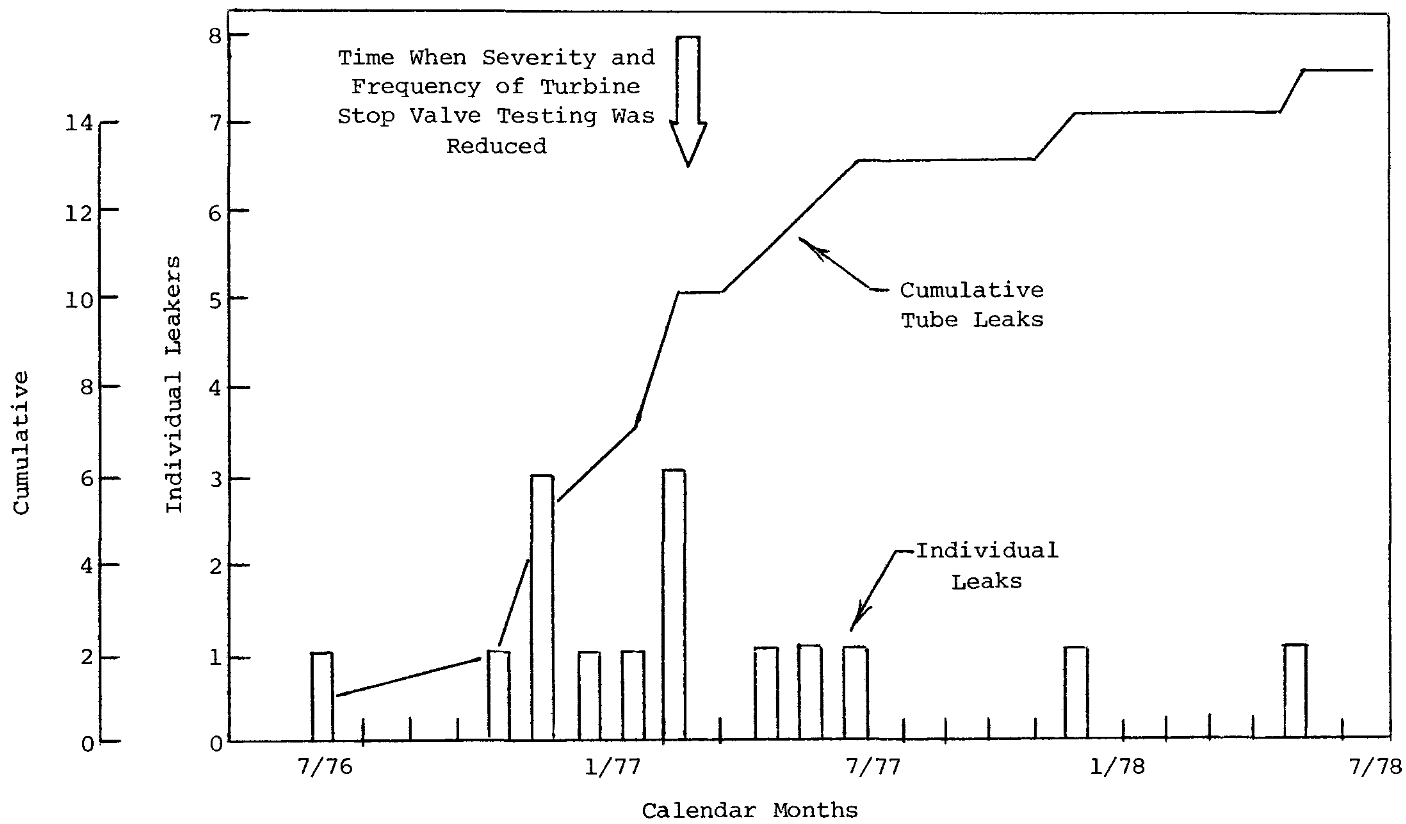
[b]Annualized.

Table 4-15. Standard Projected Performance Times

Required activity	Critical path hours	Non-critical path hours	Total
1. Shutdown/startup	94	0	94
2. Reactor building purge	0	9	9
3. Health physics survey	2	0	2
4. Move equipment in/out reactor bldg	24	0	24
5. Remove/replace RC pump seal	16	80	96
6. Remove/install shield blocks	11	0	11
7. Clean transfer canal	8	0	8
8. Remove/install incore detectors	9	51	51
9. Remove/install RV head insulation	9	0	9
10. Install/remove canal seal plate	12	0	12
11. Detension/retension RV head	24	0	24
12. Secure/reinstall CRDMs	0	32	32
13. Remove/reinstall RV head	18	0	18
14. Remove/reinstall plenum	18	0	18
15. Install/remove stud hole plugs	11	0	11
16. Fill/drain transfer canal	21	0	21
17. Check out fuel handling equipment	0	40	40
18. Refueling operations	139	0	139
19. Inspect steam generator tubes	0	120[a]	120[a]
20. Exercise vent valves	1	0	1
21. Cont. leak tests (every 3rd year)	80	28	108
22. Physics tests	72	0	72
23. System alignment and checkout	72	0	72
24. ARIS (periodic)	40	90	130

[a]Includes no time for plugging tubes.

Figure 4-1. Oconee 1, 2, and 3 Steam Generator Tube Leaks



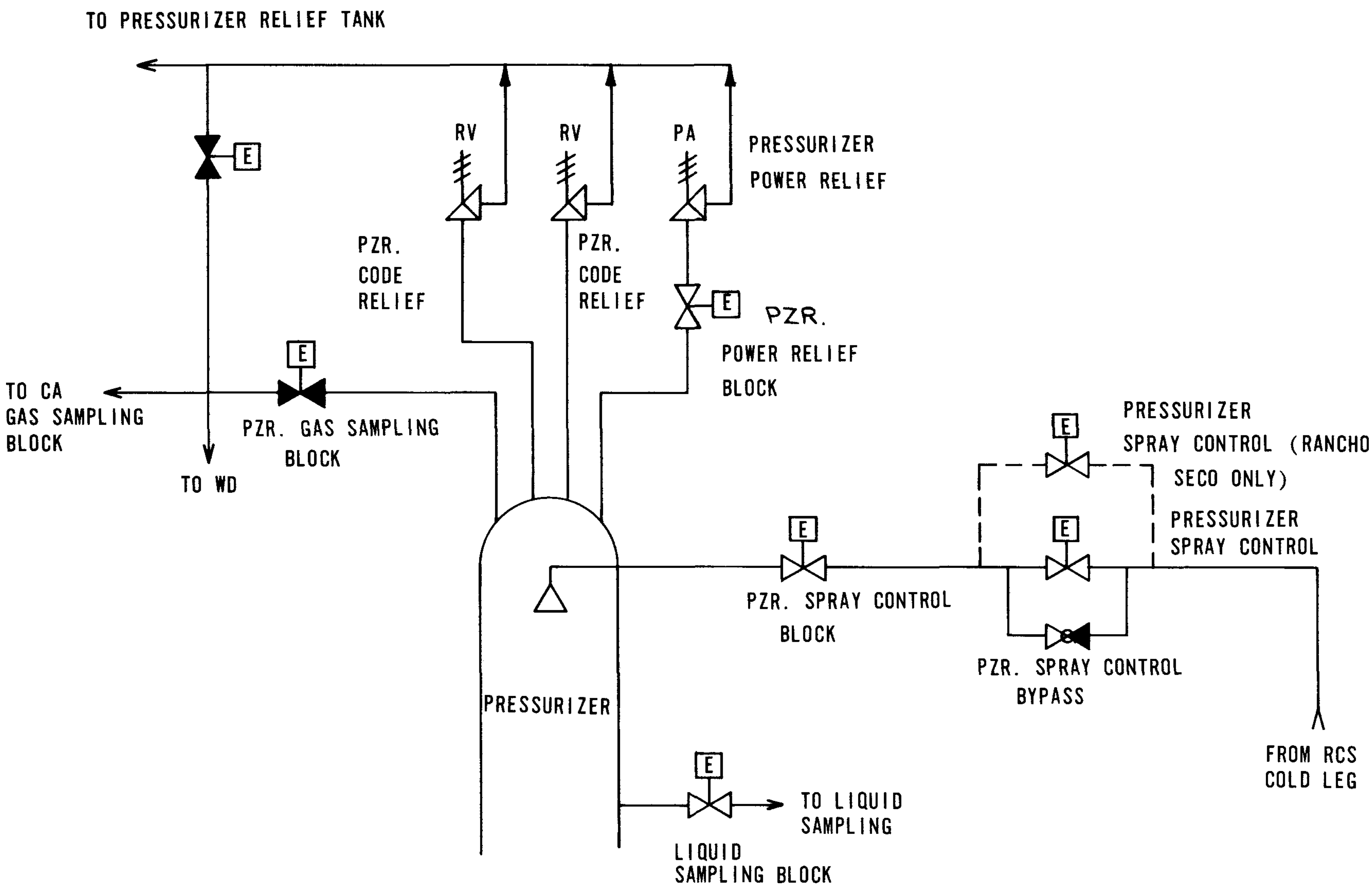


Figure 4-2. Key Valves in the Reactor Coolant System

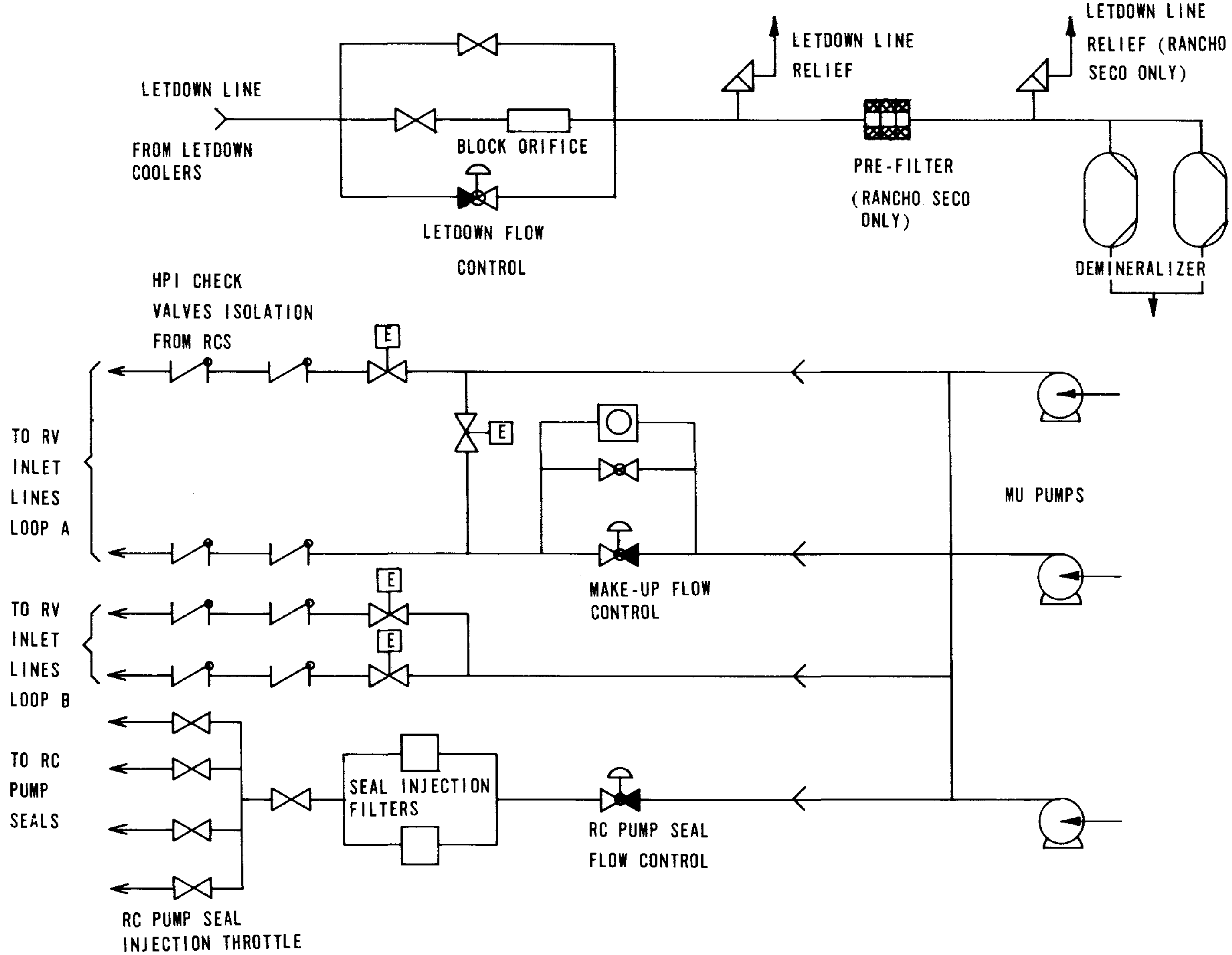


Figure 4-3. Key Valves in the Makeup and Purification System

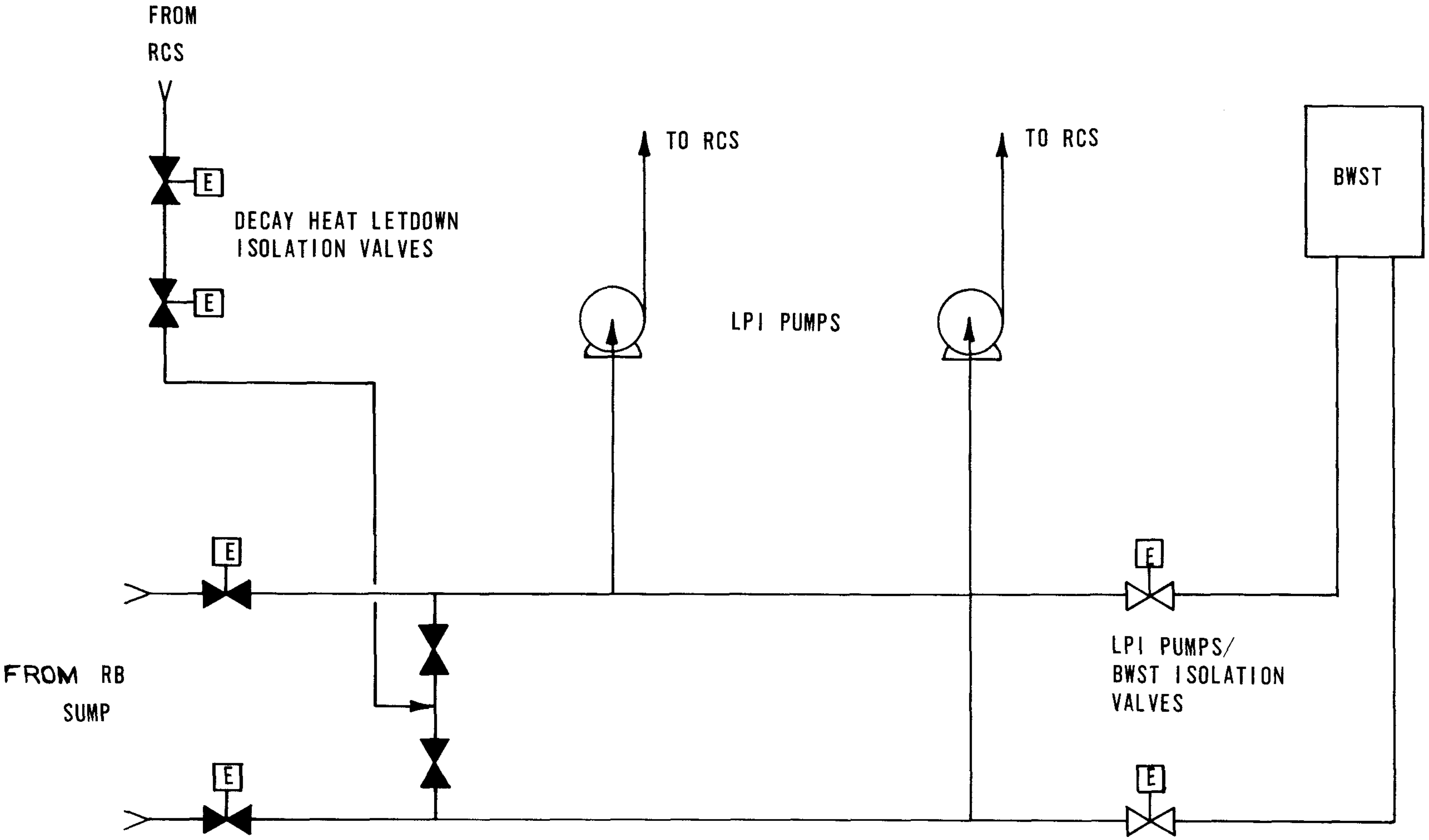


Figure 4-4. Key Valves in the Decay Heat System

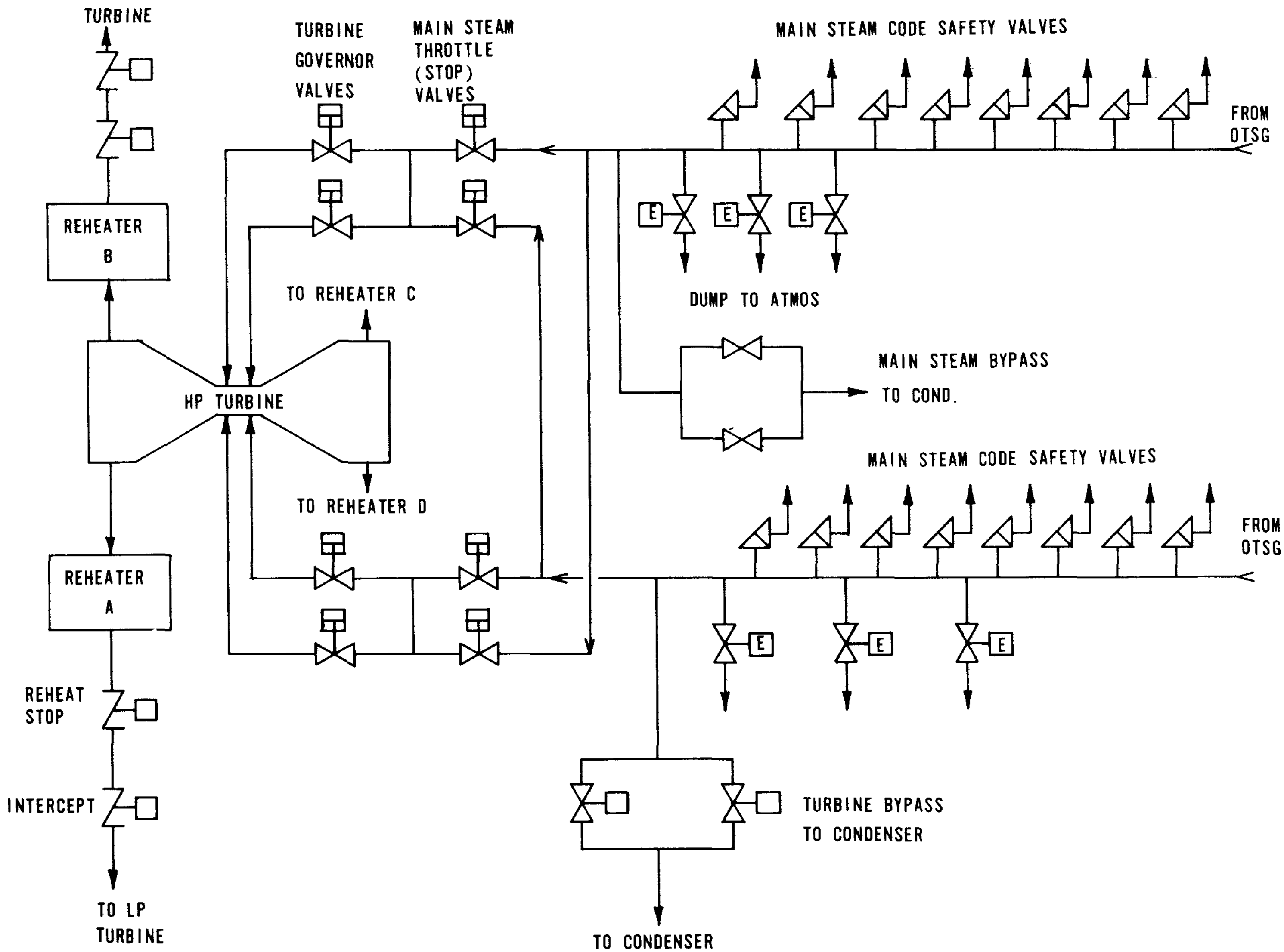


Figure 4-5. Key Valves in the Main Steam System

APPENDIX A

Study Concept

Contents

1. Background

During 1977 EPRI awarded contracts to study availability-limiting factors in LWR nuclear power plants. The team selected to study plants having B&W nuclear steam systems consisted of the following:

- B&W, representing the NSS supplier.
- Duke Power Company's Design Engineering Department, representing the architect-engineer.
- Duke Power Company's Steam Production Department, representing the utility operator.

2. Objectives

The objectives of this study were to identify factors that limit plant availability, to determine the extent of their impact on plant performance, and to determine what could be done to improve future designs or incorporate backfits in existing plants to alleviate these limitations. A further objective was to provide a data base of PWR information that could serve as a focal point for identifying additional programs that could improve future plant availability. An analysis of these data could identify items where design improvements could reduce the availability-limiting factor. Many of the limiting factors identified could become the subject of future R&D projects to solve the particular problems. Advanced PWR designs could also incorporate the major elements of high-availability design determined in this study.

3. Scope and Limitations

This study was intended to be the first phase of a series of availability improvement programs that would reduce the number of failures, the time between failures, the amount of time to perform maintenance, and the number of persons required to perform the maintenance. These reductions translate into increased plant availability and less radiation exposure — a fundamentally desirable secondary benefit.

The Oconee Nuclear Station, Unit 1 (Oconee 1), owned and operated by Duke Power Company, was selected as the reference plant for this evaluation. The project team obtained Oconee 1 records of 1977 operations and historical records back to July 1, 1974, and compiled data from these records for use in this analysis. Data from 1977 Oconee 2 and 3 operations were also collected to identify additional comparative information. The team observed major operations during the 1977 Oconee 1 refueling outage. These observations were supplemented by observations of selected operations in the 1977 refueling of Rancho Seco, the Sacramento

Municipal Utility District (SMUD) plant. Additional records of the 1977 Rancho Seco refueling were also studied, as were the records for the 1977 Three Mile Island Unit 1 (TMI-1) and the Oconee 3 plants.

As a special addition to the prime contract, 17 "key" valves that had a record of impacting plant availability were studied. Data for this special study were also obtained from the Oconee, Rancho Seco, and TMI-1 plants. To substantiate and supplement data obtained from these plants, interviews were held with operations, maintenance, and engineering personnel at the three reference plants, with B&W Engineering and Nuclear Services personnel, and with valve and valve operator vendors.

The evaluation of all collected information included identifying the causes of outages and power reductions and identifying all activities performed that necessitated plant shutdown or a power reduction. Evaluation of the refueling observations from outage planning through refueling and into subsequent operation afforded the opportunity to identify work items that may not have otherwise been identified.

Other availability-limiting factors were identified from such sources as studies sponsored by Edison Electrical Institute (EEI), EPRI, and the Department of Energy (DOE). Such public data sources as Nuclear Plant Reliability Data Systems (NPRDS), EEI, the Nuclear Regulatory Commission (NRC), and DOE contained valuable information on plant availability and have been used by EPRI and others as a basic source of data. In all cases, however, it has been recognized that the data are limited in scope and detail to that required by the objectives of the compiling organization. As examples, (1) NPRDS coverage is limited to plant safety systems and components and the depth of detail is not intended to give information on root causes of availability-limiting problems; (2) the EEI data bank is primarily intended to provide information on power generation and power distribution; and (3) the NRC data systems, like NPRDS, are primarily limited to safety-related and regulatory items and do not contain details and root causes. DOE has conducted an availability study which is similar in some respects to this study, but the coverage of that study has thus far been limited to refueling outages. The DOE study was initially directed only to the primary side, but it was later expanded to include a study of the secondary side. This study was intended to be complete and all-encompassing without the limitations of the data bases described above.

4. Project Team Concept

During the first phase of the project, a project team was organized; it comprised the following:

- A technical project manager from the NSS supplier (B&W), who was responsible for overall planning, scheduling, and coordinating the project in accordance with EPRI guidelines and contract cost limitations.
- A second key person from the NSS supplier, who was responsible for assisting the technical project manager and who also provided additional technical leadership and guidance.
- Additional technical personnel from the NSS supplier as required to effect timely completion of the work scope, including collection, reduction, and analysis of the data.
- A coordinator from Duke Power Company to coordinate the Duke/B&W efforts and to integrate Duke's dual role as A/E and the operator. The Duke coordinator was also responsible for assisting in project planning and for assigning Duke personnel as required to support the team objectives.
- Duke operations and maintenance engineers responsible for data collection at the Oconee plants also served as team advisors on plant design problems.
- Other Duke personnel as required.

5. Work Planning

After the study team was formed, detailed work plans were formulated and the procedures for close cooperation and agreement between the team members and EPRI were established. Specifically, the following were finalized:

- The project team was formed and team members assigned specific tasks.
- Division of investigative responsibilities was defined.
- A detailed work scope/schedule was established.
- The existing data base was identified.
- The procedures for data collection and handling were established.
- A format for data analysis and reporting for monthly reports of costs and technical progress was developed to compare with the detailed project plan.
- Definitions to be used in reporting were set.

APPENDIX B

Data Collection Methodology

Contents

Figures

1. Introduction

The evaluation of the operational history of the Oconee 1 plant was undertaken to identify the causes of outages and power reductions. Oconee 1 was the first nuclear plant supplied by B&W and also the first nuclear plant operated by Duke. These initial operations of Oconee 1 involved much initial training, testing, and component shakedown work. It was decided that data during this phase (prior to July 1, 1974) would not be representative of today's problems and therefore should not be included in this study. Thus, Oconee 1 historical data are defined as data from operations during the period July 1, 1974, through December 31, 1977.

The project team performed a search to identify all sources of availability-related data. Evaluation of these sources showed that the most useful data came from the following sources:

- Station work requests.
- Personal interviews and observation.
- B&W in-house records on plant performance.
- Operating unit status reports (NRC Gray Book).
- Utility reports to EEI.

The following secondary sources of data had varying degrees of usefulness:

- Operation, maintenance, health physics, and other station logs and files.
- Public records and reports, such as annual and semi-annual reports to the NRC, abnormal reports, license event reports, nuclear power experience reports, and the like.
- Nuclear Plant Reliability Data System (NPRDS).

Of all the data sources used, station work requests provided the most complete record of work performed. Figures B-1 and B-2 show a representative sample of a work request written at the Oconee 1 station for repairing a valve packing leak. As the name implies, work requests were originally used by planners and schedulers to assign and implement necessary maintenance and repairs and not as a tool for recording work performed. Consequently, older work requests often gave only estimated hours and an estimated number of men needed to complete a task. Only problem symptoms were often given, with little or no information on the nature of the failure or the repairs.

Personal interviews and observations provided valuable supplements to work requests. Data obtained by personal observation were especially useful during refueling outages and permitted the project team to identify delays caused by inadequate tools, manpower, procedures, etc., that would otherwise have gone undetected. Interviews were most valuable when more in-depth information was needed on a particular problem, such as for valves investigated in the in-depth valve study. Comparison of data from these sources with other primary data sources, such as the NRC's Operating Units Status Report (Gray Book) and EEI reports protected against omissions and misinterpretation. Secondary data sources were used to supplement the primary ones. To effect comparison of data from different sources and to help relate maintenance data and times to reactor power, the plant outage data were generally tabulated on charts as a power histogram (Figure B-3). These power histograms readily identify power reductions for such events as fuel maneuvering limits, such as xenon hold, for which no work requests are written. Figure B-3 is a representative plot of a power histogram comparison with three primary data sources. Other power histograms prepared in our study included data from more than three data sources.

2. Data Collection

With data being collected from many sources, as in this study, there were occasions where conflicts and differences were noted. In other cases, certain data sources contain more useful detail than others, such as the station work requests. Considering these facts, the following priority guidelines were established and used:

1. Station work requests
   a. Actual values for number of men, hours, etc.
   b. Planner's estimate for men, hours, etc.
   c. EPRI project team estimates.
2. EPRI project team notes from observations, discussions, log books, station records, etc.
3. NRC Gray Book/License Event Reports
4. EEI reports
5. B&W internal documents

More recent data were found to be more complete than older records. This is primarily because of the more formally regulated business environment of today, wherein work is more often accomplished by written/documented requests rather than by verbal/informal requests as was often done in the past.

EEI does not require (and Duke does not report) partial outages that cause generation reductions of less than 2% or less than 435 MW. Nor does Duke report partial outage load reductions for economy, efficiency, or system demand if the system is available to produce rated load. Similar criteria were used in this study to evaluate partial outage data; that is, power reductions that are less than 2% of full power generally are not reported and any power reduction for economy, efficiency, or system demand is considered a management decision and is not considered in the analysis.

Collection of data was limited to "key items," which are defined as those systems or components that

- Caused or extended or could have caused or extended a power reduction.
- Were critical path or near-critical path during the refueling outage.
- Resulted in high personnel radiation exposure.
- Had frequent repetitive maintenance.

Data were collected and broadly categorized as current data, historical data, refueling outage data, additional plant data, future outage data, and valve data. These categories are defined and described further in the following sections.

2.1. Current Data

"Current data" are defined as the non-refueling data obtained from Oconee Units 1, 2, and 3 during the 1977 data collecting phase of this project. Generally, these were the most detailed and most accurate of any data collected because the information was obtained by the project team during the outages specifically for this study. Also, if questions arose about any event, the occurrence usually was recent enough that operations and maintenance personnel could provide further details. Current data are described and presented in the form of power history/work activity histograms in Appendix D.

2.2. Historical Data

"Historical data" are defined as operational data from Oconee 1 for the period from July 1, 1974, through December 31, 1977. Included are both non-refueling and refueling outage data. These data were obtained entirely from historical records, such as work requests, EEI reports, and NRC Gray Books. The July 1, 1974, date was selected as the beginning of the historical data period because work requests before this date were generally not available. This date was approximately the middle of the first fuel cycle. Operations during the first half of the first fuel cycle were considered a shakedown phase for both the B&W NSS and Duke Power Company operations and were not considered to be representative of later operation. Historical data are described further and presented in Appendix E.

2.3. Refueling Outage Data

"Refueling outage data" are defined as data obtained during the annual refueling/maintenance outage. The primary source of refueling data is Oconee Unit 1 during the refueling outage period, which is defined as starting with power reduction on August 5, 1977, and ending when the unit reached 75% thermal power on October 29, 1977. These data were obtained primarily by a team of observers (project team and support personnel) assigned to the Oconee site during the refueling outage. The primary responsibility of the team was to observe critical path, near-critical path, and potential-critical path work activities on both the primary and secondary sides to determine and document the following:

- Time to complete the work activity.
- Delay times caused by such problems as lack of spare parts, inadequate equipment, inadequate procedures, scheduling, etc.
- Suggested changes and improvements, especially those that could shorten the outage time, improve working conditions, and reduce man-Rem exposure.

A brief study of past refuelings and consultation with service and operation personnel identified likely critical path operations and pinpointed other key operations that could possibly lead to critical path delays if unanticipated abnormalities occurred. The primary side critical path operations were identified as follows:

- Reactor shutdown and startup.
- Reactor building purge.
- Health physics survey.
- Moving equipment in/out of reactor building.
- Removing/installing shield blocks.
- Cleaning transfer canal.
- Removing/installing RV head insulation.
- Installing/removing canal seal plate.
- Detensioning/retensioning RV head.
- Securing/reinstalling CRDMs.
- Removing/reinstalling RV head.
- Installing/removing stud hole plugs.
- Filling/draining transfer canal.

- Handling fuel and control components (replacement/shuffle).
- Refueling operations.
- Exercising vent valves.
- Inservice inspection [automatic reactor inspection system (ARIS)].
- Containment leak tests (optional).
- Startup physics tests.

Non-critical operations that could potentially enter the critical path were identified as listed below. No secondary plant work was expected to be on the critical path.

Primary Side

- Removing and replacing RC pump seals.
- Removing and installing incore detectors.
- Checking out fuel transfer system.
- Steam generator tube inspections/repair.

Secondary Side

- Station turbine.
- Upper surge tank maintenance.

After the observation plan was established for each of these observations, personnel were selected to match their expertise with the operation to be observed wherever possible. A coordinator was assigned for each of the three working shifts. The number of observers assigned to each shift varied depending on the activities occurring during the shift. A three-shift observation team was maintained at Oconee 1 up to the beginning of fuel shuffle. The number of observers was then reduced to provide refueling activity coverage. After the refueling operation was completed, observers were assigned to cover the remaining critical path (or near-critical path) activities.

The observers recorded working and delay times, manpower employed, area radiation levels, and working conditions (i.e., temperature, accessibility, lighting, tooling, worker utilization and preparation, and support from complementary work units). In many cases, some of the data were not readily available, but sufficient data were gathered to ascertain the prevailing conditions.

Current operating procedures for the activities being studied were reviewed by the observers. A member of the observation team attended the daily refueling briefing sessions and in turn informed the other team members of the refueling plans for the day. If a portion of the operation was not observed, the missing data were reconstructed through communications with workers and/or supervisors. Where data came from sources other than direct observation, an attempt was made to verify the data from at least one other source. In general, data were obtained from the following sources:

- Direct observation.
- Station logs.
- Worker/supervisor communication.
- Individual personal logs.
- Unit Plan-a-Log.

Refueling outage data are presented in Appendix F.

2.4. Additional Plant Data

2.4.1. Refueling Data

A few weeks after the Oconee 1 refueling outage began, an observer from the Oconee 1 team went to the Rancho Seco site to observe portions of that refueling operation so that information being accumulated at Oconee 1 could be compared with Rancho Seco. Observations at Rancho Seco began after reactor cooldown and continued until that reactor was near fuel movement operations. No attempt was made to observe actual fuel movement.

Additional observations were resumed as the reactor equipment was being reassembled until this operation was stopped to run the containment leak tests. Additional data were obtained from the station logs, supervisors' logs, and communication with Rancho Seco refueling team members to complete the refueling summary. Note that at Rancho Seco, the complete turbine-generator inspection was on the critical path throughout the outage. The observers recognized this, but for the purpose of this study assigned critical path activities to the NSS as if it were a normal refueling.

In addition to the studies of the Oconee 1 and Rancho Seco refueling activities, records of the TMI-1 and Oconee 3 refuelings in 1977 were studied. Both of these units had better-than-average refueling times, and if abnormal events had not occurred, the refuelings could have been completed within about one month. The records gave a fairly true picture of daily events as they occurred. In some cases, however, time estimates had to be made. The refueling activities for these additional plants are given in Appendix F.

2.4.2. Non-Refueling Data

Additional plant data on selected problems were obtained from the Rancho Seco plant for the period of October 1, 1976, through December 31, 1977, and to a lesser extent from TMI-1 from September 1, 1975, through December 31, 1977. The basis for selecting these periods was similar to the basis for selecting the Oconee 1 period, that is, the first half of the first full cycle was considered as a shakedown phase. As mentioned above, data from these two additional plants (and from Oconee 2 and 3) were obtained to serve as a check on the Oconee 1 data and to identify problems that had not occurred at Oconee 1. Data from Rancho Seco and TMI-1 had the additional advantage of reflecting different management and architect-engineer design philosophies. Data from these plants are included in the system writeups (section 4.2).

2.5. Future Outage Data

"Future outage data" are those of known or expected outages that could occur at a future date and impact plant availability. These data are included on a case-by-case basis in the writeups of section 4.

2.6. Valve Data

As an addendum to this plant availability study, a special in-depth study was made of plant "key valves." Key valves were defined in accordance with our earlier definition of key items (see section 4.4 and Appendix B, part 2). Seventeen of the more important key valves were selected for this special study by:

1. Requesting the utilities participating in the availability-limiting factor contract to supply a list of valves that have caused or extended plant outages. The participating utilities are Duke Power Co. (Oconee Units 1, 2, and 3), Sacramento Municipal Utility District (Rancho Seco Unit 1), and General Public Utilities (Three Mile Island 1).

2. Generating a list of key valves from our availability-limiting factor data.

3. Using data and conclusions from an internal B&W study of B&W-supplied valves and valve operators for all operating B&W plants.

Additional key valve data were obtained by:

1. Conducting an internal B&W review of the key valves identified in items 1-3 above to identify design and manufacturing information and inservice conditions of key valves.

2. Making plant visits to consult with participating utility personnel to confirm and supplement information obtained from other sources.

3. Consulting with appropriate valve and valve operator manufacturers.

SEE STAT. DIR. 3.3.5 FOR DETAILS. COMPLETE BLANKS AND CHECK BLOCKS THAT APPLY. INITIAL AND DATE ALL CORRECTIONS.

DUKE POWER — STEAM PRODUCTION DEPARTMENT NUCLEAR STATION WORK REQUEST | STATION ONS | UNIT NO. 1 | PRIORITY 5 | 19167 A

I ORIGINATOR

ORIGINATOR BK DATE 12/27/76 APPROVED BKnight DATE 12/27/76 | AVAILABILITY DATE & TIME ____ OR ____ HRS. NOTICE | COMPLETE WORK BY DATE / / TIME ____

EQUIP. NO. / NAME 'B' FDW HTR SHELL RELIEF | BLD. TB | ELEV. 3 | COL. LOCATION K20 & L20 | ID TAG PLACED YES ☒ NO ☐

DESCRIPTION OF WORK REQUESTED: UNIT #1 ACT. CODE 2127

OUTAGE CREW #

Please repair seat leaks on both B heater shell relief valves

1B1 - HV-8 1RV-3 4XPX6 OM-254-14

1B2 - HV-43 1RV-4 4XPX6 " " "

Model D-30P

II PREREQUISITES

IF REQ'D SEE APPLICABLE SECTIONS	REQ'D YES	REQ'D NO	DETERMINED BY	
RED TAG(S)		X	BKnight	NO. ____ THRU ____ CLEARED ☐
RADIATION WORK PERMIT		X	"	
SAFETY RELATED		X	Don Lessing	
IX Q. C. REQUIRED				☐ Prior to job start ☐ during or after job
VIII Functional Verification				

PROCEDURE NO. ____ ACCT. NO. 23112

III SCHEDULING

JOB SEQUENCE DESCRIPTION	CRAFT	EST. MEN X HRS	DATE PERFORMED	CRAFT	ACTUAL M/H
Install scaffold	ME	2 x 4			
Repair valves	ME	2 x 8	9-25-77	2ME	20
Remove scaffold	ME	2 x 2			
	W	2.22			
				TOTAL:	40

IV MATERIALS

PARTS AND MATERIALS

ITEM DESCRIPTION	SEQUENCE NO.	PLANNED QTY.	STATUS	TAGGED	LOCATION	QTY. ISSUED	QA TAG NO.
~~GASKET~~	~~202-061333~~				~~1-GG-7-3~~	~~2EA~~	~~GRS~~
STEM 4" XPX6"	202-061253				1-GG-7-3	1 EA	GRS
INTERNAL PARTS FOR BOTH VALVES							
GASKETS -		6			1GG7-3		
FLAT BAR 1"X2"	N/A	N/A			8-24-77		

PLANNED ☐ SCHEDULED ☐ CODE ☐☐ REQ. NO. ____ EST. DELIVERY ____

PLANNER Don Lessing DATE 12/30/76 MATERIALS CONTACT: ____

V SAFETY PRECAUTIONS

SAFETY PRECAUTIONS:

WELDING/BURNING PERMIT YES ☐ NO ☐ CONFINED SPACE ENTRY PERMIT YES ☐ NO ☐

QUENCH TANK CAVITY ENTRY PERMIT YES ☐ NO ☐

OTHER SAFETY PRECAUTIONS:

VI CLEARANCE

CLEARANCE TO BEGIN WORK ON OPER. EQUIP. OPER. REP SIGNATURE [illegible] OR OPER. REP. CONTACTED ____ CONTACTED BY ____

R-17 DATE 8/23/77 TIME 2300

Figure B-1. Oconee Work Request, Side 1

MAINTENANCE VIII

ACTION TAKEN: TEST EQUIP. SER. NOS.:

Changed seat in 1RV 3 and Hydrowel

Lapped seat in 1RV4 and Hydrowel

PERFORMED BY: [signature] DATE: 8/25/77 TIME: 800

FUNCTIONAL VERIFICATION VIII

FUNCTIONAL VERIFICATION:

METHOD:

RESULTS:

SATISFACTORY PERFORMED BY: DATE: / / TIME:

Q. C. IX

FROM SECTION II (over) REQ'D YES ☐ NO ☐	INSPECTOR NOTIFIED	NOTIFIED BY	DATE	TIME	INSPECTED BY
PRIOR TO JOB START					
DURING OR AFTER JOB					

REMARKS:

TROUBLE SHOOTING X

LIST INSTRUMENT VALVES TO BE OPERATED, ELECTRICAL JUMPERS TO BE INSTALLED AND/OR ELECTRICAL CIRCUITS OPENED: ATTACH ADDITIONAL SHEETS IF REQUIRED.

ITEM	CHANGED FROM NORMAL			RETURNED TO NORMAL		
NAME / NUMBER	DATE	TIME	BY	DATE	TIME	BY

XI

PM SCH. UPDATE REQ'D YES ☐ NO ☐ PM UPDATED

[signature] 8 25 77
SUPERVISOR'S APPROVAL DATE

ACCEPTED BY: [signature] DATE: 8/25/77

Figure B-2. Oconee Work Request, Side 2

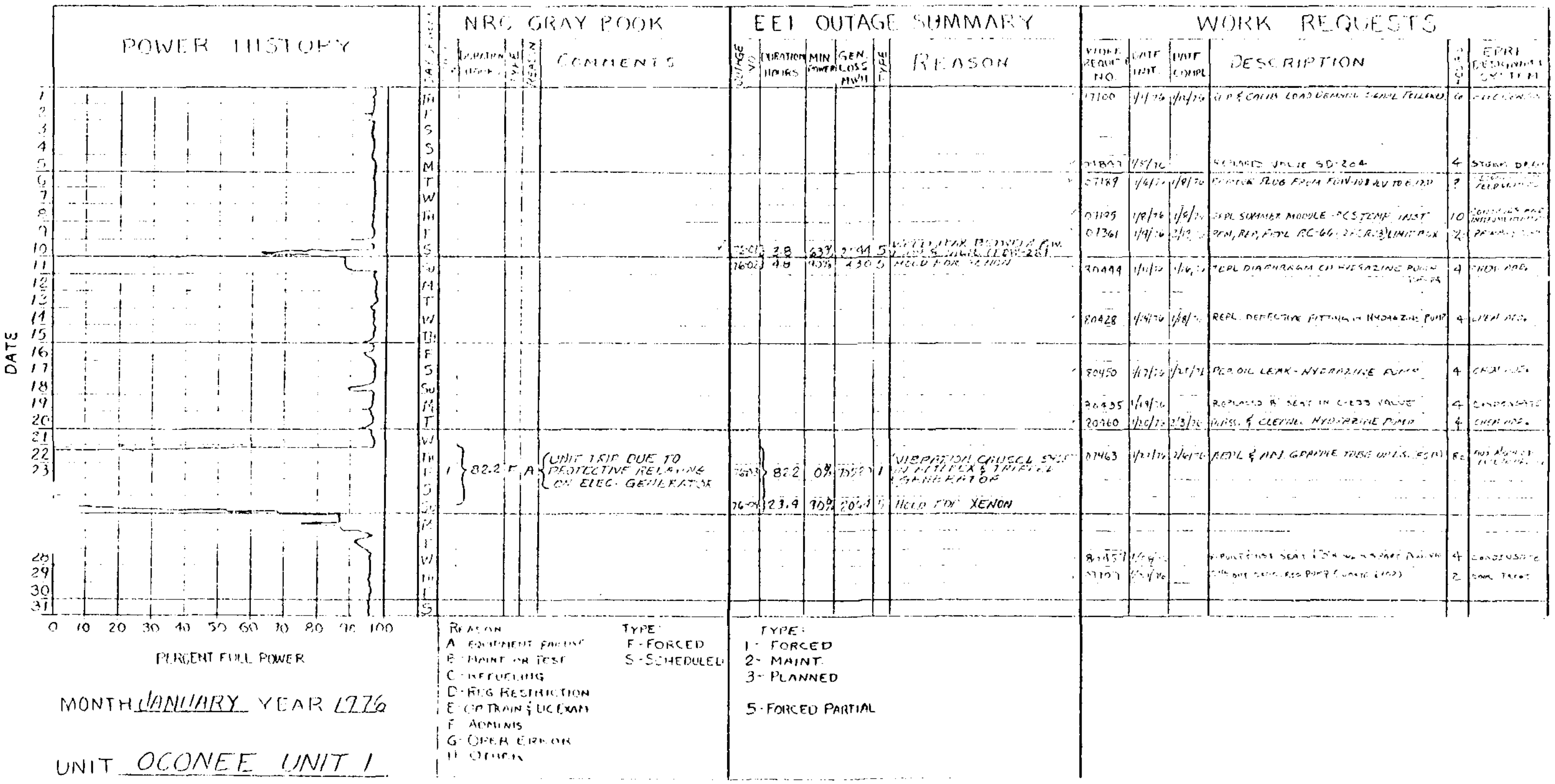


Figure B-3. Power Histogram Comparison With Three Data Sources

APPENDIX C

Data Analysis Methodology

CONTENTS

1. Introduction

The data analysis was directed toward identifying items where design improvements could reduce the availability-limiting factors and toward identifying projects that could become the subject of future research and development. It was neither the intent of this study to encroach on the prerogatives of utility management nor to encroach on the duty of the USNRC to regulate nuclear plants to ensure public health and safety. In the first step of the analysis all data were categorized (defined in Appendix D as cause category) by identifying whether the outage or power reduction was due to operating practices, regulatory requirements, or equipment. Only limiting factors that were categorized as "equipment" were selected for further study and analysis. The equipment category was defined as broadly as possible to reduce the impact of the other two. For example, if the refueling outage was extended to include equipment inspection, repair, or modification because of a regulatory requirement, the additional outage time was designated as "equipment," rather than "regulatory" if it was also concluded that the outage time could probably be reduced by making equipment design changes.

In accordance with contract requirements, the second step in the analysis was to calculate a "priority" or "importance" factor for each limiting factor categorized as being equipment-related. This priority assignment was limited to only those items that directly or indirectly affect plant availability considerations. We concluded that high maintenance items and work events involving high radiation could indirectly affect plant availability; thus, they were factored into the analysis. This phase of the analysis did not consider cost-benefit factors, problems in implementing design changes (plant outages, etc.), and other factors that must be considered before a design improvement is implemented.

To facilitate data collection and analysis, data were grouped and analyzed by systems. The systems were considered to be within one of the following major groups:

- Reactor coolant systems.
- Auxiliary fluid systems.
- Secondary systems.
- Auxiliary mechanical equipment systems.
- Electrical systems.
- Control and instrumentation systems.
- Waste handling systems.
- Other systems.

As an example, the reactor coolant systems group consists of the following:

- Reactor and internals.
- Fuel assemblies and control components.
- Reactor coolant pumps.
- Reactor coolant pump motors.
- Reactor coolant piping.
- Steam generators.
- Pressurizer.
- Core physics.

The core physics system is not a system in the same sense as the others because it does not consist of physical components. However, since this activity clearly impacts plant availability, it was convenient to group and treat this activity as we did other systems.

Within each of these system categories, individual components are listed and (where practicable) identified by individual component name and mark number. The method used to identify systems and components was based on previously established system and component designations and names by B&W and Duke, but for convenience in analysis, the identification contains some degree of arbitrariness. For example, in some systems that contain small pumps, the pump/motor combination may be designated as a component within that system. On the other hand, in the reactor coolant (RC) systems, the RC pumps and motors are, individually, major items of equipment, and each is designated a system.

All systems used in this analysis are listed in Appendix D (Table D-1). The approach used in the analysis was, as described below, to calculate a limiting factor for operation and a limiting factor for maintenance for each of the systems and each key component within the system. A similar approach was used for the refueling work activities.

2. Data Analysis

2.1. Limiting Factor for Operation

The limiting factor for operation (LFO) for Oconee 1 for the period from July 1, 1974, thorugh December 31, 1977 (historical data), is determined from the formula

$$\mathrm{LFO} = \begin{matrix}\text{No. of}\\ \text{events}\end{matrix} \left\{ \begin{matrix}\text{power loss}\\ \text{factor}\end{matrix} \times \mathrm{MTTR} + \begin{matrix}\text{additional}\\ \text{outage time}\end{matrix} \right\} \frac{1}{3.5}$$

(LFO is measured in units of EFPH/unit-year.)

where

No. of events = total number of work events relating to one component as given in the work events tables in Appendix E;

power loss factor = a multiplier to account for power generation capacity lost if the work event caused a plant shutdown or power reduction (if the plant was shut down, the power factor number takes its maximum value of 1.00; if a work event required power reduction to 20% of full power to correct, the power factor number is 0.8 or 100% minus 20%);

MTTR = mean time to repair, which is the average of the clock hours to repair for all events relating to one component, as given in Table E-1 (Appendix E);

additional outage time = time required to bring the plant to a condition that permits work to be done and to return to full power after work is completed (see further explanation below and in Tables C-1 and C-2);

1/3.5 = factor applied to historical data covering the period from 7/1/74 through 12/31/77 to normalize the 3.5 years of data to a yearly basis and thus make the historical data comparable on a one-to-one basis with the 1977 data.

The additional outage time is determined from Oconee Units 1, 2, and 3 data for the year 1977. Table C-1 lists the additional outage times for representative activities that occur during reactor shutdown (power reduction, cooldown, RCS drain, heatup, etc.). Steam generator tube plugging work, for example, would require summing all of the additional outage times listed on Table C-1, including times for cooldown, drain, RCS fill, heatup, etc., which is found to equal 131 hours. As another example, if a work event could be performed at hot shutdown, then the times for cooldown, RCS drain, preparation for startup, RCS fill, and heatup would not be included, and the additional outage time would be 27 instead of 131 hours. Since the individual additional outage times are added to each other and the sum added to the mean time to repair (corrected for power loss), all outage times must represent the same effective hours for losing the same reactor power. Thus, effective full-power hours (EFPH), not total hours, are used for ramp shutdown/startup periods and for partial-power operations (fuel maneuvering limits, xenon hold, etc.). As a part of the additional plant studies, similar additional outage data were obtained at Rancho Seco for all 1976 and 1977 outages longer than two hours. Since these Rancho Seco data give only limited data on many important events such as RCS drain and fill, heatup, etc., they were not used in this analysis and are shown in Table C-2 for information only.

A methodology and analysis similar to that above were applied to the current Oconee 1, 2, and 3 data to obtain the LFOs given in Table D-1. Since the period for these data was one year, the normalizing factor is 1. The three-unit average LFO is found by summing the limiting factors for the three units and dividing by 3. These three-unit average LFOs are also given in Table D-1.

It should be noted that the formula above and the methodology were applied to each critical path work event irrespective of which event was designated as being the cause of the outage. A complete listing of the LFOs that were calculated for Oconee 1 from historical data is given in Table 4-1.

2.2. Limiting Factor for Maintenance

The limiting factor for maintenance (LFM) for the period July 1, 1974, through December 31, 1977 (historical data), is determined from the formula

$$\mathrm{LFM} = \begin{matrix}\text{No. of}\\ \text{events}\end{matrix} \left(\begin{matrix}\text{No. of}\\ \text{men}\end{matrix} \times \mathrm{MTTR}\right) \frac{1}{3.5} \qquad \text{(manhours/unit-year)}$$

where the number of events, MTTR, and 1/3.5 normalizing factor are the same as described above for the LFO. Again, the values for these factors are given in Table E-1.

2.3. Limiting Factor for Refueling

Limiting factors for refueling (LFR) were determined from the formula

$$\mathrm{LFR} = (P - S)Fp \qquad \text{(EFPH/unit-year)}$$

where

P = average refueling loss, EFPH (hours),

S = B&W-projected standard hours for performing that work,

Fp = power loss factor (100 - % power).

The LFRs calculated in this study and the input parameters are given in Table 4-4. These input parameters are described further in the following paragraphs.

The "actual performance time" (P) is the average performance time of activities on or close to the critical path events based on performance at Oconee 1 and 3, Rancho Seco, and TMI-1. These times are not totals in that they do not include times for work or work activities that do not approach the critical path.

The "standard projected performance time" (S) is an estimated time to complete the individual task. The projected time was derived by studies of B&W service department estimates, utility estimates, a review of past performance times tempered by the team's judgment based on observation of the activities being performed using existing equipment at operating plants. Times for most of the activities have been bettered in the field for individual tasks, but the overall refueling activities have not yet matched the overall projected schedule. The projected schedule for fuel movement operations allows seven days for this task. Recent refueling operations at Arkansas One completed the task in five days. The project schedule allows 16 hours for tensioning the reactor head bolts; the Arkansas One team performed this task in 5 hours. Past estimates to remove the insulation from the reactor head allow 4 hours; the Rancho Seco refueling team removed this insulation in 1.5 hours. The projected times are achievable, but it requires good coordination and equipment to be in good working condition and available when requred.

The power factor (Fp) emphasizes the fact that the plant is idle and unproductive during refueling. For most of the refueling activities, this factor is 100% except for the plant startup and physics test activities, when a 50% loss of power is assumed.

As data were accumulated from the Oconee and Rancho Seco plants, the information was transferred in bar form onto the refueling activities chart (Appendix F). This chart depicts critical and/or near-critical path times based on daily plots. Main events and delay identifications are listed in the daily events columns. An activity that was identified as a critical path item is shown as a wide-hatched bar, while a critical path delay is shown as a wide plain bar. Non-critical path items are depicted as narrow bars.

Refueling shutdown performance times at Oconee 1, Rancho Seco, Oconee 2, and TMI-1 were compared for selected critical path/near-critical path work items, and a combined refueling limiting factor for each work item was obtained as shown in Table 4-4. These additional plant comparisons were made to test the conclusions regarding limiting factors for the Oconee 1 plant and to assess the applicability of the limiting factors to generic productivity and availability limits. A discussion of the results and recommendations for improving availability are given in section 4.3.

2.4. Combined Equipment Limiting Factor

The CELF is the loss in plant availability in EFPH per unit-year for a given system/component. It is determined from the formula

$$\text{CELF} = \frac{\text{LFO}_H + \text{LFO}_C}{2} + \begin{matrix}\text{Refueling}\\ \text{outage}\\ \text{extension}\end{matrix} \qquad \text{(EFPH/unit-year)}$$

where

LFO_H = LFO, Oconee 1 historical data from Table 4-1,

LFO_C = LFO, Oconee 1, 2, and 3 current data from Tables 4-3 and D-1,

Refueling outage extension = portion of the LFR (from Tabl3 4-4) that caused a refueling outage extension due to equipment problems.

Each historical/current LFO is on a one-reactor-unit-averaged basis. Since the data bases for both contain approximately the same number of reactor unit-years, an average of these two is appropriate for calculational purposes. The refueling outage extension values refer to maintenance on steam generators, fuel handling equipment, incore monitors, and the polar crane that extended the refueling outage period.

Refer to Table 3-1 for a complete listing of the CELFs. Tables 3-2 and 3-4 identify design category CEOFs by outage identification and resolution status, respectively.

2.5. Plant Availability Vs Summation of CELFs

Plant availability refers to the amount of time the plant is available for power production. Any activity that reduces the availability, in effect, reduces the power generation capability of the unit.

The CELF, which is measured in equivalent full-power hours, indicates the reduction in plant availability as a result of the stated event. However, each CELF includes applicable startups, shutdowns, and component access time for the evaluated item which results in a total equipment outage time that is greater than the plant outage time. Thus, any summation of CELFs should be considered a conservative reduction in plant availability.

Table C-1. Additional Outage Times — Oconee 1, 2, and 3 (EFPH)

Shutdown period (1977), start to end	Power red'n (100 to 15%), EFPH	Cooldown: A. 532 to 150F, B. 532 to 280F (a)	RCS drain: A. complete, B. to 185" pzr level (b)	Prep'n for startup (c)	RCS fill: A. complete, B. from 185" prz level (a)	Heatup (e)	Startup (f)	Power escalation (0-100%), EFPH	Fuel maneuv'g limits, EFPH	Xenon hold, EFPH	Total
Unit 1											
1/15 to 1/27	2.0	A 22	A. 17	12	A. 6	42	12	8	1.4	2	--
1/31 to 2/10	2.5	A 21	B. 13	13	B. 9	21	9	16	0	2.3	--
2/28 to 3/15	2.0	A 31	A. 24	13	A. 25	34	11	14	1.6	1.8	--
3/22 to 4/5	2.5	A 28	A. 23	NA	B. 8	32	15	5.5	1.5	2.1	--
4/23 to 4/28					Not representative						
5/7 to 6/9	1	A 32	A. 24	18	A. 9	38	14	12	3.3	1.7	--
7/5 to 7/31	4.5	NA	NA	NA	NA	NA	NA	10	0	0	--
8/5 to 11/24					Refueling and startup physics						
12/9 to 12/12	2.5	NA	NA	NA	NA	NA	NA	15	0	2.1	--
12/20 to 12/27	2.5	A 24	NA	NA	NA	40	11	15	0	0	--
12/30 to 12/31	1.5	NA	NA	NA	NA	NA	NA	6	0	2	--
Unit 2											
3/23 to 3/27	NA	NA	NA	NA	NA	NA	4	11	1	0	--
5/28 to 9/6					Refueling and startup physics						
9/11 to 9/26	1.0	NAV	NAV	NAV	NAV	24	11	12	0	1.9	--
10/7 to 11/3	2.5	NAV	NAV	NAV	NAV	NAV	NAV	9	1.3	1.5	--
11/3 to 12/28	4.5	NAV	NAV	NAV	NAV	NAV	NAV	13	3.5	0	--
								9	1.8	2.0	--
Unit 3											
2/14 to 2/28	2	A 30	A. 18	15	A. 22	13	7	9	1.8	2.0	--
4/6 to 4/8	NA	NA	NA	NA	NA	NA	5	6	0	2.2	--
4/13 to 4/14	NA	NA	NA	NA	NA	NA	5	6.5	0	2.4	--
6/10 to 6/28	1	A 24	A. 6	NA	A. 7	24	28	7.5	1.5	2.4	--
7/14 to 8/2	2.5	A 33	A. 17	17	A. 16	23	30	7.5	1.5	2.4	--
8/20 to 8/25	2.0	NA	NA	NA	NA	NA	NA	6.5	0	1.7	--
9/2 to 9/5	5.0	NA	NA	NA	NA	NA	NA	8	0	1.3	--
10/13 to 10/21					Not representative						
10/21 to 12/20					Refueling and startup physics						
Avg Full drain	2.4 ± 0.3	A 27.2 ± 1.5	A. 18.2 ± 2.4	14.7 ± 1.0	A. 13.3 ± 2.9	30.2 ± 2.6	12.5 ± 2.3	9.9 ± 0.8	0.9 ± 0.2	1.6 ± 0.2	= 131.1
Avg Drain to 185"	2.4 ± 0.3	A 27.2 ± 1.5	B. 13.0 ± %	14.7 ± 1.0	B. 9.0 ± %	30.2 ± 2.6	12.5 ± 2.3	9.9 ± 0.8	0.9 ± 0.2	1.6 ± 0.2	= 121.4
Avg Hot shutdown	2.4 ± 0.3	--	--	--	--	--	12.5 ± 2.3	9.9 ± 0.8	0.9 ± 0.2	1.6 ± 0.2	= 27.3
Avg Full cooldown	2.4 ± 0.3	A 27.2 ± 1.5	--	--	--	30.2 ± 2.6	12.5 ± 2.3	9.9 ± 0.8	0.9 ± 0.2	1.6 ± 0.2	= 84.7
Avg Cooldown to 280F	2.4 ± 0.3	B 7.5	--	--	--	8.3	12.5 ± 2.3	9.9 ± 0.8	0.9 ± 0.2	1.6 ± 0.2	= 43.1

(a) Cooldown B used for stator replacement; use 7.5 hours.

(b) Required drain for CRDM repair.

(c) Precritical and prestart checks complete.

(d) NA: not applicable, NAV: not available.

(e) From RCS fill (including establishing RCS chemistry, deboration) to hot shutdown condition — 532F, 2155 psig.

(f) From hot shutdown condition to unit critical, 10^{-8} amps.

(g) Estimated from Duke Power Company records.

Table C-2. Additional Outage Times -- Rancho Seco (EFPH)

Shutdown period, start-end	Power red'n (100 to 15%), EFPH	Cooldown: A. 532 to 150F, B. 532 to 300F	RCS drain	Prep'n for startup[a]	RCS fill	Heatup to 532F, 2155 psig	Startup[b]	Power escalation (0-100%), EFPH	Fuel maneuvering limits, EFPH	Xenon hold, EFPH	Total
3/1/76-3/4/76	2.5	NA	NA	NA	NA	NA	2	14	0	2.9	--
10/10/76-10/10/76	Not representative										
11/5/76-11/14/76	1.0	A. 22.5	NAV	NAV	16.5	31.5	9	11.5	1.5	2.1	--
11/21/76-11/23/76	2.0	NA	NA	NA	NA	NA	NA	12	0	0.3	--
12/8/76-12/9/76	1.5	NA	NA	NA	NA	NA	NA	3.5	0	0.2	--
1/13/77-1/14/77	1.0	NA	NA	NA	NA	NA	NA	5	0.5	0.2	--
2/25/77-2/26/77	1.0	NA	NA	NA	NA	NA	NA	4	0	0.4	--
4/21/77-4/23/77	3.5	B. 4.5	NA	NA	NA	7[c]	4.5	3.5	0	0.5	--
5/21/77-5/23/77	1.3	NA	NA	NA	NA	NA	2	4.5	0.3	1.0	--
7/29/77-7/30/77	0.7	NA	NA	NA	NA	NA	0.7	7.3	0	0.2	--
11/17/77-11/18/77	1.0	NA	NA	NA	NA	NA	0.5	3.0	0	0.2	--
Avg full cooldown	1.6 ± 0.3	A. 22.5	--	--	--	31.5	3.1 ± 1.3	6.8 ± 1.3	0.2 ± 0.15	0.8 ± 0.3	= 66.5
Avg cooldown to 300F	1.6 ± 0.3	B. 4.5	--	--	--	7[c]	3.1 ± 1.3	6.8 ± 1.3	0.2 ± 0.15	0.8 ± 0.3	= 24.0

[a] ECP calculations complete; precritical and prestart checks complete.

[b] From hot shutdown to unit critical, 10^{-8} amps.

[c] From 300 to 532F (stator replacement).

[d] NA: not applicable, NAV: not available.

APPENDIX D

1977 Oconee Power History/
Work Activity Record

CONTENTS

1. Introduction

Current data, as defined in Appendix B, are the non-refueling data for key items on Oconee Units 1, 2, and 3 for the year 1977. These data are described below and shown on the power history/work activities histograms, Figures D-1 through D-3. Data given on these histograms are summarized in Figures D-4, D-5, and D-6, the "1977 Operating Records for Oconee 1, 2, and 3," to show on a more comprehensive scale the power history and the major causes of lost plant operating time for each of the Oconee units for 1977. Using the current data and the methodology described in Appendix c (2.1), equipment-related LFOs for Oconee 1, 2, and 3 were calcualted and are shown in current data summary Table D-1 and discussed in section 4.

2. Oconee 1, 2, and 3 Power History/Work Activities Histograms

The power history/work activities histograms shown in Figures D-1 through D-3 were prepared from data secured from the following sources:

- Work requests.
- Duke reports to EEI.
- Operators'/shift supervisors' logs.
- Discussions with Duke operations and maintenance personnel.
- B&W notes compiled from log books, personal observation, discussions, and power history charts supplied to B&W by Duke.

Key item work performed during outages is shown as a bar graph below the power historgram on the power history/work activities figures. The key item work events given here are those that caused the shutdown/power reduction, were critical path for that outage, or influenced the duration of that outage. Additional definition of "key items" is given in Appendix B, part 2. Further explanation of the power history/work activities (Figures D-1 through D-3) follows.

Key item work performed during an outage is shown as "additional key items" on the power history/work activities figures, if the work activity could cause a shutdown or power reduction at a later time or delay plant startup. Hydraulic suppressors and some valves are typical of the components listed as "additional key items" because they could and have caused delays in plant startup during an outage and have had repetitive maintenance.

The "events/delays" box on the power history/work activities figures contains definitions of the abbreviations used on the bar graphs. Capital letters designate the plant operational status. Lower-case letters designate the nature of a work event, and numerals refer to causes of delays. The capital letter designations are the same on all figures. The lower-case letter and number designations refer only to items on that figure.

The "work category" box states the action taken to correct an equipment deficiency or operating problem:

- Repair/Correction (R/C) — Work performed as a result of equipment failure or degradation below acceptable operating limits.
- Inspection, Testing, Calibration (ITC) — Work supporting equipment inspection, testing, or calibration during scheduled or unscheduled shutdowns.
- Nuclear Station Modification (NSM) — Work directed to equipment changes as a result of changes in requirements, equipment management, equipment addition, equipment removal, etc.
- Operational Maintenance (OM) — Work performed to restore equipment to acceptable operational performance standards (there are no references to OM on the power history/work activities figures).
- Preventive Maintenance (PM) — Action taken to replace, adjust, or refurbish equipment or equipment components on a scheduled basis to provide a higher degree of assurance that the equipment will operate without failure for the required operating time.

Applicable category abbreviations are shown in parentheses at the end of the bar graph activities descriptions.

The "cause category" box indicates the cross-hatch pattern on the bar graphs for each of four cause categories. Cause categories sort the work events identified with respect to the cause or reason for the work being performed. This category also identifies items that could be the subject of further analysis.

- Operating practice/Requirements — Factors that directly influence plant productivity through utility policy and the plant's role in the overall electrical grid requirements and design requirements, and indirectly through the plant's operating and maintenance philosophy. Operating requirements to comply with Technical Specifications were included but may not necessarily reflect policy.

  Shortages of qualified maintenance personnel were included in this category, but the lack of space in which to use those personnel was in the equipment cause category. Operator errors were included in this category, but operator "traps" (i.e., items where the plant configuration caused the operator to err) were included in the equipment cause category.

- Regulatory Requirements — Regulatory requirements comprise the factors that can be clearly defined and quantified as being solely due to regulatory requirements and that would not otherwise reduce plant productivity.
- Equipment Deficiencies and Failures (EQ) — All items identified and not included in categories 1 and 2 would be considered equipment. Included in this category are NSS features, balance-of-plant features, plant layout, maintainability, redundancy (or lack of it), preventive maintenance specified by vendors, inadequate water storage capability, insufficient demineralized water supply, etc. Only the items in this category were analyzed further. As noted in paragraph 1 of Appendix C, the equipment category was broadly defined. The example was given of a regulatory requirement requiring an inspection that was categorized as "equipment" because it was concluded that the work could be done more quickly by implementing an equipment design change.
- Delays — Delays in completing work activities on the power history/work activities; figures consist of such items as evacuating the reactor building because of airborne radioactivity, manpower shortages, work breaks, lack of tools, parts, or other equipment, waiting for equipment (e.g., use of polar crane), equipment access, and similar items. Delay time is included with and charged to the work item in progress when the delay(s) occurred, as shown on the power history/work activities figures.

The alphanumeric designations shown before the bar graph activity description identify the system/component affected by the event. The name of the system/component corresponding to the alphanumeric designation is shown in the equipment-related LFOs for Oconee Units 1, 2, and 3 in Table D-1.

The "power history" plot of percent power versus date was obtained from the Duke Power Co. operations log and checked against power history curves also supplied to B&W by Duke. EEI outage numbers listed above the "power history" plot are those numbers assigned by Duke to outages they reported to EEI. These numbers may not be in strict conformance with final numbers assigned to outages because in final reporting to EEI, outages were collected and reported under a Duke computerized reporting system. Using the data shown on the power history/work activities figures and the methods described in Appendix C (paragraph 2.1), the limiting factors for operation were calculated for Oconee 1, 2, and 3. The results for each system and for each unit are shown in Table D-1.

The three-unit average limiting factor, shown on Figure D-1, is obtained by summing the individual limiting factors and dividing by 3. As noted in Appendix C, the formula for calculating the LFO was applied to each critical path work event identified, irrespective of whether or not that event caused the power reduction.

Equipment-related LFOs for Oconee 1, 2, and 3 (1977) are summarized in Table D-2 and repeated as Table 4-3 with the systems ranked by the average limiting factor numbers. The number of events and the number of units affected are also given. This table permits comparison of the current data for the three Oconee units with the historical data summary for Oconee 1 shown in Table 4-1. Discussions of the results of the study and recommendations for improving availability are given in section 4.

		Events that forced or extended power reduction — Unit 1						Events that forced or extended power reduction — Unit 2						Events that forced or extended power reduction — Unit 3						
Report section	System/component	No. of events	Power loss factors	Mean time to repair, hours	Additional loss per event, EFPH	Average loss per event, EFPH	LFO (normalized	No. of events	Power loss factors	Mean time to repair, hours	Additional loss per event, EFPH	Average loss per event, EFPH	LFO (normalized)	No. of events	Power loss factors	Mean time to repair, hours	Additional loss per event, EFPH	Average loss per event, EFPH	LFO (normalized)	Three-unit avg
4.2.1	1 Reactor Coolant System																			
4.2.1.1	1A Reactor & internals	0					0	0					0	0					0	
4.2.1.2	1B Fuel & rods	0					0	0					0	0					0	
4.2.1.3	1C Reactor coolant pumps	2	1	75	65.5	140.5	281	0					0	0					0	93.7
4.2.1.4	1D RC pump motors	0					0	0					0	2	0.63	50.5	131	162.8	325.6	108.5
4.2.1.5	1E Piping	0					0	0					0	0					0	
4.2.1.6	1F Steam generators	5	1	170	131	301	1505	1[a]	0.59	1891	48.6	1164	1164	3	1	98.3	131	229	688	1119
4.2.1.7	1G Pressurizer	1	1	8	84.7	92.7	92.7	1	1	19	84.7	103.7	103.7	0					0	65.5
4.2.1.8	1H Core physics & Rx safety	21					406.8	18					275.4	21					255.9	312.7
	1H1 Fuel maneuvering	6	0.33	10.8	0	3.6	21.6	5	0.46	13.6	0	6.3	31.3	4	0.30	6.5	0	1.95	7.8	20.7
	1H2 Core tilt	2	0.25	204	0	51	102	4	0.15	41.5	--	6.2	24.9	3	0.58	7.3	0	4.3	12.8	46.7
	1H3 Xenon hold	12	0.1	15.1	0	1.51	18.2	8	0.1	11.5	0	1.15	9.2	13	0.1	17.2	0	1.72	22.3	16.6
	1H4 Startup physics tests	1	1	0	265	265	265	1	1	0	210[b]	210	210	1	1	0	213	213	213	229.3
4.2.2	2 Auxiliary Fluid System																			
4.2.2.1	2A Makeup & purif'n/HPI	2	1	37.5	0	37.5	75	1	1	8	84.7	92.7	92.7	1	0.09	95	0	8.35	8.35	58.7
4.2.2.2	2B Decay heat/LPI	0					0	1	1	60	36	96	96	2	1	3	12	15	30	42
4.2.2.3	2C Chem add'n and sampling	0					0	0					0	1	1	18	30	48	48	16
4.2.2.4	2D Spent fuel cooling system	0					0	0					0	0						0
4.2.2.5	2E Rx building spray	0					0	0					0	1	1	1.5	27.3	28.8	28.8	9.6
4.2.2.6	2F Core flooding system	0					0	0					0	0					0	
4.2.2.7	2G Low-pressure serv. water	0					0	0					0	0					0	
4.2.2.8	2I Component cooling system	0					0	0					0	0					0	
4.2.2.9	2J Penetr room vent/RG purge	0					0	0					0	0					0	
4.2.3	3 Secondary System																			
4.2.3.1	3A Main turbine	1	1	1	0	1	1	1	1	4	10	14	14	0					0	5.0
4.2.3.2	3B Main steam	1	0.4	3	6	7.2	7.2	3	0.25	6.7	2	3.67	11	2	0.3	3	2	2.9	5.8	8
4.2.3.3	3C Feedwater	3	0.6	5	5	8	24	3	0.71	54	0	38.3	115	3	1	12.3	7	19.3	58	65.6
4.2.3.4	3D Condensate	1	0.1	105	0	10.5	10.5	4	0.72	24	17	34.3	137	0					0	49.2
4.2.3.5	3E Cond circ. water	0					0	0					0	0					0	
4.2.3.6	3F Recirc cooling water	0					0	0					0	0					0	
4.2.3.7	3G Auxiliary steam	0					0	0					0	0					0	
4.2.3.8	3H Moisture sep reheaters	0					0	0					0	1	0.5	3	0	1.5	1.5	0.5
4.2.3.9	3I Generator stator cooling	1	1	2	27.3	29.3	29.3	1	0.75	3	10.8	13	13	1	0.75	2	10.8	12.3	12.3	18.2
4.2.3.10	3J Heater drains	2	0.6	20	15.7	27.7	55.4	2	0.05	245	0	12.3	24.6	0					0	26.7
4.2.3.11	3K Instrument air	0					0	0					0	1	0.5	2	5	6	6	2.0
4.2.3.12	3L Turbine lube oil	2	1	38	27.3	65.3	130.6	2	1	3	27.3	31.3	62.6	2	0.45	20.5	7.1	16.3	32.6	75.3
4.2.3.13	3M EHC system	0					0	2	0.93	7.5	12.7	19.7	39.4	2	1	4	27.3	31.3	62.6	34.0
4.2.3.14	3N HP service water	0					0	0					0	0					0	
4.2.3.15	3P Nitrogen supply	0					0	0					0	0					0	
4.2.3.16	3Q Steam drains	0					0	0					0	0					0	
4.2.3.17	3R Vacuum system	0					0	0					0	0					0	
4.2.4	4 Auxiliary Mechanical Equipment																			
4.2.4.1	4A Control rod drive system	9					326.9	21					654.9	7					320.2	434
	4A1 Drives	0					0	0					0	0					0	
	4A2 Stators	1	1	8	43.1	51.1	51.1	17	1	13.1	20.3	33.4	568.3	3	1	17	43.1	60.1	180	226.4
	4A3 Position indicators	5	0.9	16.4	5.4	20.16	100.8	0					0	0					0	33.6
	4A4 Power & T/C cables	3	1	25.3	19.9	61.3	135.8	1	1	34	19.9	53.9	53.9	0					0	63.2
	4A5 Closure/vent system	0	1	4	35.2	39.2	39.2	0					0	1	1	3	121.4	124.4	124.4	54.5
	4A6 CRD control system	0					0	3	0.7	3	8.8	10.9	32.7	3	0.75	7	0	5.3	15.8	16.2
4.2.4.2	4B Fuel handling bridges	0					0	0					0	0					0	
4.2.4.3	4C Fuel transfer equipment	0					0	0					0	0					0	
4.2.4.4	4D CRDM serv struc fans/ducts	0					0	0					0	0					0	
4.2.4.5	4E Suppressors & hangers	0					0	0					0	0					0	

[a] Does not include the following:

400 hours downtime to remove mechanic's file starting 7/17/77.

83 hours to conduct special steam generator equipment tests starting 9/4/77 and 9/11/77.

[b] 28 EFPH for fuel maneuvering and xenon equilibrium.

Table D-1. 1977 Equipment-Related Limiting Factor for Operation of Oconee Units 1, 2, and 3

Report section	System/component	Unit 1: No. of events	Unit 1: Power loss factors	Unit 1: Mean time to repair, hours	Unit 1: Additional loss per event, EFPH	Unit 1: Average loss per event, EFPH	Unit 1: LFO (normalized)	Unit 2: No. of events	Unit 2: Power loss factors	Unit 2: Mean time to repair, hours	Unit 2: Additional loss per event, EFPH	Unit 2: Average loss per event, EFPH	Unit 2: LFO (normalized)	Unit 3: No. of events	Unit 3: Power loss factors	Unit 3: Mean time to repair, hours	Unit 3: Additional loss per event, EFPH	Unit 3: Average loss per event, EFPH	Unit 3: LFO (normalized)	Three-unit avg
4.2.5	5 Electrical																			
4.2.5.1	5A Generator	1	0.85	13	12.4	23.4	23.4	0					0	0					0	7.8
4.2.5.2	5B Switchgear	0					0	0					0	0					0	
4.2.5.3	5C Controls	0					0	0					0	0					0	
4.2.5.4	5D Exciter	0					0	0					0	0					0	
4.2.5.5	5E Transformer	0					0	0					0	0					0	
4.2.5.6	5F Substation	0					0	0					0	0					0	
4.2.5.7	5G Isolation phase bus	0					0	0					0	0					0	
4.2.5.8	5H Batteries	0					0	0					0	0					0	
4.2.5.9	5I Chargers	0					0	0					0	0					0	
4.2.6	6 Controls & Instrumentation																			
4.2.6.1	6A Control & monitoring equip	2					23.6	1					5.1	1					2.2	10.3
	6A1 Integr control system	1	0.35	3	5.9	6.9	6.9	1	0.25	2	4.6	5.1	5.1	1	0.3	2	1.6	2.2	2.2	4.7
	6A2 Non-nucl instrument'n	1	0.6	13	8.9	16.7	16.7	0					0	0					0	5.6
	6A3 Incore detectors	0					0	0					0	0					0	
	6A4 Computers	0					0	0					0	0					0	
	6B Plant protection equipment	2					2.4	0					0	1					1.5	1.3
	6B1 NI/RPS	2	0.6	2	0	1.2	2.4	0					0	1	0.5	3	0	1.5	1.5	1.3
	6B2 Safety-related C&I	0					0	0					0	0					0	
	6B3 ESFAS	0					0	0					0	0					0	
4.2.7	7 Waste Handling Systems																			
4.2.7.1	7A Liquid waste disposal	1	0.19	243	0	47.2	47.2	0					0	2	0.17	126.5	0	22	44	30.4
4.2.7.2	7B Gaseous waste disposal	0					0	0					0	0					0	
4.2.7.3	7C Solid waste disposal	0					0	0					0	0					0	
4.2.7.4	7D Coolant storage	1	1	4	27.3	31.3	31.3	0					0	0					0	10.4
4.2.7.5	7E Coolant treatment	0					0	0					0	0					0	
4.2.8	8 Other																			
4.2.8.1	8A Polar crane						0	0					0	0					0	

Table D-1. (Cont'd)

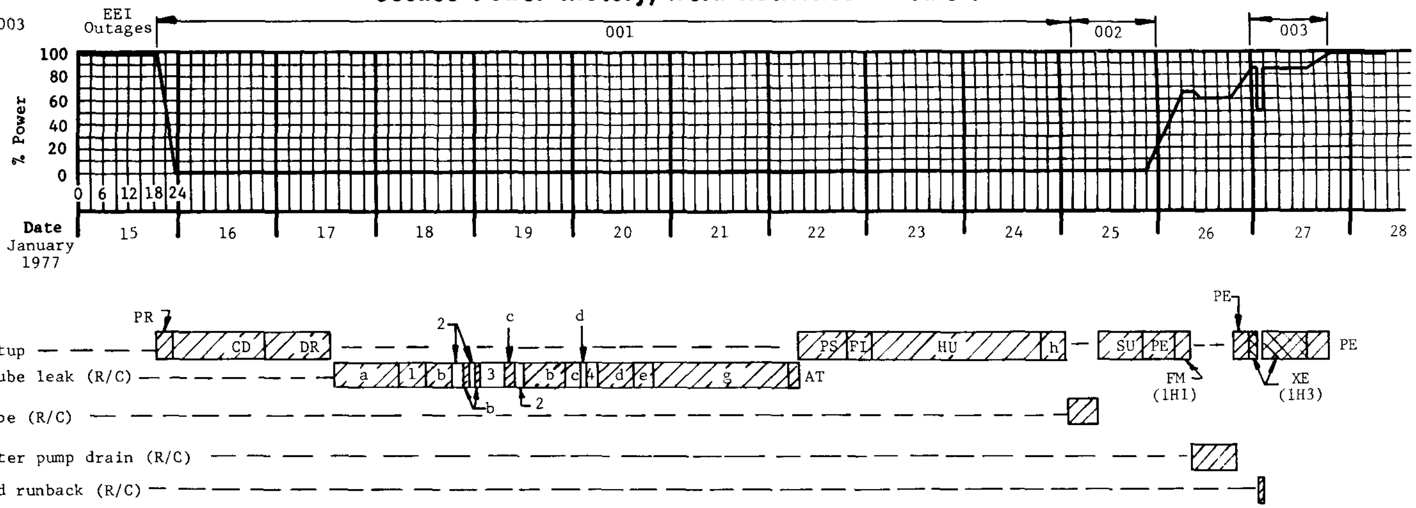
Table D-2. System-Related Limiting Factors — Oconee Units 1, 2, and 3 (1977)

Rank	System/component	No. of events	No. of units affected	Average limiting factor
1	1F Steam generator	9	3	1119
2	4A Control rod drive	22	3	434
3	1H Core physics and RX safety	60	3	313
4	1D RC pump motors	2	1	109
5	1C RC pumps	2	1	94
6	3L Turbine lubricating oil	6	3	75
7	3C Feedwater	9	3	66
8	1G Pressurizer	2	2	65
9	2A Makeup and purification/HPI	4	3	59
10	3D Condensate	5	2	49
11	2B Decay heat/LPI	3	2	42
12	3M Turbine EHC system	4	2	34
13	7A Liquid waste	3	2	30
14	3J Heater drains	4	2	27
15	3I Generator stator cooling	3	3	18
16	2C Chem add'n and sampling	1	1	16
17	7D Coolant storage	1	1	10
18	6A Control and monitoring equip.	4	3	10
19	2E Reactor building spray	1	1	10
20	3B Main steam	6	3	8
21	5A Generator (electrical)	1	1	8
22	3A Main turbine	2	2	5
23	3K Instrument air	1	1	2
24	6B Plant protection equipment	3	2	1
25	3H Moisture separator/reheaters	1	1	0

DATES: 1/5-1/28/77

OUTAGES: 001, 002, 003

Oconee Power History/Work Activities — Unit 1



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	a – Hydro for Tube Leak
1 – RB Evacuated, High Xe Activity (6.5 h)	b – Eddy Current Test Tubes
2 – Worker Break (2.5 h)	c – Fiber Optics Test Tubes
3 – Manpower Shortage (6 h)	d – Plugging Tubes
4 – Lack of Weld Mach (2 h)	e – Leak Test
	g – Repaired Weld on Tube
	h – RCS Leak Test

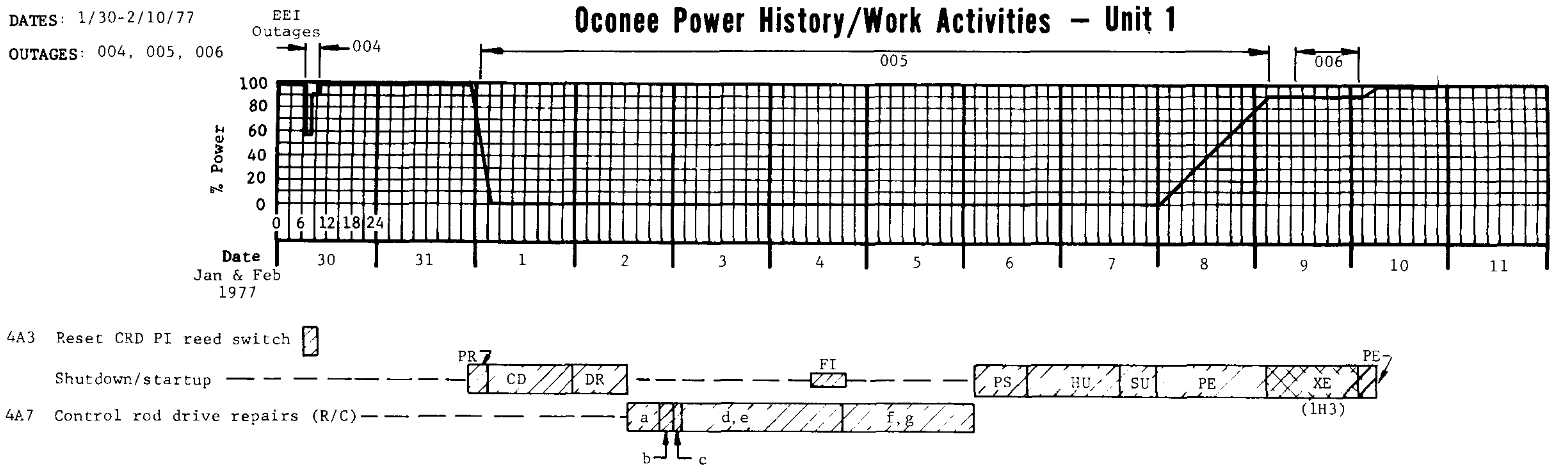
WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM - Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse
36	Insp & repair 14 auxiliary FW nozzles	R/C	EQ
238	Insp safety-related hydraulic suppressors	ITC, /RC	EQ
39	Add shims to generator exciter	R/C	EQ
102	Repair incore inst tube leak	R/C	EQ

Figure D-1. Oconee Power History/Work Activities — Unit 1

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Replace Stator F-12, Noz 32
b – Repair Enclosure assy, Rod 6, Gr 3, Noz 6
c – Replace One PI Tube
d – Clean & Repair 20 PI Tubes (40 h)
e – Replace 10 Power Cables (20 h)
f – Repair 20 PI Tube Cables (30 h)
g – Replace All Thermocouple Cables (32 h)

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

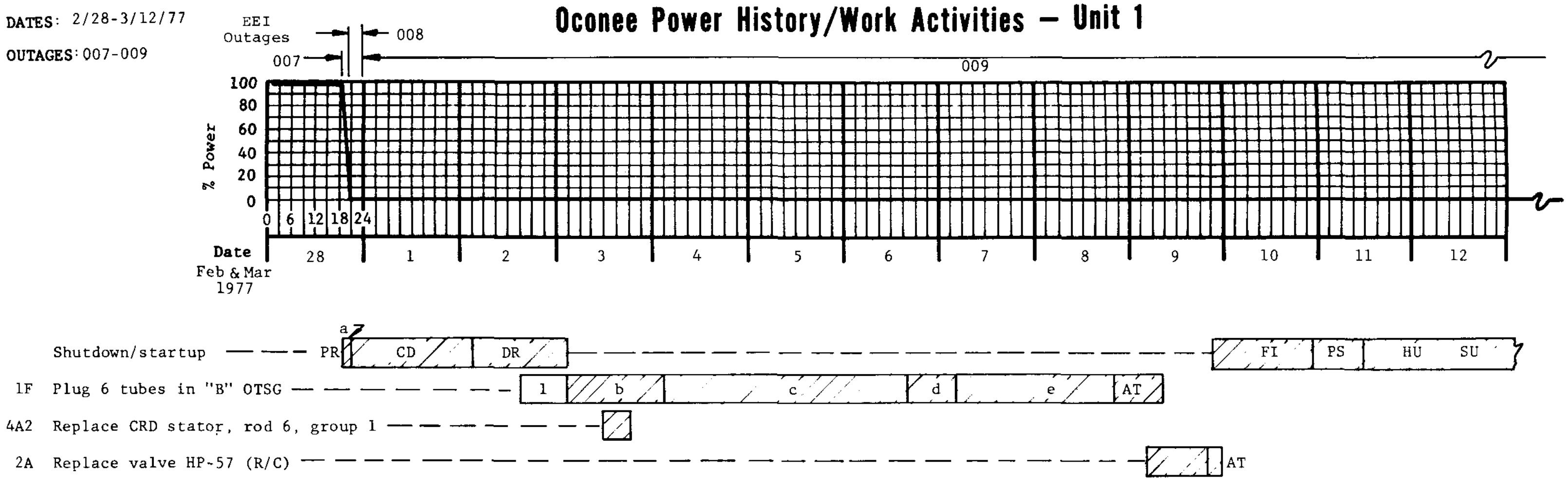
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

Hours	Description	Work	Cse
8	Modify duct to RV heat	NSM	EQ
8	Replace 12 CRD head fans	R/C	EQ
2	Repack RC-22 drain valve	R/C	EQ

Figure D-1. (Cont'd)

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	1 – RB Evacuated, High Xe Activity (12 h)

a – Rod 6, Group 1 Dropped in Core
b – Hydrotest "B" OTSG for Leak Test
c – Eddy-Current Test
d – Cut & Pull One Tube

WORK CATEGORY	
R/C	Repair Correction
ITC	Inspection Testing Calibration
NSM	Nuclear Station Modification
OM	Operational Maintenance
PM	Preventive Maintenance

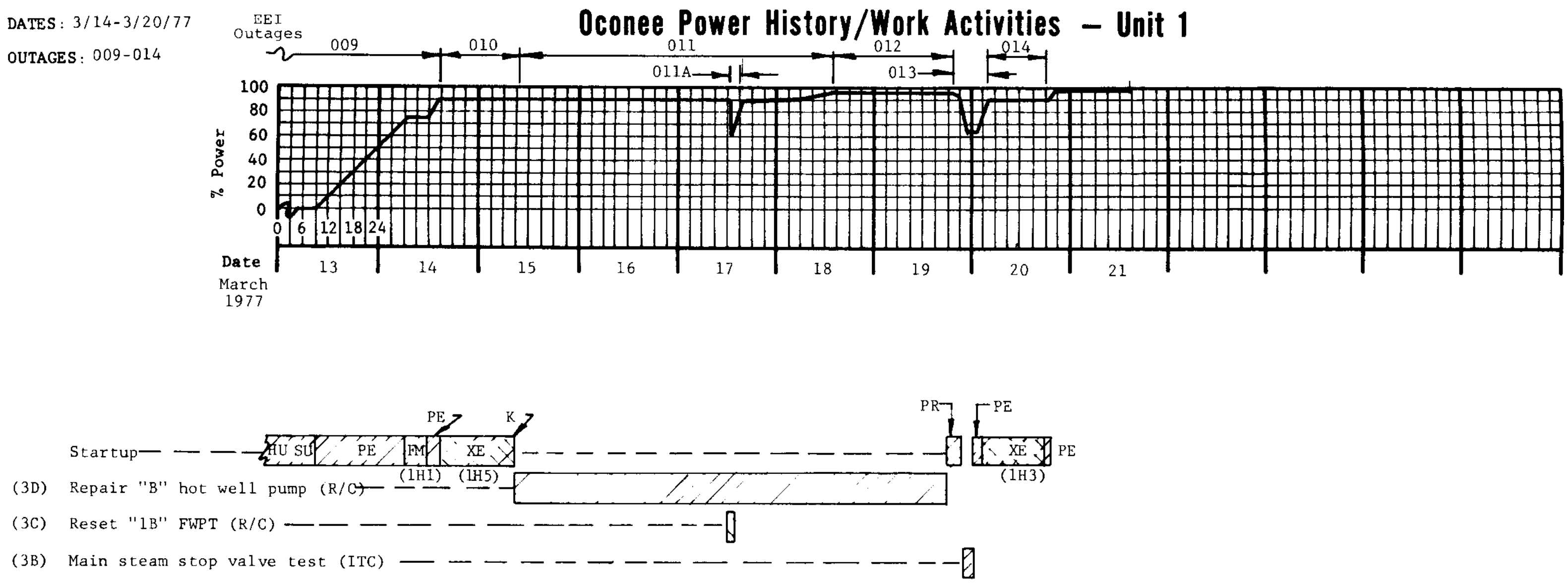
CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

Hours	Description	Work	Cse
36	Insp hot & cold leg pipe hangers	ITC	EQ
16	Replace CRDM closure head gaskets	R/C	EQ
8	Block aux. FDW nozzle on lane	NSM	EQ
6	Repair HP-52 & -53 Foxboro meter	R/C	EQ
6	Repair valve HP-79	R/C	EQ
12	Chang oil in RC pump motors 1A1 & 1B2	PM	EQ
2	Repair leaking drain valve RC-22	R/C	EQ
4	Repair leaking bonnet to HP-153	R/C	EQ
3	Repack valve HD-150	R/C	EQ
3	Repair malf. alarm for "B" stator pump	R/C	EQ
2	Clean seal return filter for RC pumps	PM	EQ
10	Repair valve MS-24	R/C	EQ
2	Repair valve CS-72	R/C	EQ

Figure D-1. (Cont'd)



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	K – Stopped Xe for "B" Hot Well Pump Repair

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

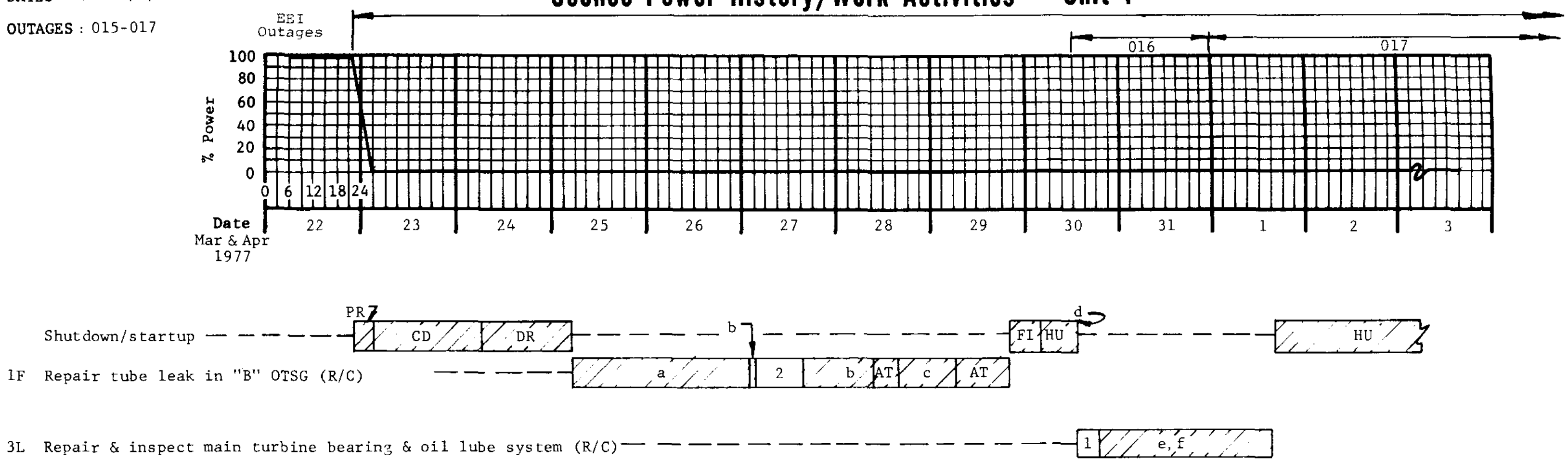
Hours	Description	Work	Cse

Figure D-1. (Cont'd)

DATES : 3/22-4/2/77

OUTAGES : 015-017

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	1 – Delay due to Lack of Proper Jack
a – Hydrotest for Leak	2 – Delay due to Breakdown of Eddy-Current Test Equip
b – EC Test & Weld Repair Prev. Plugged Tubes	
c – Plugged Tubes	
d – Broke Vacuum for Repair of Turning Gear Oil Pump	
e – Inspect No. 4 Bearing (24 h)	
f – Repair Turning Gear Oil Pump (50 h)	

WORK CATEGORY	
R/C	Repair Correction
ITC	Inspection Testing Calibration
NSM	Nuclear Station Modification
OM	Operational Maintenance
PM	Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

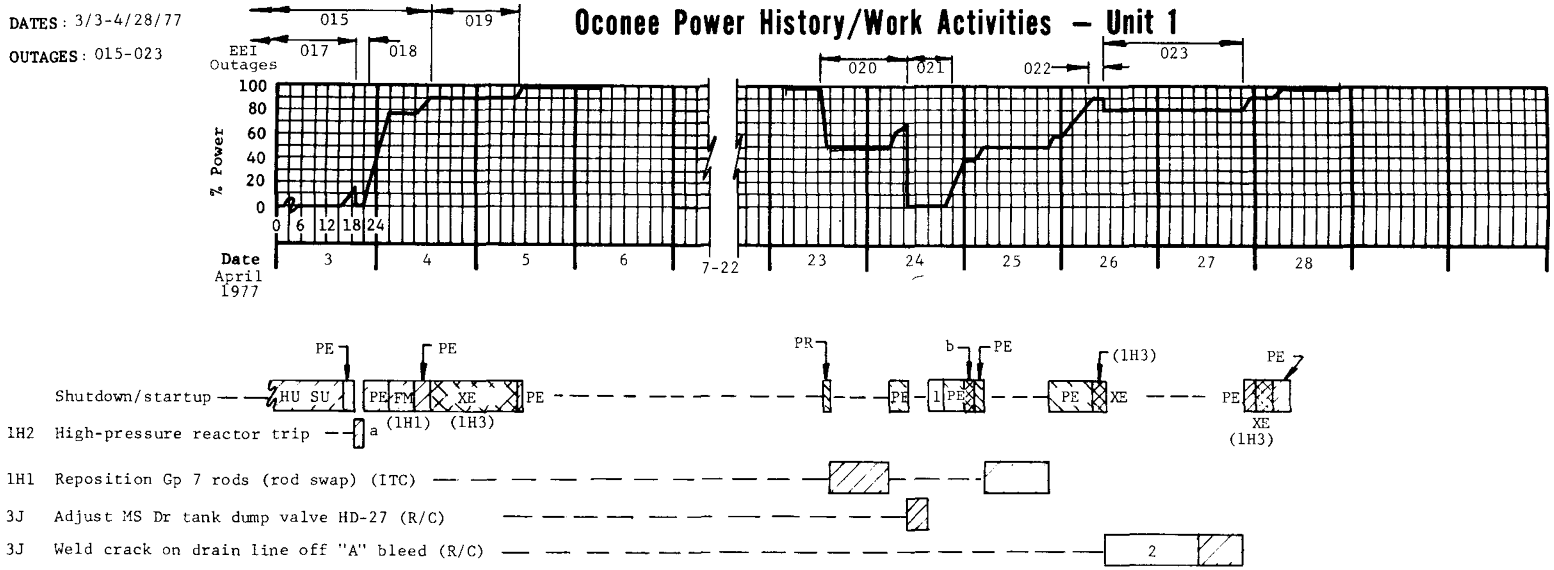
Hours	Description	Work	Cse
4	Setpoint test on MS relief valves	ITC	EQ
64	Preventive maint., RB hoist & crane	PM	EQ
6	Repair valve FDW-247	R/C	EQ
4	Alter service structure ductwork	NSM	EQ
8	Unclog RC bleed sample line	R/C	EQ
4	Replace CRDM "A" PI tube switch	R/C	EQ
8	Retorque A & B auxiliary flow nozzles	R/C	EQ
102	Change out 34 hyd suppressors in RB	R/C	EQ
2	Tram mark hooks	PM	EQ
4	Inspect NI channel 3 cable for noise	PM	EQ
4	Inspect "white rabbit" hook	ITC	EQ
3	Inspect tram hook marks (4th floor)	PM	EQ
4	Remove auxiliary FDW nozzle (OTSG)	NSM	EQ
24	Repair expansion joint leak	R/C	EQ
6	Repair reach rod for valve HP-60	R/C	EQ
4	Repair LDST valve CS-72	R/C	EQ
4	Repair valve LWD-22	R/C	EQ
4	Repair valve CS-85	R/C	EQ
36	Pipe hanger readings in RB	ITC	
162	Performed instrument calibrations	ITC	

Figure D-1. (Cont'd)

DATES: 3/3-4/28/77

OUTAGES: 015-023

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Rx Trip Due to Bad Summer Module in ICS, FDW Valve on "B" Loop
b – Xenon Profile to Start "D" Heater Drain Pumps

1 – Delay – Estimated Critical Position Not Met.
2 – Delay Due to Trying to Isolate Leak; Valves Would Not Close.

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM – Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

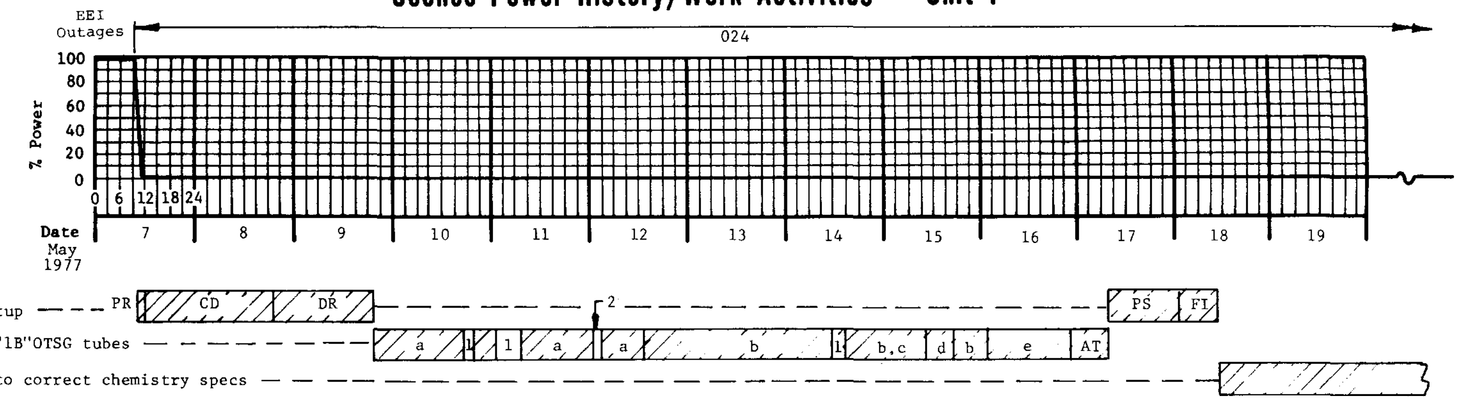
Hours	Description	Work	Cse

Figure D-1. (Cont'd)

DATES: 5/7-5/19/77

OUTAGES: 024

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup

SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

1 – Delays Due to High Xenon Activity in RB
2 – Modified EC Test Equipment

a – Hydrotest OTSG for Tube Leak
b – EC Test
c – Start Fiber Optics Insp on Primary Side
d – Remove Section of Tube 77-18
e – Plug Tubes

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM – Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

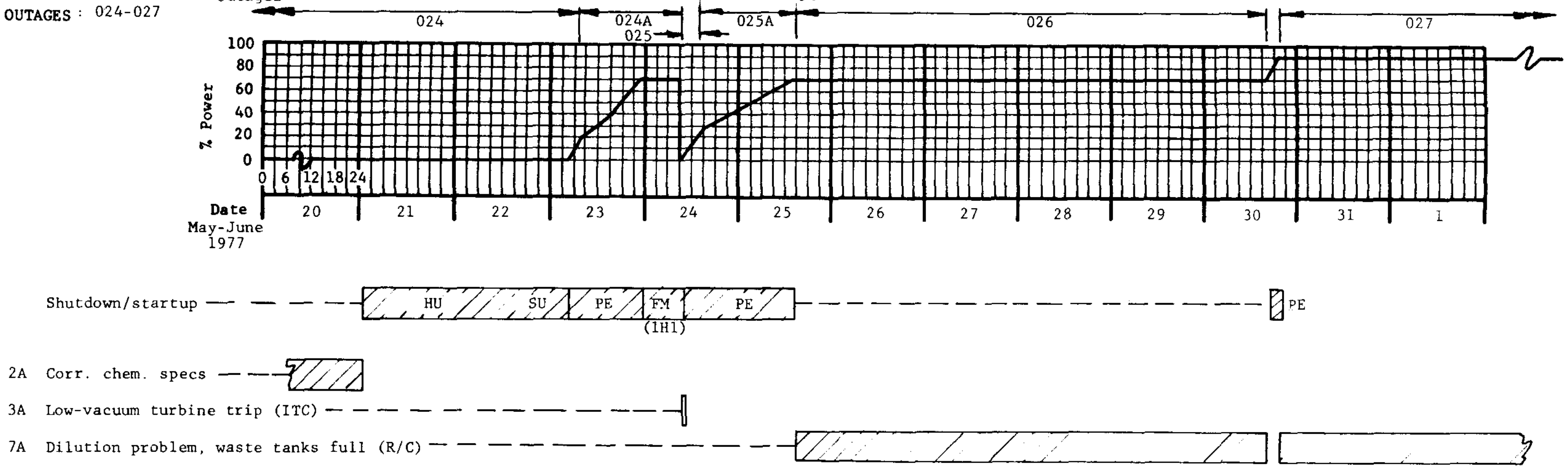
Hours	Description	Work	Cse
105	Changed out 13 hyradulic suppressors.	R/C	EQ
36	Readings of pipe hangers in RB	ITC	EQ
3	Repair oil leak on transformer cooler	R/C	EQ
14	Repack valves HP-98, HP-107, HP-118, HP-249, and MS-88	R/C	EQ

Figure D-1. (Cont'd)

DATES : 5/20-6/1/77

OUTAGES : 024-027

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

WORK CATEGORY
R/C - Repair Correction
ITC - Inspection Testing Calibration
NSM - Nuclear Station Modification
OM - Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

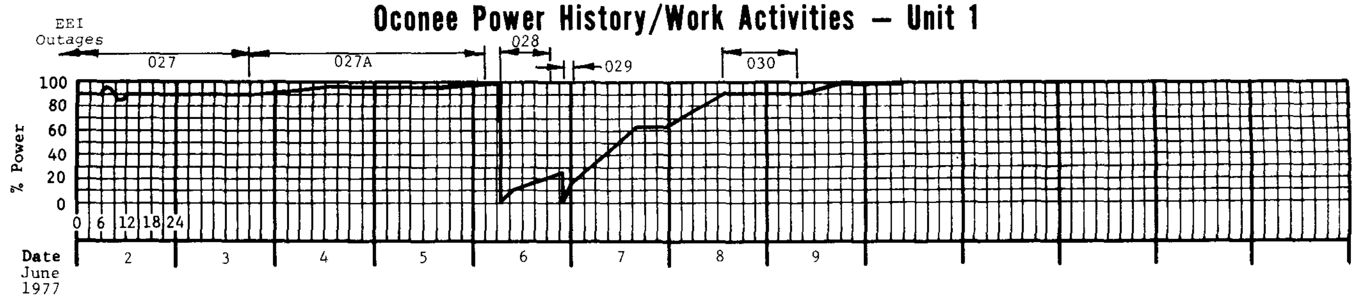
ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse

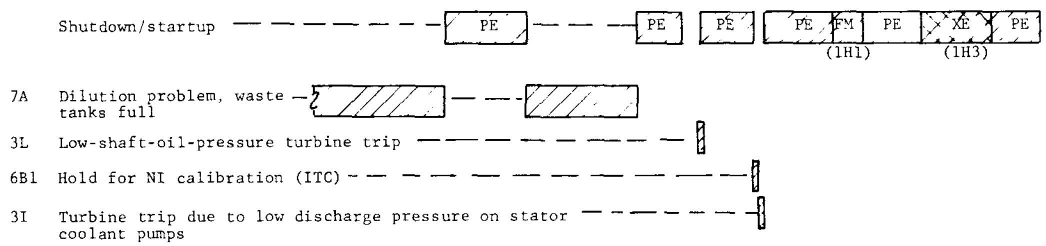
Figure D-1. (Cont'd)

DATES: 6/2/77-6/9/77

OUTAGES: 027-030

Oconee Power History/Work Activities – Unit 1





EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

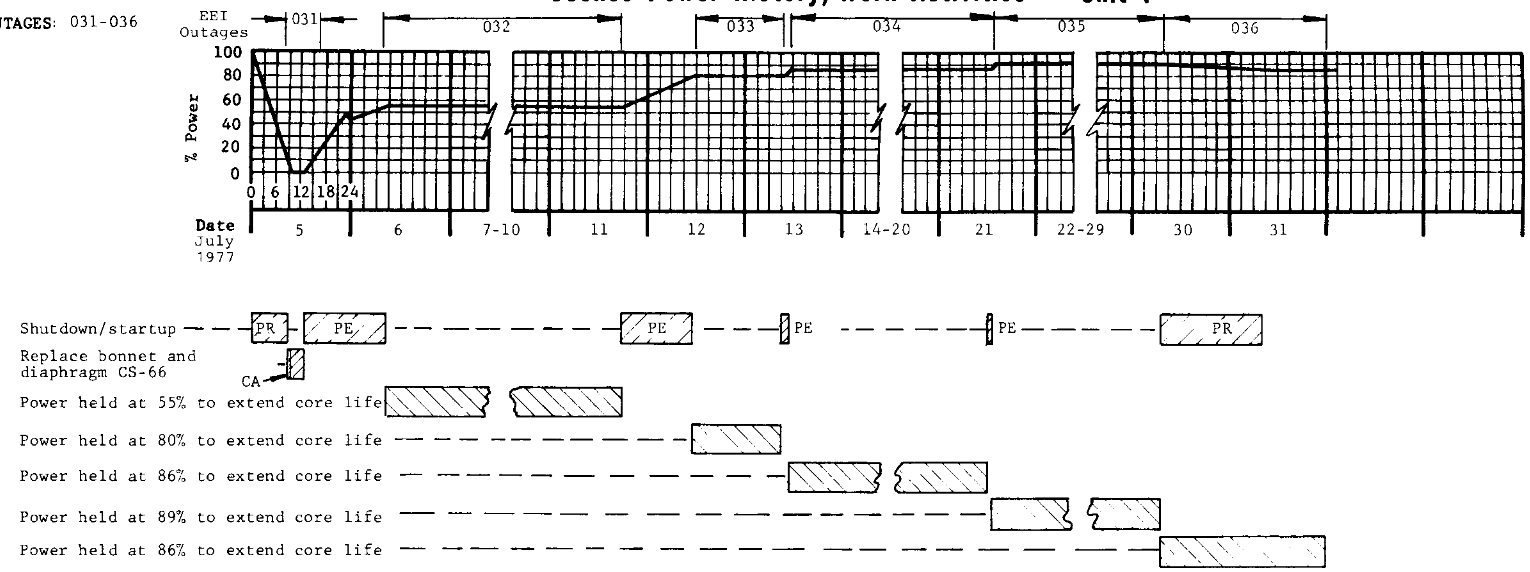
Hours	Description	Work	Cse

Figure D-1. (Cont'd)

DATES: 7/5-7/31/77

OUTAGES: 031-036

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

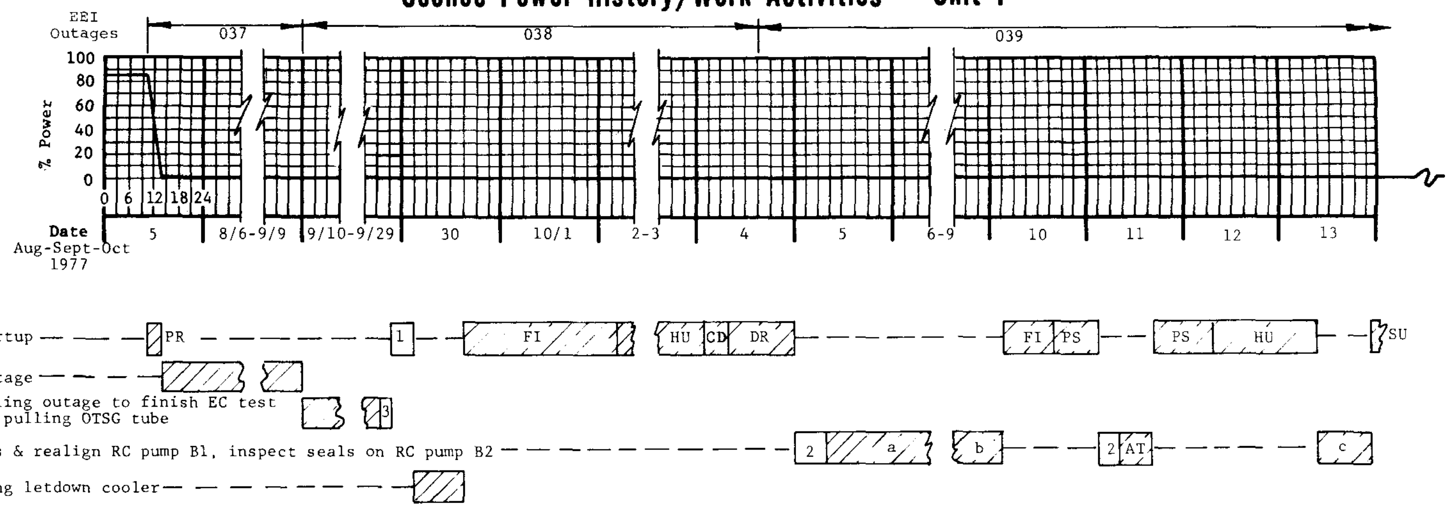
Hours	Description	Work	Cse

Figure D-1. (Cont'd)

DATES: 8/5-10/13/77

OUTAGES: 037-039

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Replace seals & realign RCP 1B1 (75 h)
b – Inspect seals RCP 1B2 (42 h)
c – balance RCP 1B2 (13 h)

1 – Valve found shut checked open, delayed RCS fill for 4.5 h
2 – Delay due to insufficient number of qualified personnel (13 h)
3 – Delay due to Maintenance not informing Operations that work was complete (3 h)

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

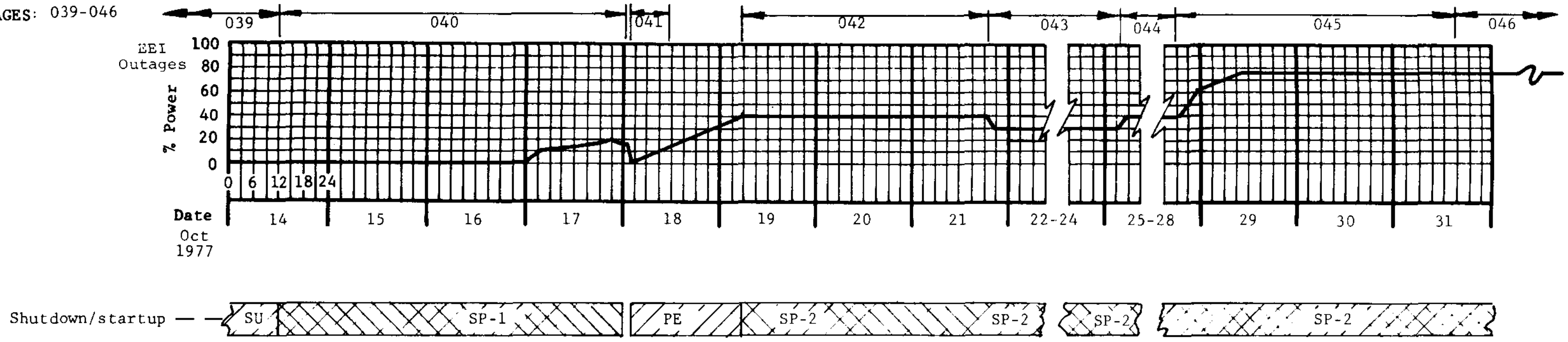
Hours	Description	Work	Cse

Figure D-1. (Cont'd)

DATES: 10/14-10/31/77

OUTAGES: 039-046

Oconee Power History/Work Activities – Unit 1



3C Turbine trip due to unepexted loss of "A" FDW pump (FDW swing)

1H2 Tilt in core (See SP-3, Sheet 3)

EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering
SP-1 – Startup Physics Test, Part 1
SP-2 – Startup Physics Test, Part 2

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

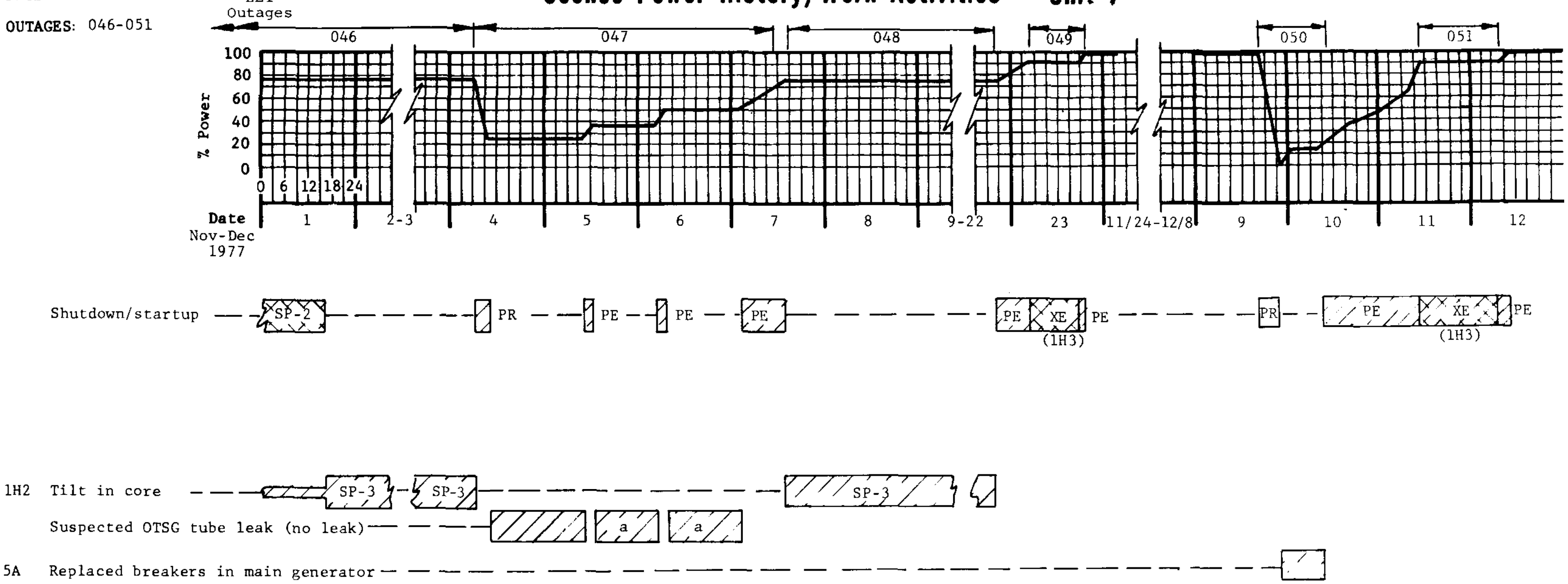
Hours	Description	Work	Cse

Figure D-1. (Cont'd)

DATES: 11/1-12/12/77

OUTAGES: 046-051

Oconee Power History/Work Activities – Unit 1



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

SP-2 – Startup Physics Test, Part 2
SP-3 – 155-hour delay in achieving full power because of an indicated core power tilt

a – RX power held at 38% and 50% to sample R1A-40 for activity check

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

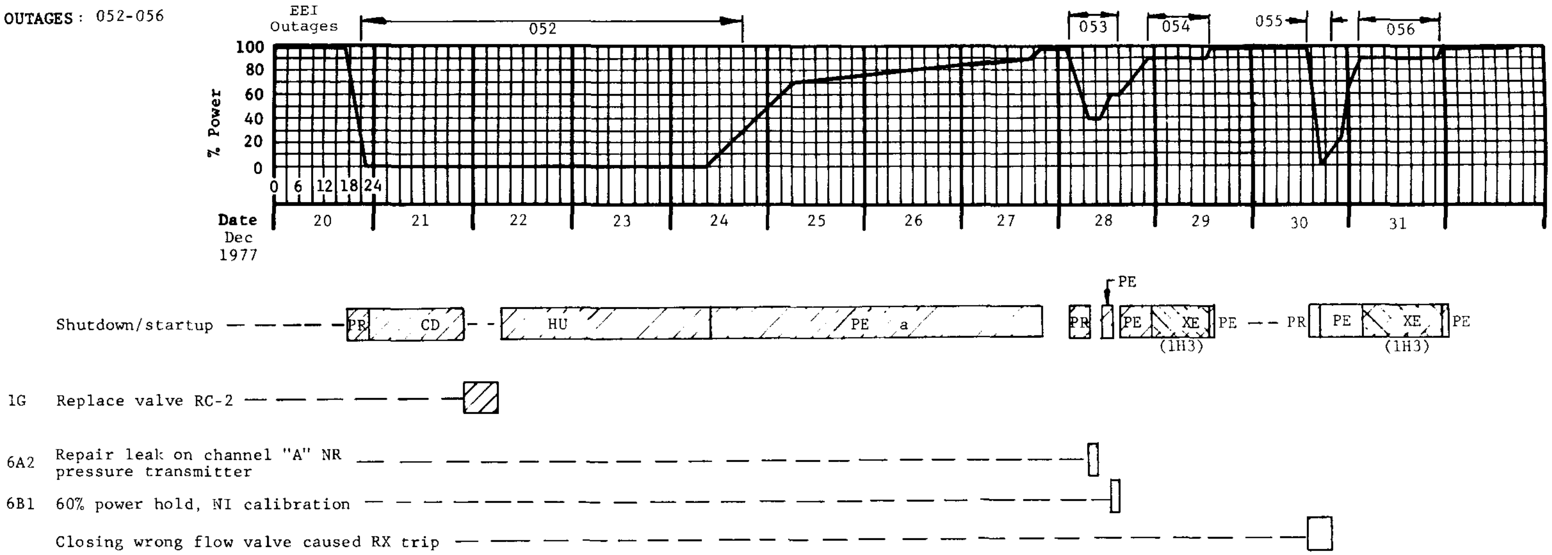
ADDITIONAL KEY ITEMS

Hours	Description	Work	Cse

Figure D-1. (Cont'd)

Oconee Power History/Work Activities – Unit 1

DATES: 12/20-12/31/77



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Slow rate of power escalation due to high chlorides in system-

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

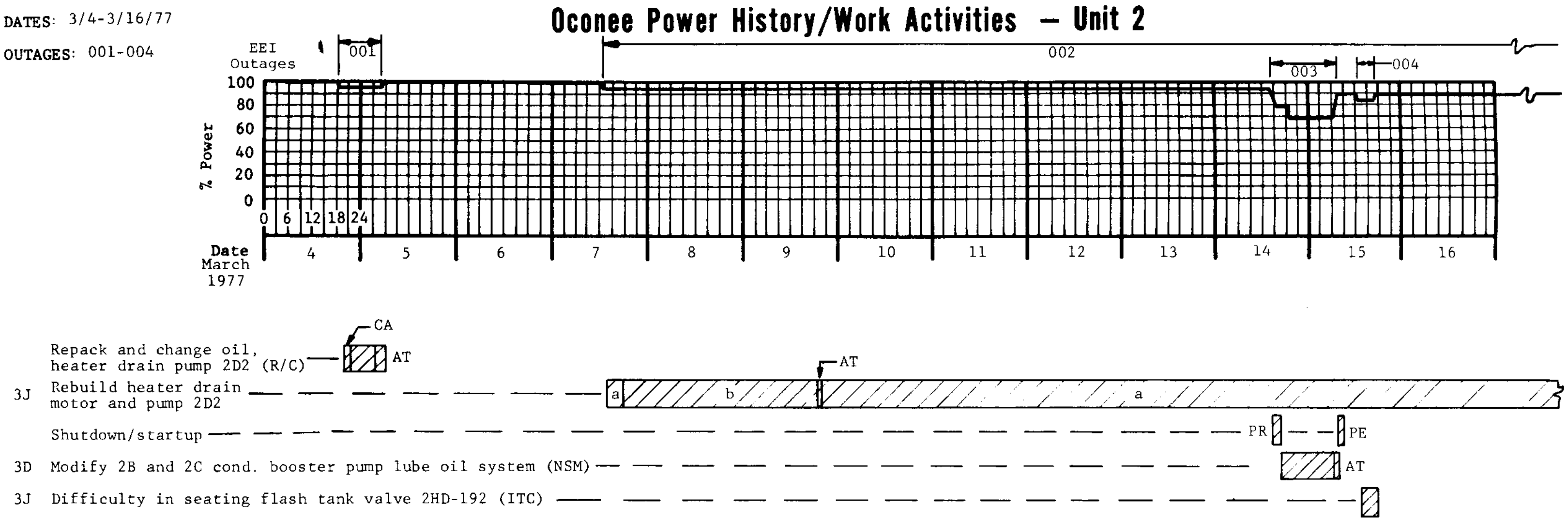
CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

Hours	Description	Work	Cse

Figure D-1. (Cont'd)



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Rebuilt heater drain pump 2D2 (R/C).
b – Cleaned motor lower bearing and shaft.

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

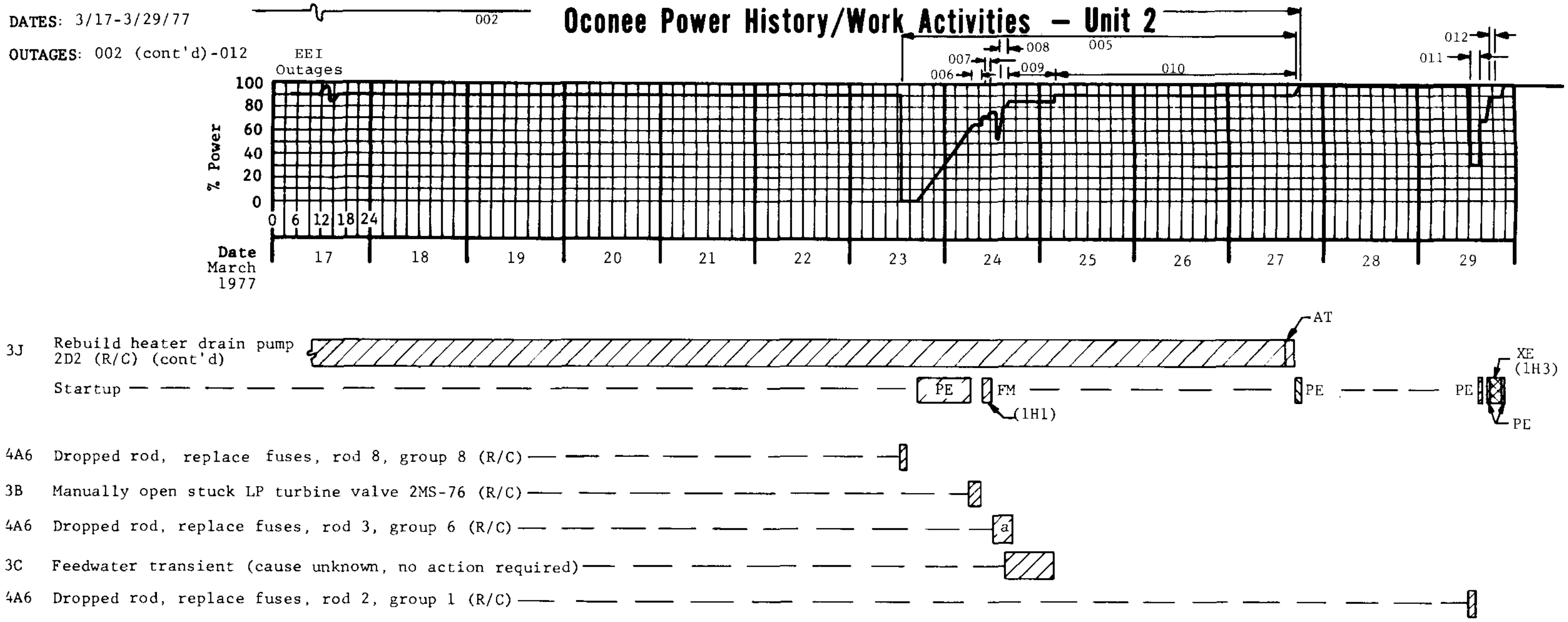
Hours	Description	Work	Cse

Figure D-2. Oconee Power History/Work Activities — Unit 2

Oconee Power History/Work Activities – Unit 2

DATES: 3/17-3/29/77

OUTAGES: 002 (cont'd)-012



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – Replaced CRD gate drive relay, phase & diode.

WORK CATEGORY	
R/C	Repair Correction
ITC	Inspection Testing Calibration
NSM	Nuclear Station Modification
OM	Operational Maintenance
PM	Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

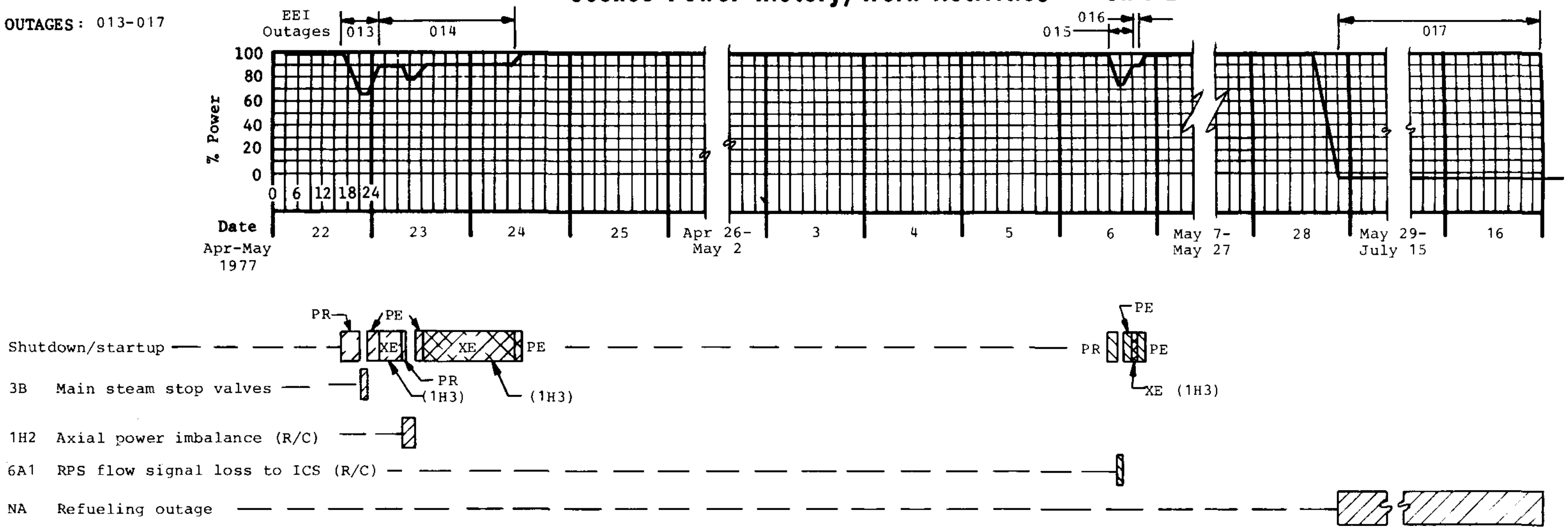
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 4/22/77-7/16/77

OUTAGES: 013-017

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

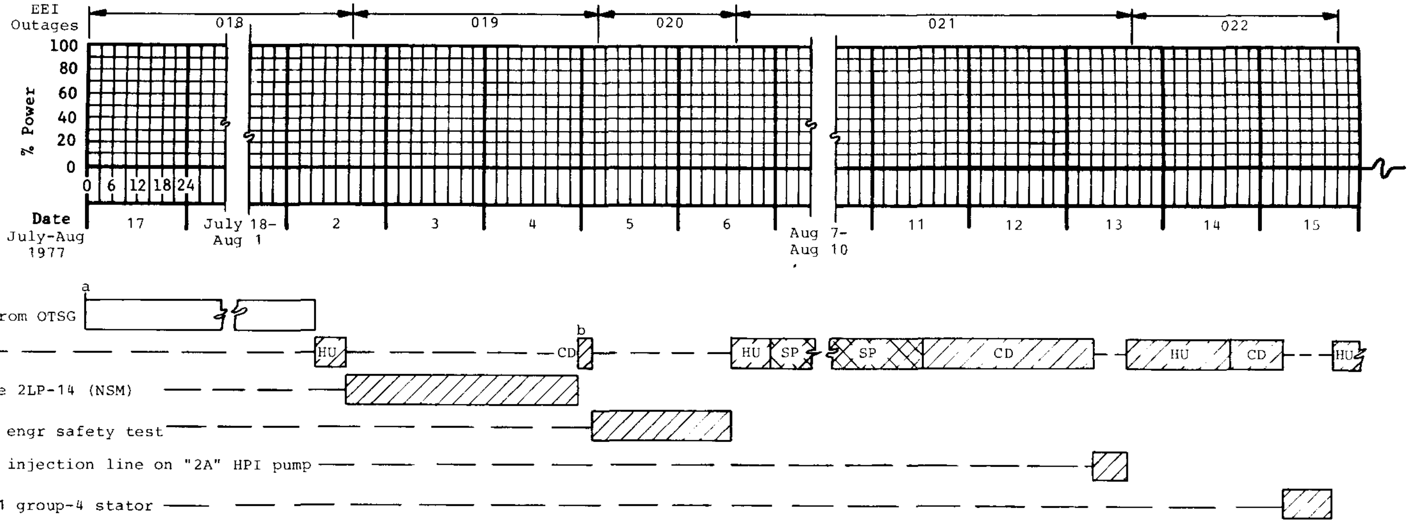
ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 7/17/77-8/15/77

OUTAGES: 018-022

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a — Completed refueling outage (EEI-017).

b — Unit had to be cooled down to perform LPI engineering safety test.

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

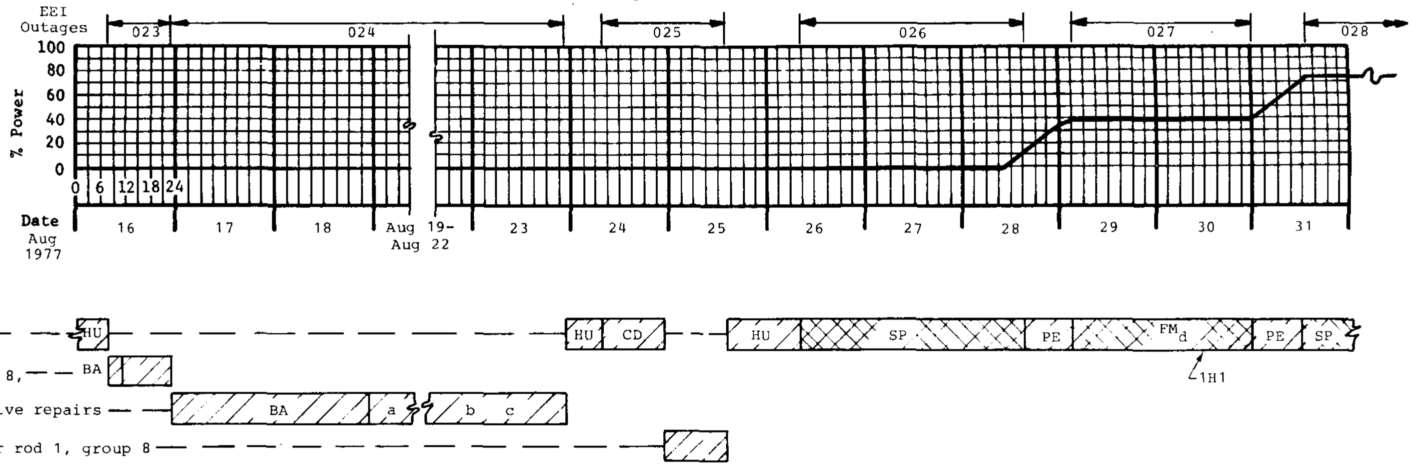
ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 8/16/77-8/31/77

OUTAGES: 023-028

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – Changed 26 power cables
b – Replaced 10 stators
c – Repaired 12 stators
} 120 hours
d – Xenon equilibrium

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

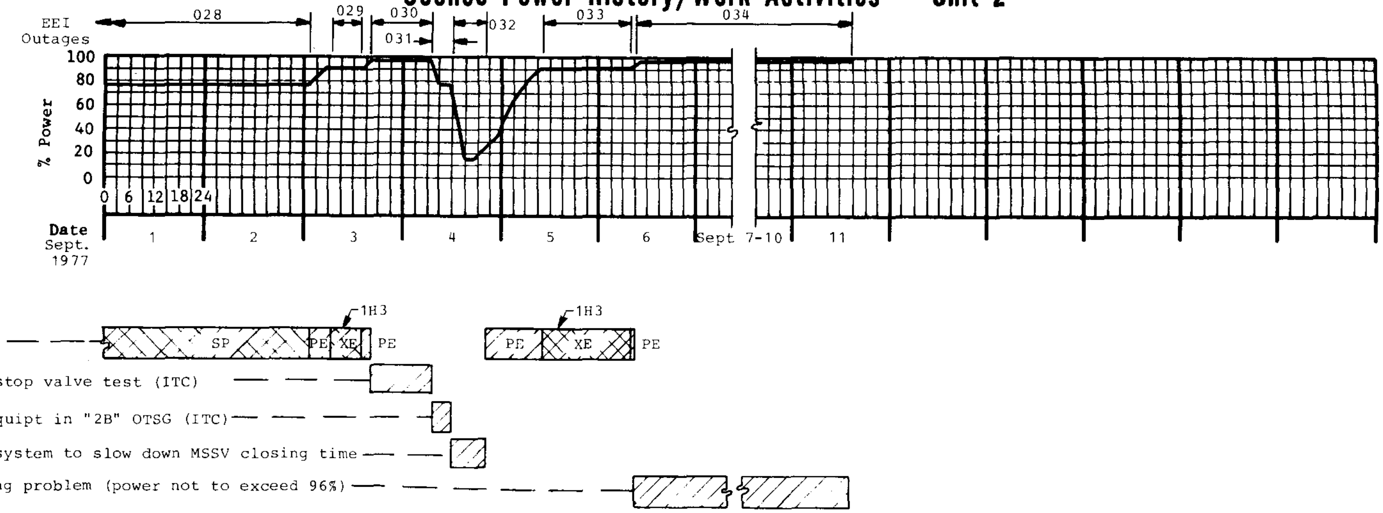
ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

Oconee Power History/Work Activities — Unit 2

DATES: 9/1/77-9/11/77

OUTAGES: 028-034



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

Abbreviations – SOAK: Second of a kind
MSSV: Main steam stop valve

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

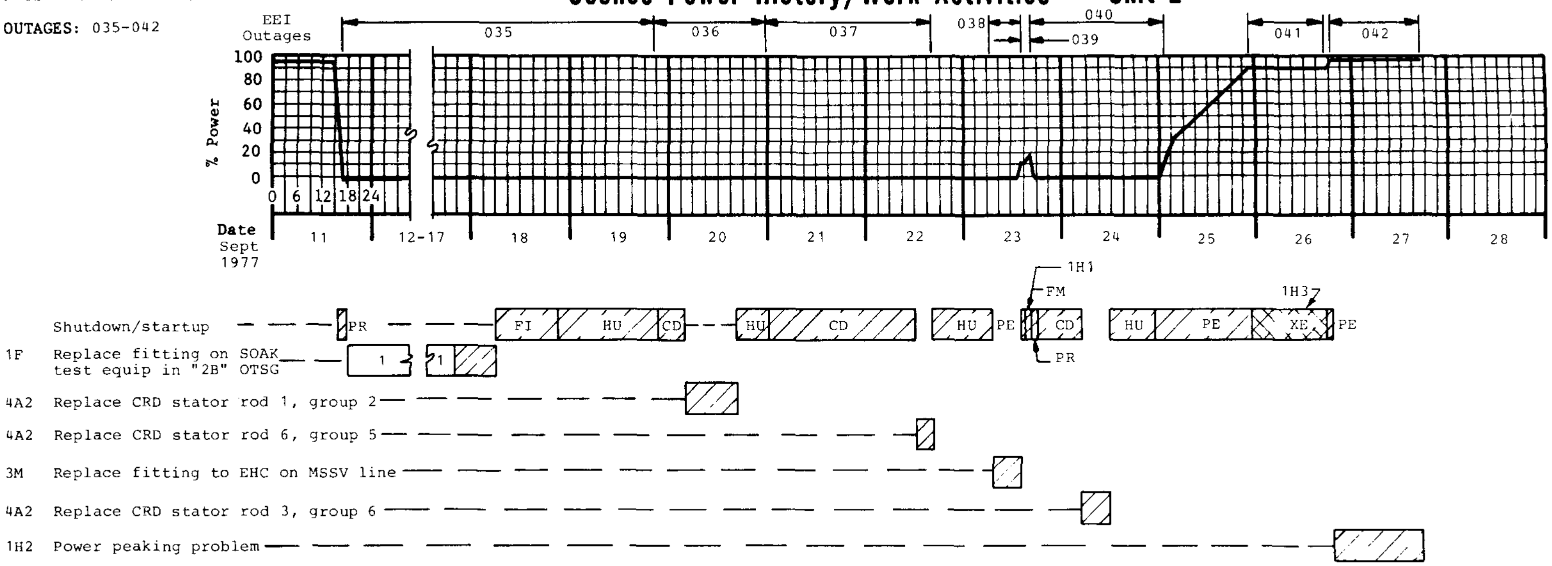
ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 9/11/77-9/27/77

OUTAGES: 035-042

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

1 – Delay due to gas in containment (cooldown and drain times not recorded, but in this time span, use average time tables).

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

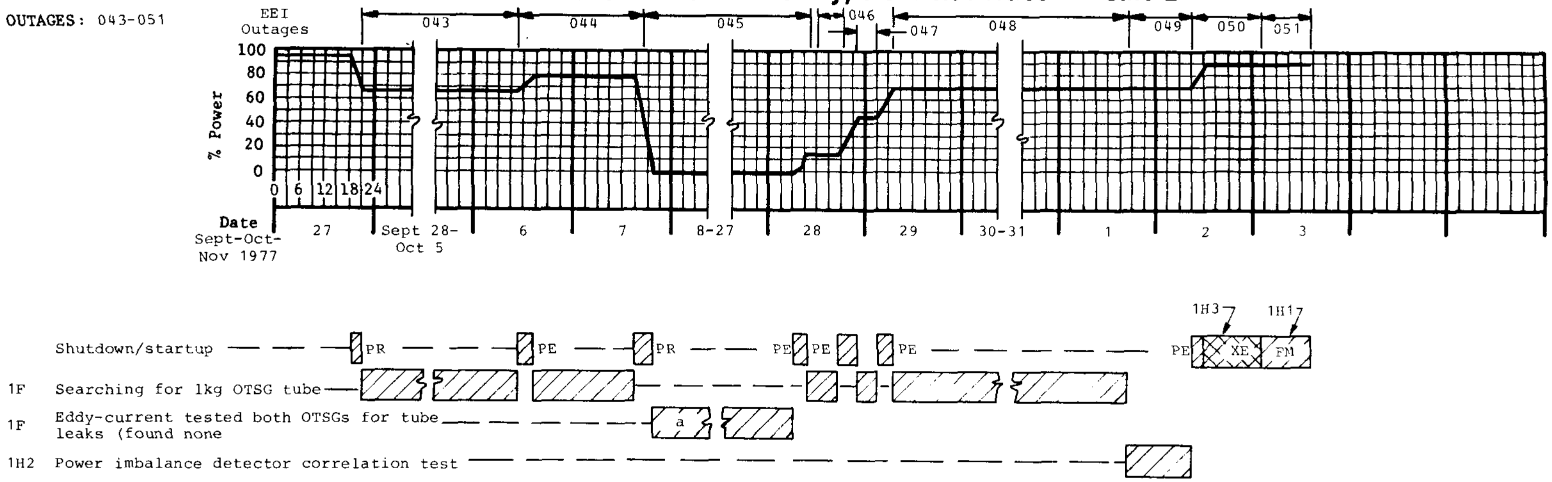
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 9/27/77-11/2/77

OUTAGES: 043-051

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a — No times for cooldown, drain, fill, heatup, and startup, etc. available; use average times from past outages.

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

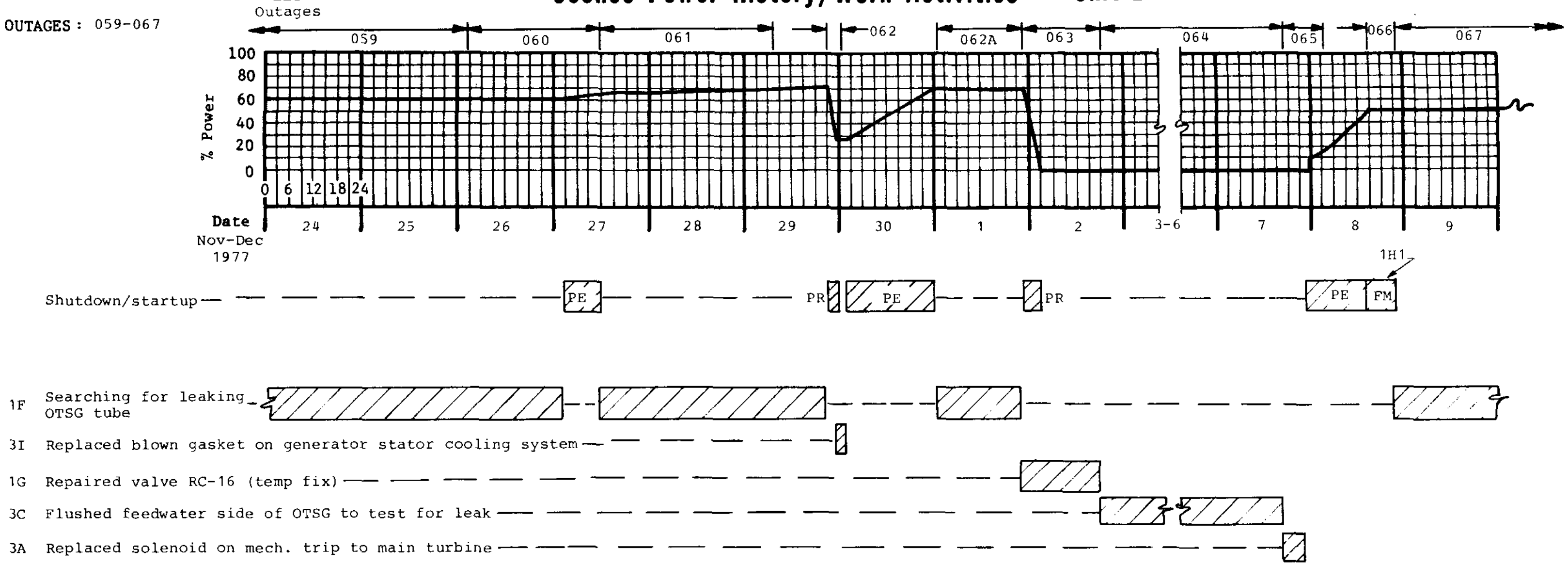
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES : 11/24/77-12/9/77

OUTAGES : 059-067

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS

- PR – Power Reduction
- CD – Cooldown
- DR – RCS Drain
- AT – Acceptance Test
- FI – RCS Fill
- PS – Prep. for Startup
- HU – RCS Heatup
- SU – Startup
- PE – Power Escalation
- BA – Building Access
- CA – Component Access
- XE – Xenon Hold
- FM – Fuel Maneuvering

WORK CATEGORY

- R/C – Repair Correction
- ITC – Inspection Testing Calibration
- NSM – Nuclear Station Modification
- OM – Operational Maintenance
- PM – Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

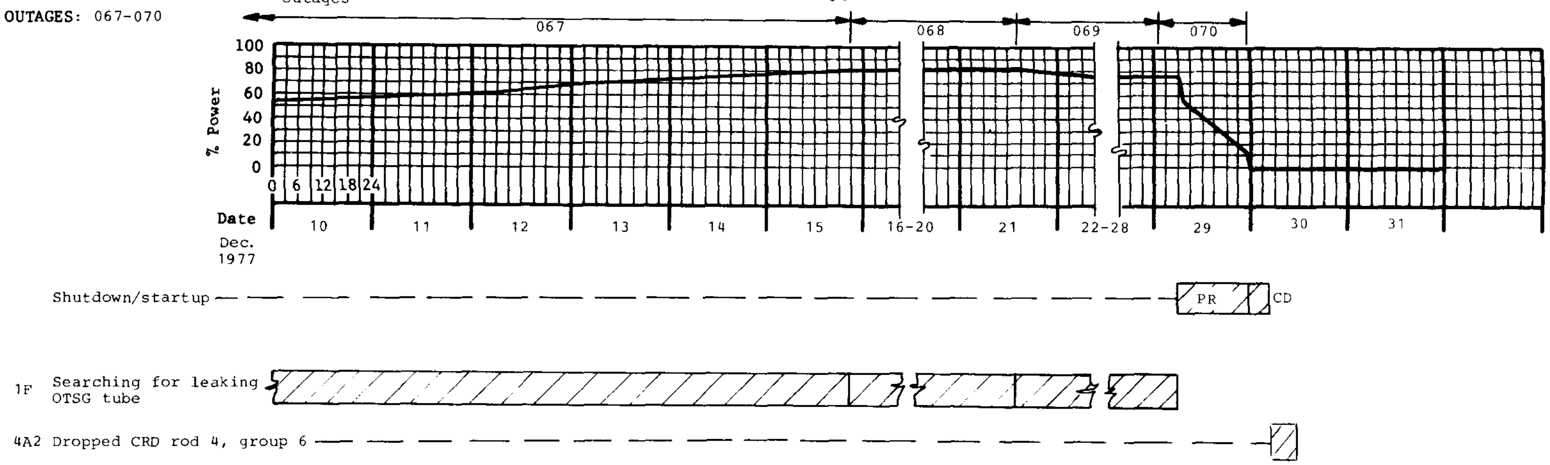
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 12/9/77-12/31/77

OUTAGES: 067-070

Oconee Power History/Work Activities – Unit 2



D-35

EVENTS/DELAYS	
PR – Power Reduction	SU - Startup
CD – Cooldown	PE - Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM · Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

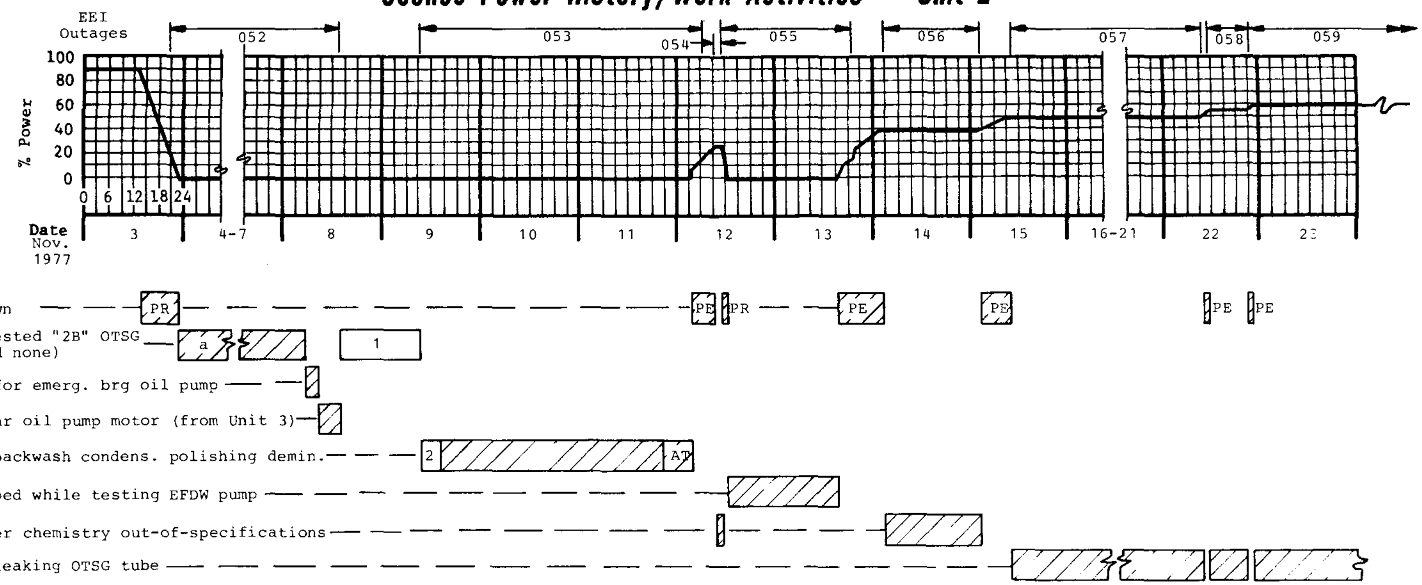
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 11/3/77-11/23/77

OUTAGES: 052-059

Oconee Power History/Work Activities – Unit 2



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – No times for cooldown, drain, fill, heatup, startup, etc. available; use average times from past outages.

b – Several Powdex cells found depleted; had to be flushed to resin pond.

1 – High activity in both OTSGs; drained, flushed, and refilled system (19 hours).

2 – Delay in obtaining material (5 hours).

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

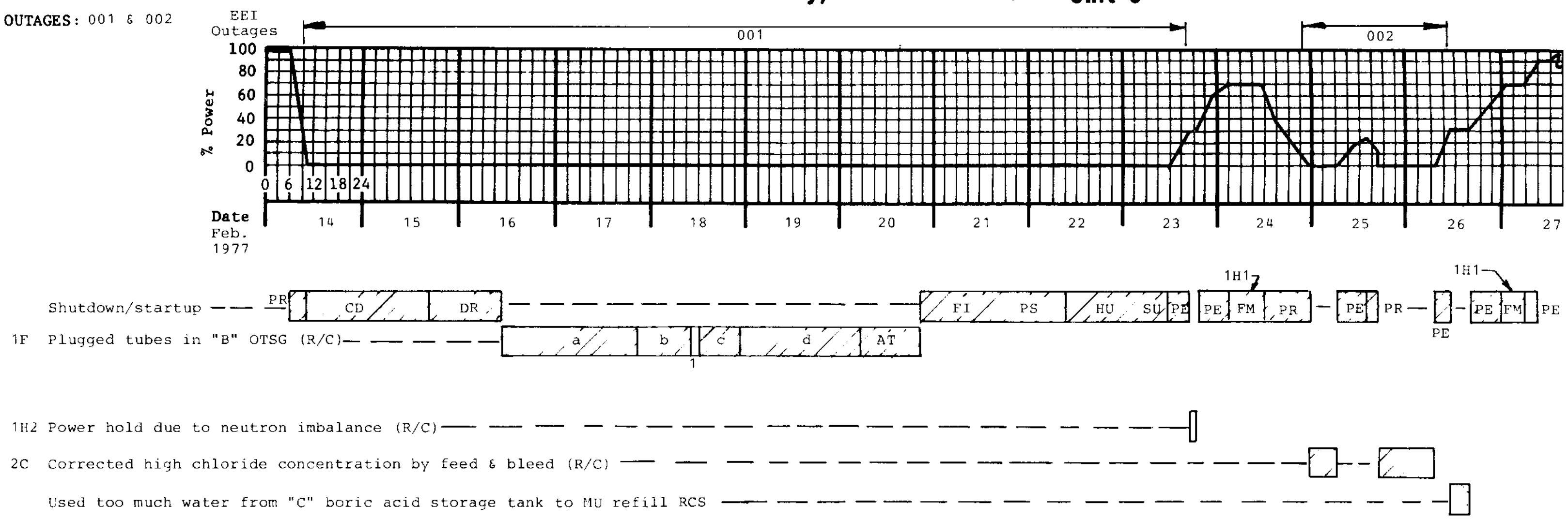
Hours	Description	Work	Cse

Figure D-2. (Cont'd)

DATES: 2/14/77-2/26/77

OUTAGES: 001 & 002

Oconee Power History/Work Activities – Unit 3



EVENTS/DELAYS	
PR – Power Reduction	SU - Startup
CD – Cooldown	PE - Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – Hydrotest "B" OTSG for tube leak.
b – Eddy-current test.
c – Fiber optics test.
d – Tube plugging.
1 – Fiber optics equipment broken.

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

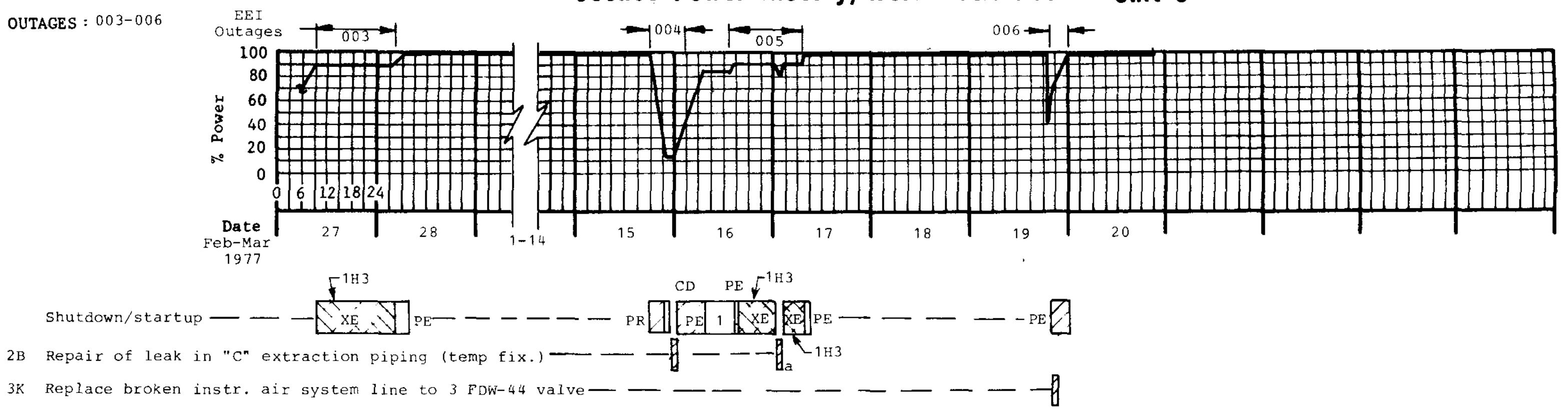
Hours	Description	Work	Cse
8	Repair oil leak on "A" FW pump bearing	R/C	EQ
4	Repair oil leak on "B" FW pump casing	R/C	EQ
4	Repair oil leak on 3A FW bearing	R/C	EQ
6	Repair oil leak suction 3B EHC pump	R/C	EQ
2	Repair 3A FWFT pump casing vent	R/C	EQ
4	Check alarm 3SA-8 CFT "B" outlet valve	itc	EQ
2	Repair air leak 3MS-19 cont. turb byp vlv	R/C	EQ
20	Repair hyd suppressor 3-03-0-2480B-H6B	R/C	EQ
5	Inspect hydraulic suppressors	ITC	EQ
16	Revise hanger & hanger sketch	NSM	EQ
8	Replace valve stem in 3HP-355	R/C	EQ
18	Repair leak in 3CF-5	R/C	EQ
6	Repair damaged suppressor link 3RC-3	R/C	EQ
6	Replace valve stem in 3HP-236	R/C	EQ
8	Repair 3A EHC pump oil leak	R/C	EQ
4	Repair leak CRD motor tube F2 clos. HD	R/C	EQ
2	Replace leaking gasket 3B letdown filter	R/C	EQ

Figure D-3. Oconee Power History/Work Activities – Unit 3

DATES : 2/27/77-3/20/77

OUTAGES : 003-006

Oconee Power History/Work Activities – Unit 3



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Lost "C" extraction.
1 – Problem deborating.

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

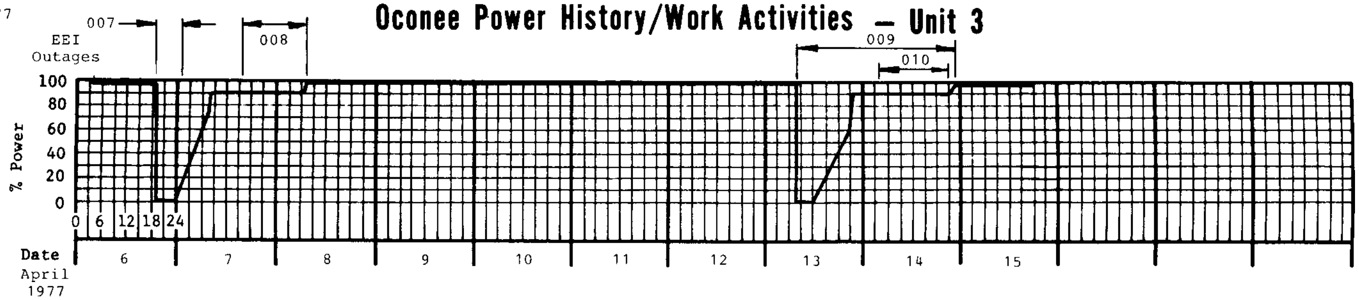
ADDITIONAL KEY ITEMS			
Hours	Description	Work	Cse
14	Repack valves 3HP-214, -145, -144, -356	R/C	EQ
2	Repair 3HP-254 plug leak valves	R/C	EQ
16	Repack 3HP-240, -127, -200 valves	R/C	EQ
2	Repack 3A1 RC pump, west side	R/C	EQ
4	Repack 3RC-2 valve	R/C	EQ
24	Check all CRD support structure fans	ITC	EQ
8	Repair 3MS-78 position indicator	R/C	EQ
2	Check valve HP-3, improper alarm	ITC	EQ
3	Repair 3 MS-126	R/C	EQ
7	Repair 3A2 RCP upper seal instrumentation	R/C	EQ
2	Repair 3A1 RCP motor cooler instrumentation	R/C	EQ
11	Repack valve 3HD-96	R/C	EQ
2	Repair oil leak on main turb @ turning gear	R/C	EQ
5	Repack valve 3MS-22	R/C	EQ
2	Clean 3B seal supply filter	R/C	EQ
2	Replace 3A1 chgm. sample pl. gaskets, bolts	R/C	EQ
10	Check RC pump motor oil levels	ITC	EQ
1	Clean 3A seal supply filter	R/C	EQ
5	Repack valve 3MS-31	R/C	EQ
24	Repair hydraulic suppressor 3B1-S53	R/C	EQ
4	Repack valve 3RC-7	R/C	EQ
4	Repack valve 3HP-206	R/C	EQ

Figure D-3. (Cont'd)

DATES : 4/6/77-4/14/77

OUTAGES : 007-010

Oconee Power History/Work Activities – Unit 3



3M Turbine trip due to loss of 120 V dc to EHC system

Startup — PE XE PE — 1H3 — PE XE PE — 1H3

1H2 Reactor trip on flux/flow imbalance — BA AT

D-39

EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

WORK CATEGORY

- R/C – Repair Correction
- ITC – Inspection Testing Calibration
- NSM – Nuclear Station Modification
- OM – Operational Maintenance
- PM Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

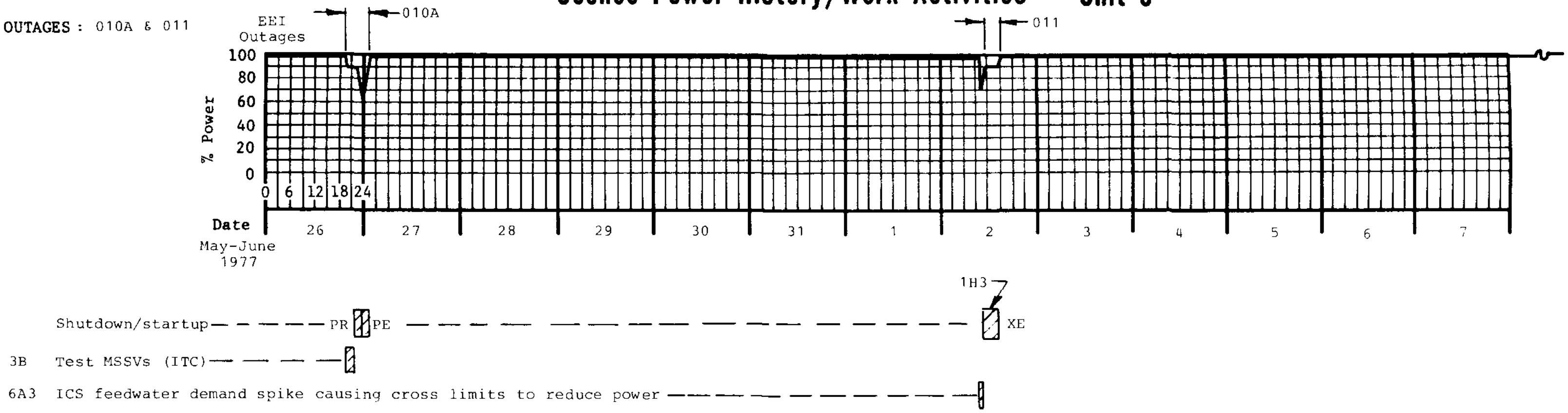
Hours	Description	Work	Cse

Figure D-3. (Cont'd)

DATES : 5/26/77-6/7/77

OUTAGES : 010A & 011

Oconee Power History/Work Activities – Unit 3



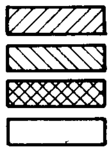
EVENTS/DELAYS

- PR – Power Reduction
- CD – Cooldown
- DR – RCS Drain
- AT – Acceptance Test
- FI – RCS Fill
- PS – Prep. for Startup
- HU – RCS Heatup
- SU - Startup
- PE – Power Escalation
- BA – Building Access
- CA – Component Access
- XE – Xenon Hold
- FM – Fuel Maneuvering

WORK CATEGORY

- R/C – Repair Correction
- ITC - Inspection Testing Calibration
- NSM – Nuclear Station Modification
- OM – Operational Maintenance
- PM Preventive Maintenance

CAUSE CATEGORY



- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

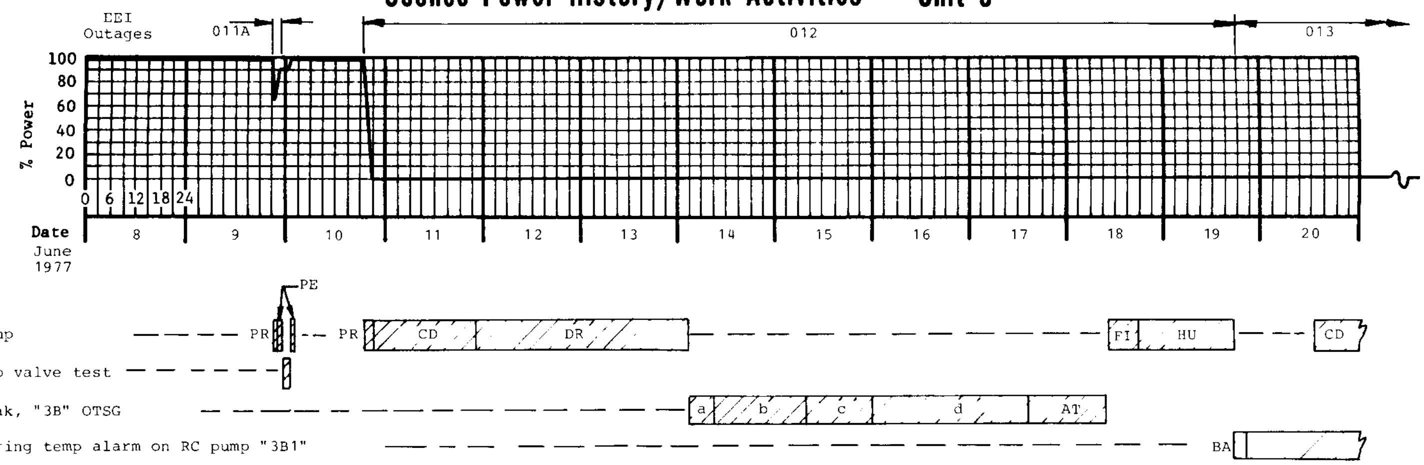
Hours	Description	Work	Cse

Figure D-3. (Cont'd)

DATES: 6/8/77-6/20/77

OUTAGES: 011A-013

Oconee Power History/Work Activities – Unit 3



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – Flushed "3B" OTSG due to high activity.
b – Hydrotested "3B" OTSG for tube leak.
c – Eddy-current test.
d – Tube plugging.

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

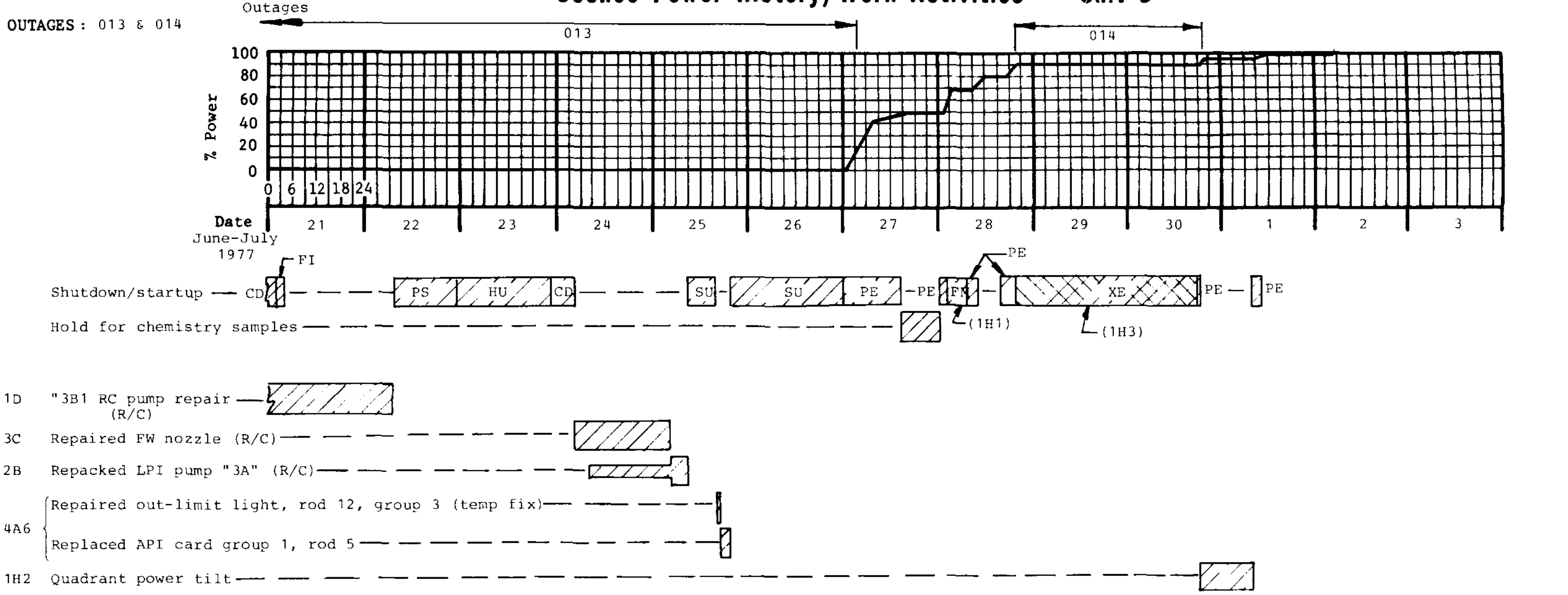
Hours	Description	Work	Cse
14	Repair 3A stator pump	R/C	EQ
4	Check & repair EFWPT oil relief valve	R/C	EQ
4	Repair leak on steam trap off 3A FDWPT HP chest	R/C	EQ
3	Repack valve 3MS-82	R/C	EQ
29	Repack valves 3FDW-23, -28, -40, -53, -65	R/C	EQ
7	Repair 3A FWPT steam trap	R/C	EQ
4	Weld repair body of valve LPSW-117	R/C	EQ
5	Repair valve 3C-7	R/C	EQ
5	Check & repair 3A2, 3B2 seal leak sig in RB	R/C	EQ
5	Replace PI tube, group 7, rod 1	R/C	EQ
3	Repair reactor building personnel hatch	R/C	EQ

Figure D-3. (Cont'd)

Oconee Power History/Work Activities – Unit 3

DATES : 6/21/77-7-1/77

OUTAGES : 013 & 014



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
SU - Startup
PE - Power Escalation
BA - Building Access
CA - Component Access
XE - Xenon Hold
FM - Fuel Maneuvering

WORK CATEGORY

R/C - Repair Correction
ITC - Inspection Testing Calibration
NSM - Nuclear Station Modification
OM - Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

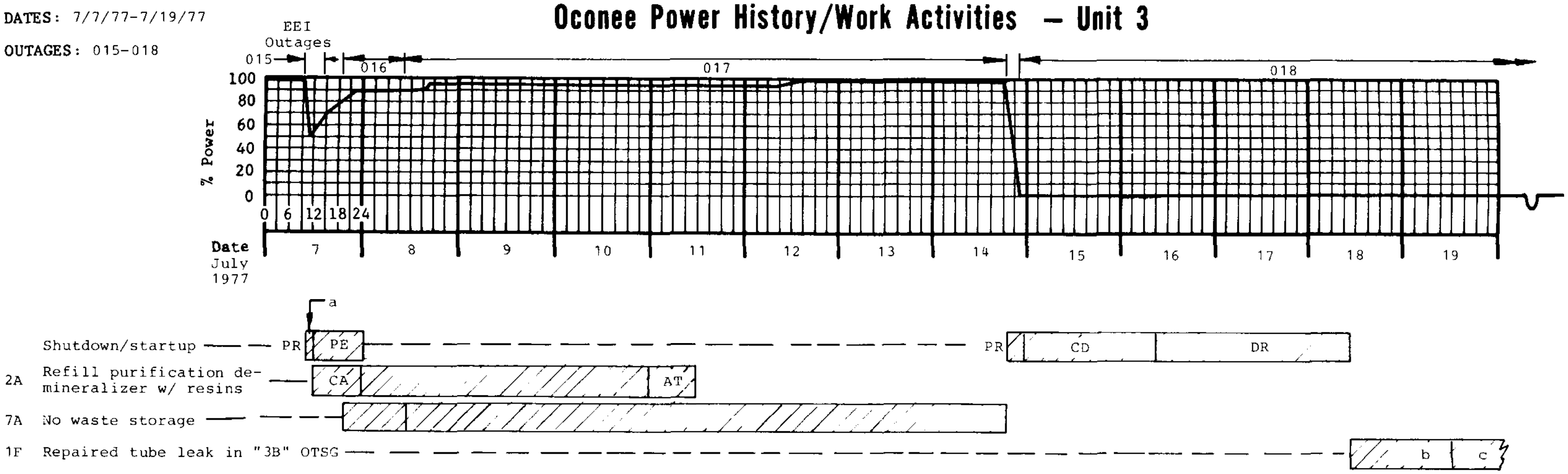
Hours	Description	Work	Cse
4	Change "A" seal supply filter	PM	EQ
3	Repair plug leak valve 3RC-38	R/C	EQ
5	Repair leak valve 3HP-126	R/C	EQ
8	Change oil in RC pump motor 3B2	PM	EQ
8	Change oil in RC pump motor 3A2	PM	EQ
1	Check 3D2 heater drain pump	R/C	EQ
10	Replace term. blks, clean leads for all three hotwell pumps	R/C	EQ

Figure D-3. (Cont'd)

DATES: 7/7/77-7/19/77

OUTAGES: 015-018

Oconee Power History/Work Activities – Unit 3



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – Demineralizer lost ability to remove chlorides.
b – Hydrotest "3B" OTSG for leaking tube.
c – Drained "3B" OTSG after hydrotest.

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM – Preventive Maintenance

CAUSE CATEGORY

- Equipment Deficiencies and Failures
- Operating Practice/Requirements
- Regulatory
- Delays

ADDITIONAL KEY ITEMS

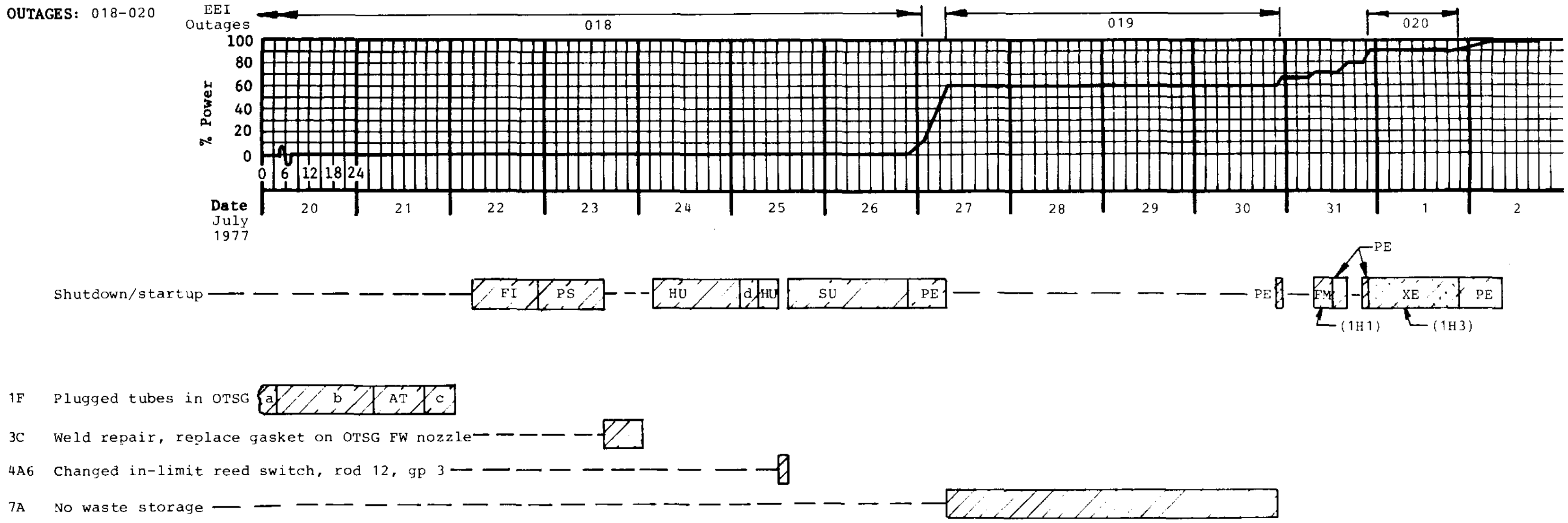
Hours	Description	Work	Cse
40	Replaced valves GWD-59, LWD-230	R/C	EQ
	Repaired 3B stator cooling pump	R/C	EQ
10	Repaired FA-20 feedwater nozzle on "3B" OTSG	R/C	EQ
10	Repaired "3B" air ejector	R/C	EQ

Figure D-3. (Cont'd)

DATES: 7-20/77-8/2/77

Oconee Power History/Work Activities – Unit 3

OUTAGES: 018-020



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

a – Drained "3B" OTSG after hydrotest.
b – Plugged tubes.
c – Secured "3B" OTSG.
d – Deborated reactor coolant system.

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

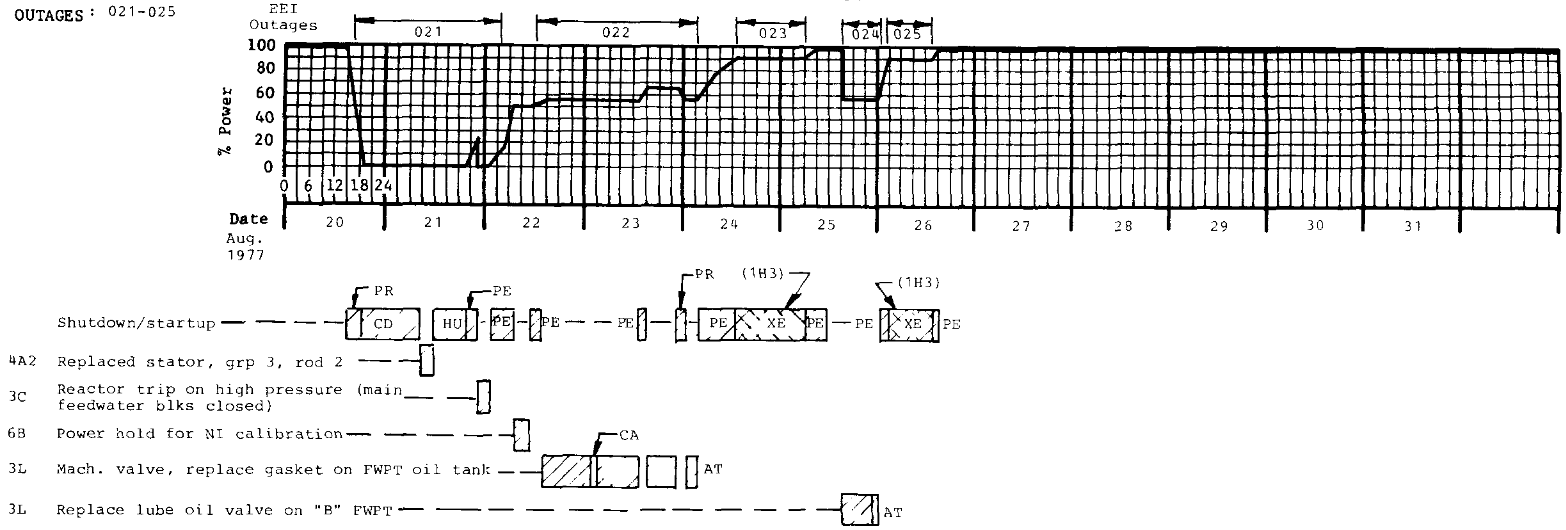
Hours	Description	Work	Cse

Figure D-3. (Cont'd)

DATES: 8/20/77/8/26/77

OUTAGES: 021-025

Oconee Power History/Work Activities – Unit 3



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	

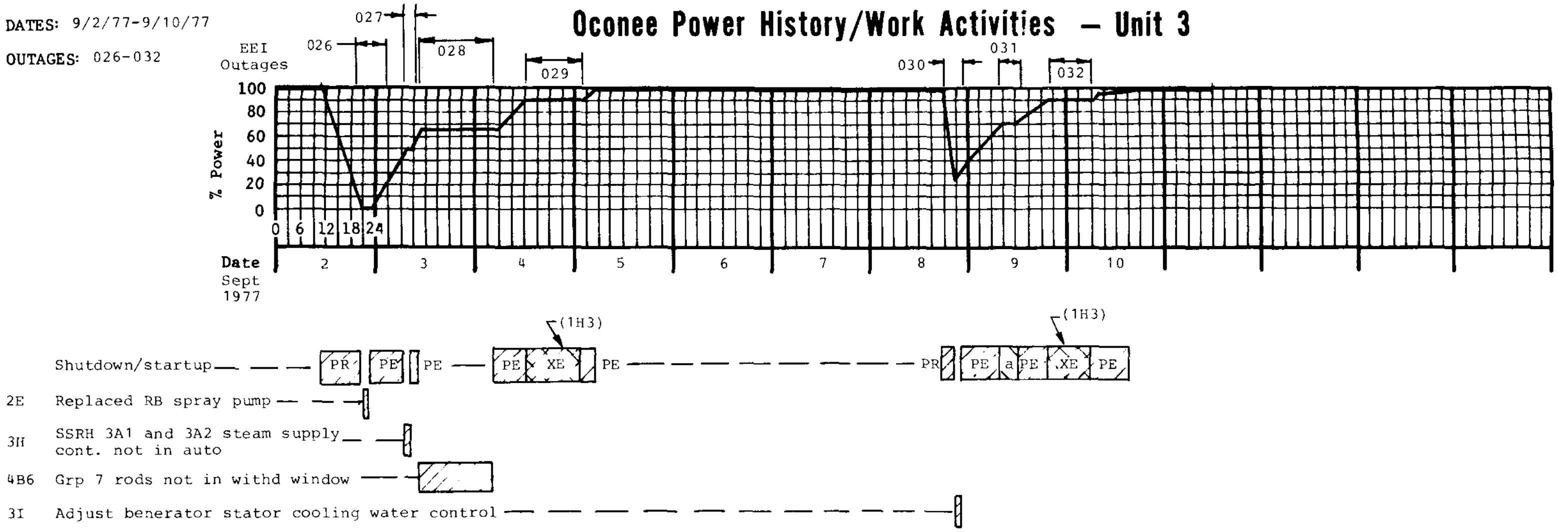
WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

Hours	Description	Work	Cse

Figure D-3. (Cont'd)



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	
a – EOCL reactivity (control rod) adjustment	

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

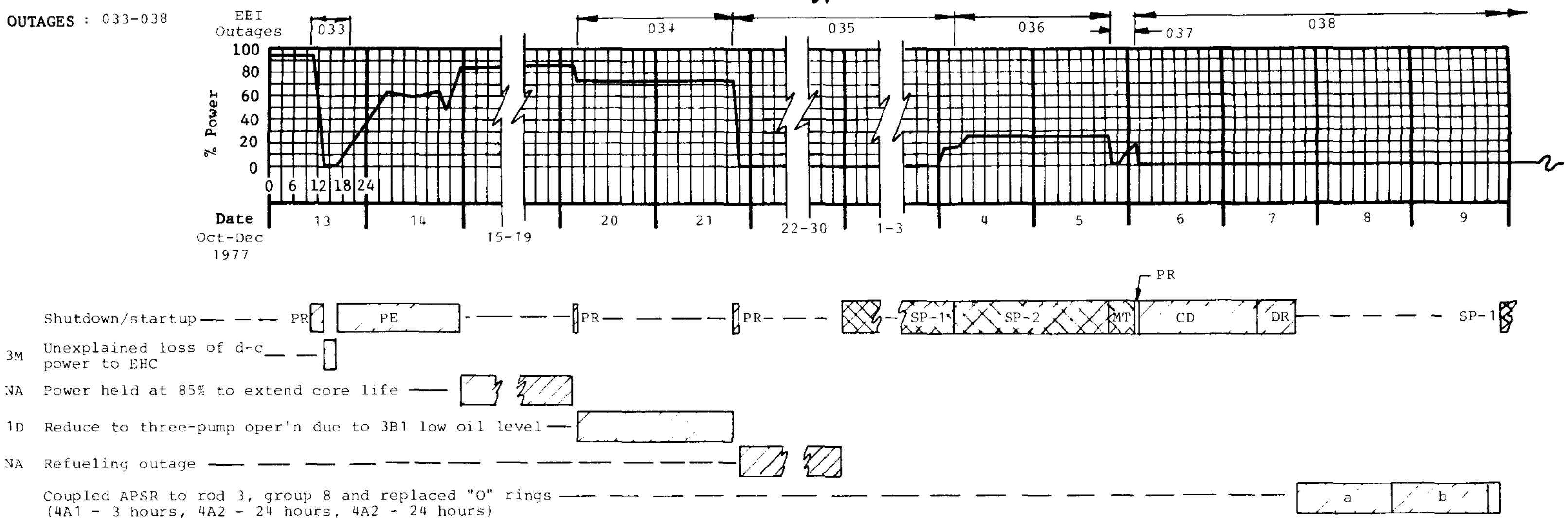
Hours	Description	Work	Cse

Figure D-3. (Cont'd)

DATES : 10/13/77-12/9/77

OUTAGES : 033-038

Oconee Power History/Work Activities – Unit 3



EVENTS/DELAYS

PR – Power Reduction
CD – Cooldown
DR – RCS Drain
AT – Acceptance Test
FI – RCS Fill
PS – Prep. for Startup
HU – RCS Heatup
MT – Manual trip test
SP-1 – Startup physics test, part 1
SP-2 – Startup physics test, part 2
SU – Startup
PE – Power Escalation
BA – Building Access
CA – Component Access
XE – Xenon Hold
FM – Fuel Maneuvering

a – Dry out two stators (24 hours).
b – Replaced bad stator (24 hours).

WORK CATEGORY

R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY

Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

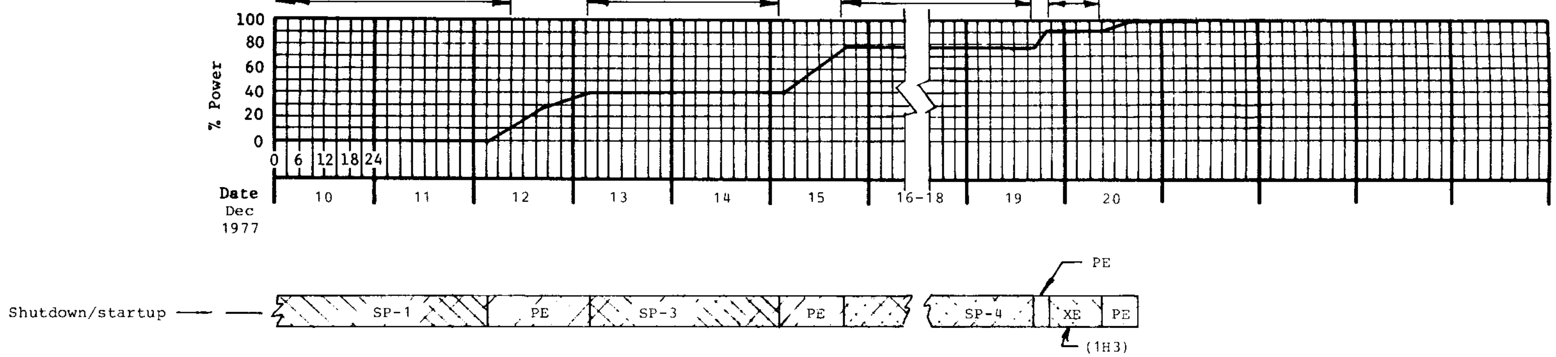
Hours	Description	Work	Cse

Figure D-3. (Cont'd)

DATES: 12/10/77-12/20/77

OUTAGES: 038-041

Oconee Power History/Work Activities — Unit 3



EVENTS/DELAYS	
PR – Power Reduction	SU – Startup
CD – Cooldown	PE – Power Escalation
DR – RCS Drain	BA – Building Access
AT – Acceptance Test	CA – Component Access
FI – RCS Fill	XE – Xenon Hold
PS – Prep. for Startup	FM – Fuel Maneuvering
HU – RCS Heatup	
SP-1 – Startup physics test, part 1	
SP-3 – Startup physics test, part 3	
SP-4 – Startup physics test, part 4	

WORK CATEGORY
R/C – Repair Correction
ITC – Inspection Testing Calibration
NSM – Nuclear Station Modification
OM – Operational Maintenance
PM Preventive Maintenance

CAUSE CATEGORY
Equipment Deficiencies and Failures
Operating Practice/Requirements
Regulatory
Delays

ADDITIONAL KEY ITEMS

Hours	Description	Work	Cse

Figure D-3. (Cont'd)

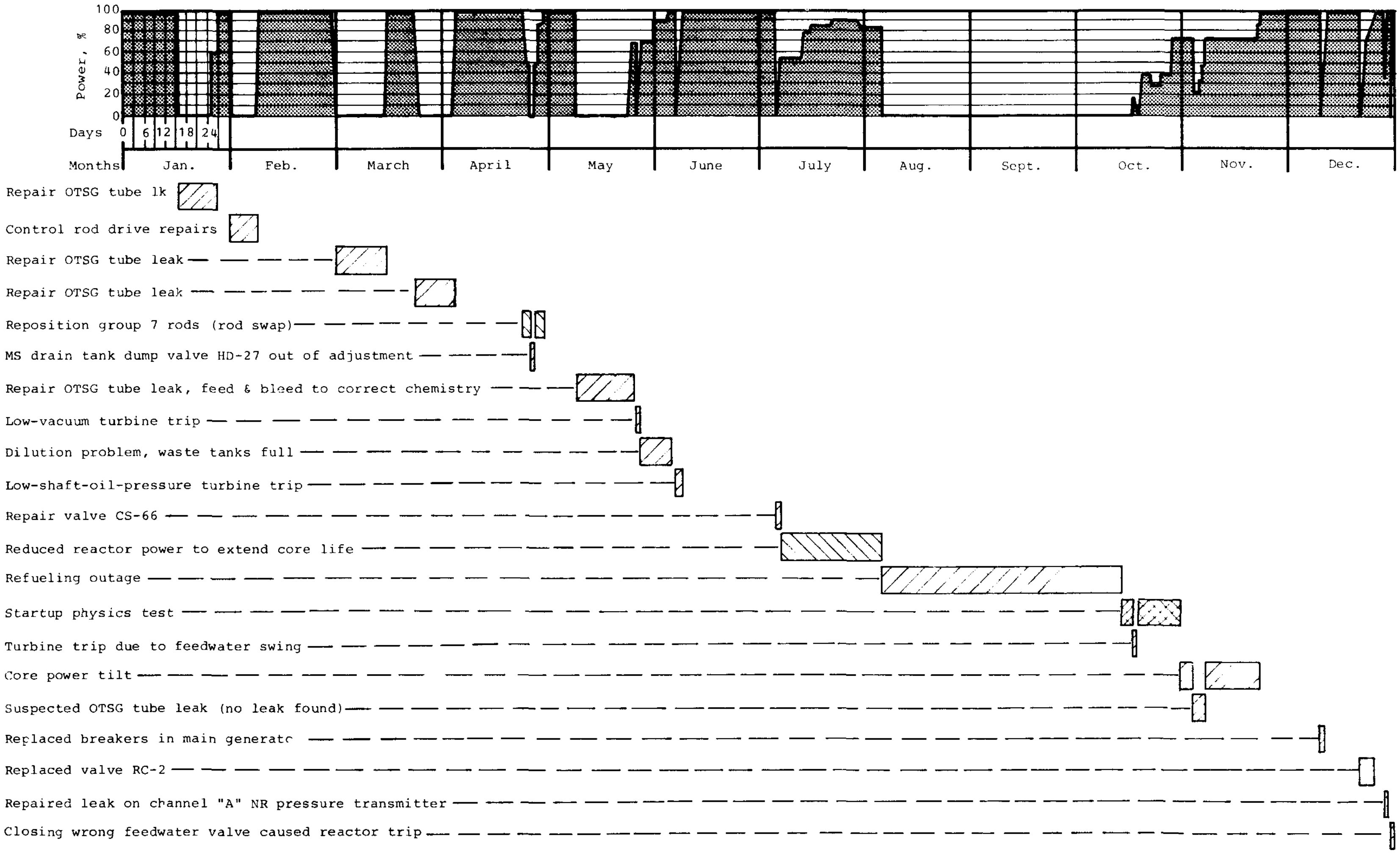


Figure D-4. 1977 Oconee 1 Operating Record

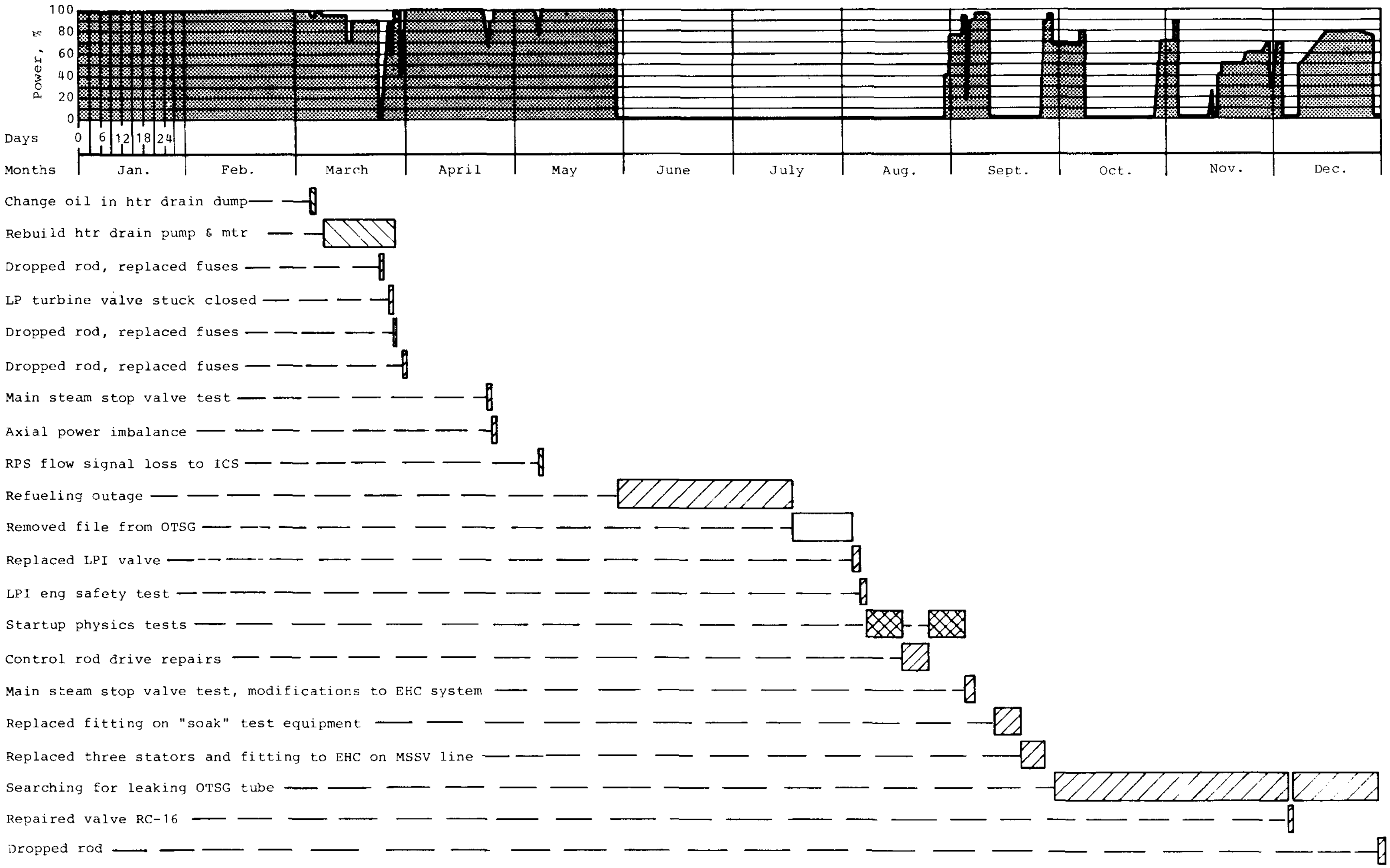


Figure D-5. 1977 Oconee 2 Operating Record

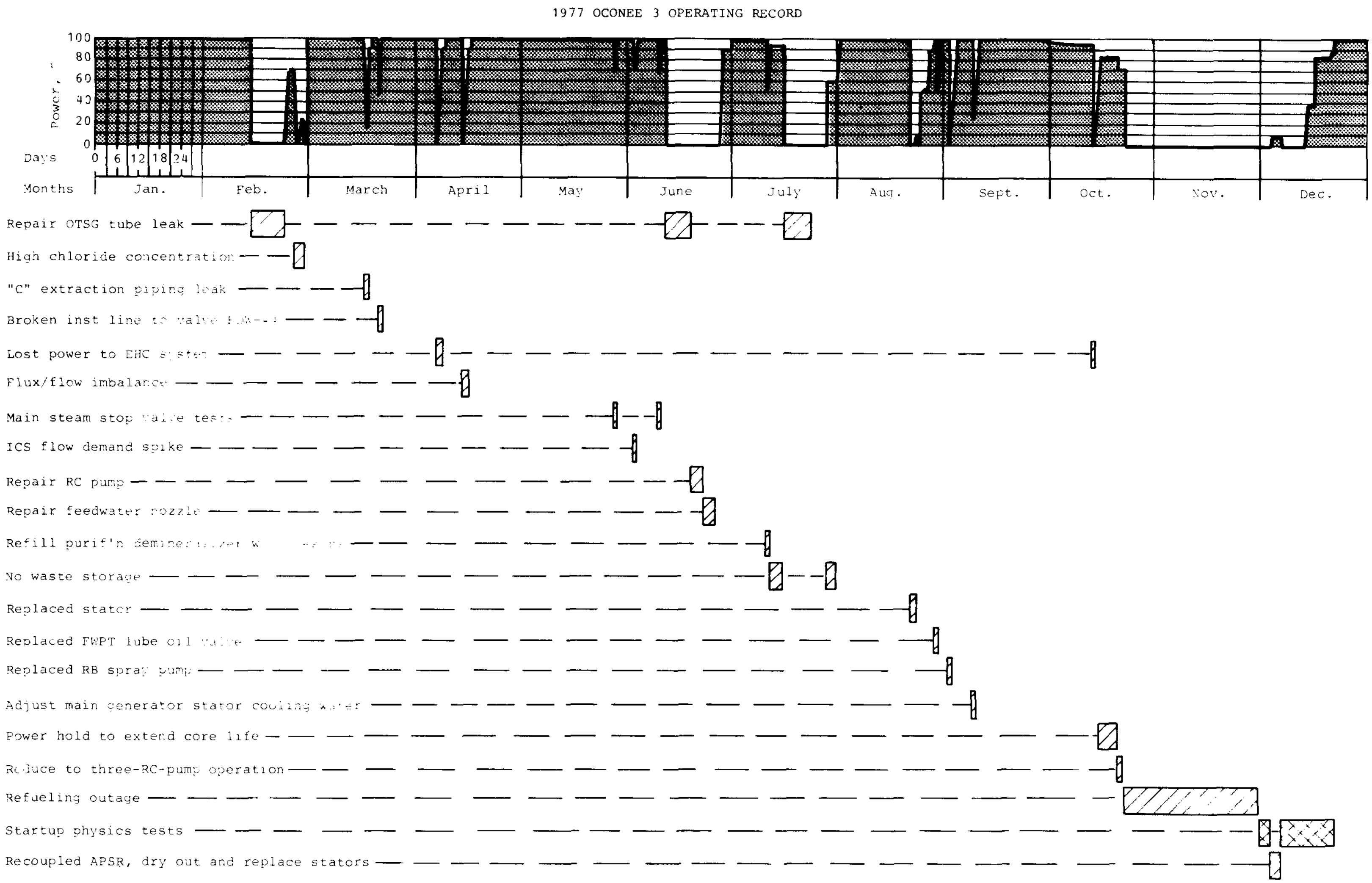


Figure D-6. 1977 Oconee 3 Operating Record

APPENDIX E

Oconee Unit 1 Work Events Tables

As discussed in Appendix B, historical data were obtained entirely from historical records, such as Station Work Requests, NRC Gray Books, and EEI reports. From these sources as much detail as possible was identified for each work event and recorded on the historical data work event sheets given as Table E-1.

Key item work, as defined in Appendix B, paragraph 2, and as identified in Table D-1, was categorized and grouped by system and component. Failure data were listed chronologically within each component. The listing gives such basic information as component identification, manufacturer, date of failure, the number of men and clock hours to repair, the plant's actual power level, and states whether the event forced or extended an outage. Actual work times were used if available; if actual times were not available, work request planning estimates or "best estimates" by the project team were used. As with the current data given in Appendix D, each work event was assigned work category and cause category designations as follows:

Work category		Cause category	
RC	Repair correction	EQ	Equipment deficiency
ITC	Inspection, testing, calibration	OP	Operating practice/requirements
NSM	Nuclear station modification	Reg	Regulatory
OM	Operational maintenance		
PM	Preventive maintenance		

Using the data given in these work event tables and the methodology described in Appendix C, limiting factors for operation and for maintenance were calculated for each system. The results of these calculations are given in Tables 4-1 and 4-2 for operation and maintenance, respectively. In these tables, the systems are ranked by the average limiting factor numbers.

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
1 REACTOR COOLANT SYSTEM											
1A REACTOR AND INTERNALS											
Reactor	← No Data →										
Internals	NA	B&W	Gray Book	4/18/76	Removed specimen holder tubes	RC	EQ	6	1034	0	Yes
1B FUEL AND RODS											
Fuel	← No Data →										
Rods	← No Data →										
1C REACTOR COOLANT PUMPS											
Pumps	1RCP-1A1	Westinghouse	Duke	12/24/74	RC pump seal maintenance	OM	EQ	4	169	0	No
			RADCAS	1/5/75	Replaced seal assembly	RC	EQ	4	60	0	No
			04589	1/14/75	Replaced gaskets, seal, ring	RC	EQ	4	60	0	No
			00342	2/29/75	Replaced leaking seals	RC	EQ	4	60	0	No
			22080	9/22/77	Replaced No. 3 seal	RC	EQ	4	6	0	No
	1RCP-1A2	Westinghouse	Duke	12/24/74	RC pump seal maintenance	OM	EQ	4	169	0	No
			09727A	4/5/76	Balanced pump - high vibration	RC	EQ	2	4	0	Yes
			22081	9/22/77	Removed and inspected seals	PM	EQ	12	32	0	No
	1RCP-1B1	Westinghouse	Duke	12/24/74	RC pump seal maintenance	OM	EQ	4	169	0	No
			22082	9/21/77	Removed and inspected seals	PM	EQ	12	32	0	No
			EE1-39	10/5/77	Replaced seals	RC	EQ	4	101	0	Yes
	1RCP-1B2	Westinghouse	Duke	6/3/74	Sealed coupling leak	RC	EQ	2	2	0	No
			04552	11/18/74	Lap seals	RC	EQ	2	16	0	No
			Duke	12/24/74	RC pump seal maintenance	OM	EQ	4	169	0	No
			Duke	1/14/75	Replaced seal package	OM	EQ	4	60	0	No
			022083	9/21/77	Removed and inspected seals	PM	EQ	12	32	0	No
			EEI-39	10/5/77	Inspected seals	RC	EQ	4	81	0	Yes
1D REACTOR COOLANT PUMP MOTORS											
	1RCPM-1A1	Westinghouse	50162	7/25/74	Changed out oil lift sys filter	PM	EQ	10	4	0	No
			01793	9/4/74	Cleaned oil pots and coolers	PM	EQ	4	33	0	No
			01857	9/16/74	Changed oil	PM	EQ	5	78	0	Yes
			50377	10/18/74	Installed new filter	PM	EQ	2	1	0	No
			05662	1/21/75	Cleaned oil pots, repaired brgs	RC	EQ	3	8	0	Yes
			50002A	2/14/75	Installed vibration pickup	RC	EQ	2	4	0	No
			00492A	3/9/75	Cleaned upper oil pot vent	PM	EQ	2	2	0	No
			00452A	3/21/75	Obtained oil sample	PM	EQ	2	2	0	Yes
			01831A	5/16/75	Obtained oil sample	PM	EQ	2	1	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work cate-gory	Cause cate-gory	No. men	Clock hours	Actual plant power,	Did event force or extend outage?
			Duke	7/11/75	Installed spare cooler, changed oil	PM	EQ	4	28	0	Yes
			02710A	7/11/75	Cleaned spare oil coolers	RC	EQ	3	4	0	No
			02725A	7/17/75	Collected oil sample	PM	EQ	2	2	0	No
			05741A	12/6/75	Changed oil in upper oil pot	RC	EQ	18	8	0	Yes
			50393	12/3/75	Collected oil sample	RC	EQ	2	2	0	No
			EEI-023	8/9/76	Repaired oil lift system	RC	EQ	?	112	73	Yes
			80893A	12/15/76	Replaced bolt in cover	RC	EQ	2	4	0	No
			Duke	3/1/77	Changed oil in motor	PM	EQ	2	5	0	No
			53470	8/21/77	PM oil coolers	PM	EQ	4	3	0	No
			22086	8/26/77	Cleaned oil coolers	PM	EQ	4	9	0	No
			53472	8/26/77	PM ac/dc oil lift pump motors	PM	EQ	4	5	0	No
			95881	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	6	22	0	No
			95897	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	4	19	0	No
			95518	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	4	20	0	No
			53385	9/8/77	Replaced bolts in flow chamber	PM	EQ	4	12	0	No
			22087	8/21/77	PM — motor	PM	EQ	4	79	0	No
			53580	9/21/77	Relocated oil fill/drain lines	NSM	EQ	3	6	0	No
	1RCPM-1A2	Westinghouse	50162	7/25/74	Changed out oil lift sys filter	PM	EQ	10	4	0	No
			01793	9/4/74	Cleaned oil pots, coolers	PM	EQ	4	33	0	No
			01857	9/16/74	Changed oil, PM	PM	EQ	5	78	0	Yes
			50377	10/18/74	Installed new filter	PM	EQ	2	1	0	No
			05662	1/21/75	Cleaned oil pots, repaired brgs	RC	EQ	3	8	0	Yes
			00211A	2/24/75	Balanced pump motor	RC	EQ	2	8	0	No
			00492A	3/9/75	Cleaned upper oil pot vent	PM	EQ	2	2	0	No
			00670A	3/12/75	Checked motor stand vibration	RC	EQ	1	7	0	No
			00452A	3/21/75	Obtained oil sample	PM	EQ	2	2	0	Yes
			01831A	5/16/75	Obtained oil sample	PM	EQ	2	1	0	No
			Duke	7/11/75	Installed spare cooler, changed oil	PM	EQ	4	28	0	Yes
			02710A	7/11/75	Cleaned spare oil coolers	RC	EQ	3	4	0	No
			02725A	7/17/75	Collected oil sample	PM	EQ	2	2	0	No
			03398A	7/18/75	Checked erratic upper brg temp	RC	EQ	2	2	0	Yes
			05741A	12/6/75	Changed oil in upper oil pots	RC	EQ	18	8	0	Yes
			50393	12/3/75	Collected oil sample	PM	EQ	2	2	0	No
			18387	11/9/76	Checked motor stand vibration	RC	EQ	2	3	0	No
			21153	4/2/77	Repaired oil lift line	RC	EQ	3	6	0	No
			95518	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	4	20	0	No
			95881	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	6	22	0	No
			95897	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	4	19	0	No
			53385	9/8/77	Replaced bolts in flow chamber	RC	EQ	4	48	0	No
			22086	8/26/77	Cleaned oil coolers	PM	EQ	4	36	0	No
			53470	8/21/77	PM — air coolers	PM	EQ	4	3	0	No
			53472	8/26/77	PM ac/dc oil lift pump motors	PM	EQ	4	5	0	No
			22088	9/21/77	PM — RC pump motor	PM	EQ	4	79	0	No
			53580	9/21/77	Relocated oil fill/drain lines	NSM	EQ	3	6	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work cate-gory	Cause cate-gory	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	1RCPM-1B1	Westinghouse	50162	7/25/74	Changed out oil lift sys filter	PM	EQ	10	4	0	No
			01717	8/2/74	Replaced vibration alarm light	RC	EQ	2	4	0	No
			01793	9/4/74	Cleaned oil pots, coolers	PM	EQ	4	33	0	No
			01857	9/16/74	Changed oil, PM	PM	EQ	5	78	0	Yes
			50377	10/18/74	Installed new filter	PM	EQ	2	1	0	No
			50662	1/21/75	Cleaned oil pots, repaired brgs	RC	EQ	3	8	0	Yes
			00492A	3/9/75	Cleaned upper oil pot vent	PM	EQ	2	2	0	No
			00452A	3/21/75	Obtained oil sample	PM	EQ	2	2	0	Yes
			01831A	5/16/75	Obtained oil sample	PM	EQ	2	2	0	No
			Duke	7/11/75	Installed spare cooler, changed oil	PM	EQ	4	28	0	Yes
			02710A	7/11/75	Cleaned spare oil coolers	RC	EQ	3	4	0	No
			02725A	7/17/75	Collected oil sample	PM	EQ	2	2	0	No
			05741A	12/6/75	Changed oil in upper oil pot	RC	EQ	18	8	0	Yes
			50393	12/3/75	Collected oil sample	RC	EQ	2	2	0	No
			53470	8/21/77	PM — air coolers	PM	EQ	4	3	0	No
			53472	8/26/77	PM ac/dc oil lift pump motors	RC	EQ	4	5	0	No
			22086	8/26/77	Cleaned oil coolers	PM	EQ	4	9	0	No
			53385	9/8/77	Replaced bolts in flow chamber	RC	EQ	4	48	0	No
			95881	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	6	22	0	No
			95518	9/8/77	Modified upper brg thrust rnrs	NSM	EQ	4	20	0	No
			53580	9/21/77	Relocated oil fill/drain lines	NSM	EQ	3	6	0	No
			22089	9/21/77	PM, RC pump motor	PM	EQ	4	79	0	No
			95897	9/8/77	Modified upper thrust runner	NSM	EQ	4	19	0	No
	1RCPM-1B2	Westinghouse	50162	7/25/74	Changed out oil lift sys fltr	PM	EQ	10	4	0	No
			01793	9/4/74	Cleaned oil pots, coolers	PM	EQ	4	33	0	No
			01857	9/4/74	Changed oil, PM	PM	EQ	5	78	0	Yes
			50377	10/18/74	Installed new filter	PM	EQ	2	1	0	No
			50616	12/3/74	Made oil temperature check	RC	EQ	2	4	0	No
			05539	12/26/74	Replaced speed indicator pickup	RC	EQ	2	4	0	No
			05662	1/21/75	Cleaned oil pots, repaired brgs	RC	EQ	3	8	0	Yes
			06451	2/5/75	Inspected thrust bearings	RC	EQ	2	8	0	No
			00031A	2/17/75	Checked for motor ground	RC	EQ	2	8	0	No
			00492A	3/9/75	Cleaned upper oil pot vent	RC	EQ	2	2	0	No
			00898A	3/4/75	Investigated lower brg oil leak	RC	EQ	2	8	0	Yes
			01831A	5/16/75	Obtained oil samples	RC	EQ	2	2	0	No
			Duke	7/11/75	Installed spare cooler, changed oil	PM	EQ	4	28	0	Yes
			02710A	7/11/75	Cleaned spare oil coolers	RC	EQ	3	4	0	No
			02725A	7/17/75	Collected oil sample	PM	EQ	2	2	0	No
			05741A	12/6/75	Changed oil in upper oil pot	RC	EQ	18	8	0	Yes
			05743A	12/5/75	Repaired lower oil pot leak	RC	EQ	?	24	0	Yes
			50393	12/3/75	Collected oil sample	RC	EQ	2	2	0	No
			19121A	12/20/76	Corrected frame vibration	RC	EQ	4	22	0	No
			Duke	3/1/77	RC pump lubrication test	PM	EQ	?	9	0	No
			53470	8/21/77	PM — air coolers	PM	EQ	4	3	0	No
			53472	8/26/77	PM ac/dc oil lift pump motors	PM	EQ	4	5	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
			22086	8/26/77	Cleaned oil coolers	PM	EQ	4	9	0	No
			53385	9/8/77	Replaced bolts in flow chambers	PM	EQ	4	12	0	No
			95881	9/8/77	Modified thrust runner	NSM	EQ	6	20	0	No
			95897	9/8/77	Modified thrust runner	NSM	EQ	4	19	0	No
			95518	9/8/77	MOdified thrust runner	NSM	EQ	4	20	0	No
			22090	9/21/77	PM, motor	PM	EQ	4	79	0	No
			53580	9/21/77	Relocated oil fill/drain lines	NSM	EQ	3	6	9	No
1E PIPING					No Data						
1F STEAM GENERATORS											
		B&W	Gray Book	11/7/75	Replaced instrumentation packing	RC	EQ	2	16	0	Yes
	1A	B&W	03457A	7/30/75	Furmanited FW header	RC	EQ	2	16	?	No
			EEI-036	10/31/76	Plugged OTSG tubes	RC	EQ	?	168	0	Yes
			18969	12/30/76	Replaced AFW nozzle gaskets	RC	EQ	2	4	0	No
	1B	B&W	03613	10/27/74	Removed FOAK inst.	RC	EQ	5	23	0	No
			03553A	7/19/75	Furmanited OTSG root valves	RC	EQ	2	48	0	Yes
			Gray Book	12/8/76	Plugged tubes	RC	EQ	?	105	0	Yes
			WA sheet	1/15/77	Plugged tubes	RC	EQ	?	114	0	Yes
			WA sheet	2/28/77	Plugged tubes	RC	EQ	?	162	0	Yes
			WA sheet	3/22/77	Plugged tubes	RC	EQ	?	111	0	Yes
			WA sheet	5/7/77	Plugged tubes	RC	EQ	?	180	0	Yes
	1A/1B	B&W	WA sheet	9/18/77	Plugged tubes	RC	EQ	?	483	0	Yes
1G PRESSURIZER											
Valves	RC-1	Rockwell	04310	11/13/74	Lapped seat and repacked	RC	EQ	2	8	0	No
			95878	8/25/77	Replaced valve	NSM	EQ	4	46	0	No
			RADCAS	4/14/77	Repacked	RC	EQ	?	12	0	No
	RC-2	Rockwell	EEI 77-52	12/20/77	Repacked	RC	EQ	2	8	0	Yes
	RC-3		22718	5/8/77	Valve stuck	RC	EQ	2	2	0	No
	RC-4	Dresser	10189	4/19/76	Valve stuck	RC	EQ	2	32	0	No
			10190	4/19/76	Valve won't open	RC	EQ	2	2	0	No
			10191	4/19/76	Valve won't open	RC	EQ	?	?	0	No
			17071	11/11/76	Flange leak repaired	RC	EQ	2	2	0	No
			95525	8/26/76	Machined	RC	EQ	2	4	0	No
	RC-5	Rockwell	06539	3/10/77	Repaired indicator light	RC	EQ	2	3	0	No
	RC-6	Rockwell			No Data						
	RC-7	Rockwell	Duke	8/29/74	Spurious operation	RC	EQ	2	2	0	No
			00241A	3/1/75	Replaced coil	RC	EQ	2	2	0	No
			06482A	12/2/75	Improper oper'n, cleaned and lubricated	RC	EQ	2	4	0	No
			22786	9/27/77	Replaced valve	RC	EQ	2	5	0	No
	RC-15	Velan			No Data						
	RC-16	Velan	24957	8/26/77	Repacked	RC	EQ	2	6	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	RC-66	Dresser	Duke	2/3/75	Lapped seat to stop leak	RC	EQ	2	12	0	No
			07361	1/9/76	Replaced limit box	RC	EQ	2	2	0	No
			18953A	1/19/77	Lapped seat to stop leak	RC	EQ	3	16	0	No
			24208	9/23/77	Repaired seat leak	RC	EQ	2	12	0	No
			53403	9/23/77	Lapped main, pilot valves	RC	EQ	3	16	0	No
	RC-67	Dresser	25861	9/26/77	Replaced valve	RC	EQ	2	8	0	No
	RC-68	Dresser	25860	9/26/77	Replaced valve	RC	EQ	4	6	0	No
Heaters	NA	B&W	13746A	9/15/76	Replaced contactor on htr bndl	RC	EQ	1	1	0	No
1H CORE PHYSICS											
1H1 Fuel Maneuvering		NA	EEI-034	12/12/75	Hold at 70% power	NA	EQ		17	70	Yes
		NA	WA sheet	1/26/77	Hold at 65% power	NA	EQ		4	65	Yes
		NA	WA sheet	3/14/77	Hold at 75% power	NA	EQ		5	75	Yes
		NA	WA sheet	4/4/77	Hold at 75% power	NA	EQ		6	75	Yes
		NA	WA sheet	5/24/77	Hold at 70% power	NA	EQ		10	70	Yes
1H2 Core Power Tilt		NA	EEI-010	5/31/76	No trouble found	NA	EQ		2	10	Yes
		NA	EEI-046	10/31/77	Hold at 75% power	NA	EQ		88	75	Yes
		NA	EEI-048	11/7/77	Hold at 75% power	NA	EQ		399	75	Yes
1H3 Xenon Hold		NA	Pwr Hist	8/25/74	Hold at 90% power	NA	REG		16	90	Yes
		NA	Pwr Hist	8/27/74	Hold at 90% power	NA	REG		24	90	Yes
		NA	Pwr Hist	3/25/75	Hold at 90% power	NA	REG		108	90	Yes
		NA	Pwr Hist	4/1/75	Hold at 90% power	NA	REG		36	90	Yes
		NA	Pwr Hist	4/11/75	Hold at 90% power	NA	REG		24	90	Yes
		NA	Pwr Hist	4/23/75	Hold at 90% power	NA	REG		30	90	Yes
		NA	Pwr Hist	5/19/75	Hold at 90% power	NA	REG		26	90	Yes
		NA	Pwr Hist	6/10/75	Hold at 90% power	NA	REG		40	90	Yes
		NA	EEI-015	7/4/75	Hold at 90% power	NA	REG		26	90	Yes
		NA	EEI-017	7/11/75	Hold at 85% power	NA	REG		3	85	Yes
		NA	EEI-020	7/20/75	Hold at 85% power	NA	REG		39	85	Yes
		NA	EEI-023	8/2/75	Hold at 90% power	NA	REG		36	90	Yes
		NA	EEI-026	8/9/75	Hold at 80% power	NA	REG		58	85	Yes
		NA	EEI-029	11/10/75	Hold at 90% power	NA	REG		21	90	Yes
		NA	EEI-032	12/8/75	Hold at 90% power	NA	REG		14	90	Yes
		NA	EEI-035	12/13/75	Hold at 90% power	NA	REG		19	90	Yes
		NA	EEI-001	1/10/76	Hold at 90% power	NA	REG		5	90	Yes
		NA	EEI-004	1/26/76	Hold at 90% power	NA	REG		23	90	Yes
		NA	EEI-006	4/13/76	Hold at 90% power	NA	REG		17	90	Yes
		NA	EEI-008	4/14/76	Hold at 90% power	NA	REG		86	90	Yes
		NA	EEI-011	6/1/76	Hold at 90% power	NA	REG		22	90	Yes
		NA	EEI-014	6/9/76	Hold at 90% power	NA	REG		22	90	Yes
		NA	EEI-016	6/22/76	Hold at 90% power	NA	REG		22	90	Yes
		NA	EEI-018	6/27/76	Hold at 90% power	NA	REG		20	90	Yes
		NA	EEI-020	7/8/76	Hold at 90% power	NA	REG		21	90	Yes

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
		NA	EEI-022	7/14/76	Hold at 90% power	NA	REG		17	90	Yes
		NA	EEI-025	8/16/76	Hold at 90% power	NA	REG		25	90	Yes
		NA	EEI-027	8/29/76	Hold at 90% power	NA	REG		4	90	Yes
		NA	EEI-029	9/1/76	Hold at 90% power	NA	REG		19	90	Yes
		NA	EEI-031	9/6/76	Hold at 90% power	NA	REG		2	90	Yes
		NA	EEI-033	10/10/76	Hold at 90% power	NA	REG		17	90	Yes
		NA	EEI-035	10/27/76	Hold at 90% power	NA	REG		22	90	Yes
		NA	EEI-037	11/16/76	Hold at 90% power	NA	REG		25	90	Yes
		NA	EEI-041	12/20/76	Hold at 90% power	NA	REG		13	90	Yes
		NA	EEI-003	1/27/77	Hold at 90% power	NA	REG		21	90	Yes
		NA	EEI-006	2/9/77	Hold at 90% power	NA	REG		23	90	Yes
		NA	EEI-010	3/14/77	Hold at 90% power	NA	REG		18	90	Yes
		NA	EEI-014	3/20/77	Hold at 90% power	NA	REG		15	90	Yes
		NA	EEI-019	4/4/77	Hold at 90% power	NA	REG		22	90	Yes
		NA	EEI-022	4/26/77	Hold at 90% power	NA	REG		4	90	Yes
		NA	WA sheet	4/27/77	Hold at 90% power	NA	REG		4	90	Yes
		NA	EEI-030	6/8/77	Hold at 90% power	NA	REG		17	90	Yes
		NA	EEI-049	11/24/77	Hold at 90% power	NA	REG		12	90	Yes
		NA	EEI-051	12/11/77	Hold at 90% power	NA	REG		21	90	Yes
		NA	EEI-054	12/28/77	Hold at 90% power	NA	REG		15	90	Yes
		NA	EEI-056	12/31/77	Hold at 90% power	NA	REG		19	90	Yes
1H4 Startup Physics Tests		NA	Pwr Hist	3/1/75	Startup Physics, Part 1	NA	REG		285	0	Yes
		NA	Pwr Hist	3/12/75	Startup Physics, Part 2	NA	REG		72	40	Yes
		NA	Pwr Hist	3/16/75	Startup Physics, Part 2	NA	REG		108	75	Yes
		NA	Pwr Hist	3/23/75	Startup Physics, Part 2	NA	REG		44	75/90	Yes
		NA	Pwr Hist	4/1/76	Startup Physics, Part 1	NA	REG		116	0	Yes
		NA	Pwr Hist	4/5/76	Startup Physics, Part 2	NA	REG		175	40/75	Yes
		NA	EEI-012	6/3/76	75% power escalation test	NA	REG		7	75	Yes
		NA	EEI-040	10/14/77	Startup Physics, Part 1	NA	REG		85	0	Yes
		NA	EEI-042	10/19/77	Startup Physics, Part 2	NA	REG		298	40	Yes

2 AUXILIARY FLUID SYSTEMS

2A MAKEUP AND PURIFICATION

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
Valves	1HP-1	Rockwell	04823A	9/18/75	Repaired oper closing circuit	RC	EQ	2	3	100	No
			Duke	?	Repacked three times				12		
	1HP-2	Rockwell	04140	11/3/74	Replaced fuse	RC	EQ	2	1	0	No
			Duke	?	Repacked one time				4		
	1HP-3	Rockwell	←		No Data						→
	1HP-4	Rockwell	←		No Data						→
	1HP-5	Rockwell	20659	2/25/77	Replaced solenoid coil	RC	EQ		4	98	No
	1HP-6	Rockwell	←		No Data						→

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	1HP-7	Leslie	23884	7/10/77	Tightened packing	RC	EQ		1	55	No
			02339	8/29/74	Repaired controller	RC	EQ	2	3	100	No
			02492	9/6/74	Replaced regulator	RC	EQ	2	$3\frac{1}{2}$	75	No
			03755	10/12/74	Adjusted air supply regulator	RC	EQ	2	1	80	No
			24664	8/6/77	Checked diaphragm, reset air supply	RC	EQ	4	$2\frac{1}{2}$	0	No
			Duke	?	Repacked four times						
	1HP-8	Aloyco	03061A	7/4/75	Repaired operator	RC	EQ	2	4	100	No
	1HP-9	Aloyco	03068A	7/3/75	Replaced limit switch	RC	EQ		4	100	No
	1HP-11	Aloyco	03068A	7/3/75	Replaced limit switch	RC	EQ	2	3	100	No
	1HP-14	Fisher	←		No Data						→
	1HP-15	BMCo	03371	10/9/74	Replaced controller unit	RC	EQ	2	3	55	No
			Duke	?	Repacked one time				4		
	1HP-16	Aloyco	←		No Data						→
	1HP-17	Aloyco	00570A	3/7/75	Adjusted limit switch	RC	EQ	2	2	0	No
			00486A	3/6/75	Disassembled, lapped seat	RC	EQ	2	8	0	No
	1HP-20	Rockwell	05608	12/27/74	Repaired positioner	RC	EQ	2	3	0	No
	1HP-21	Univalve	←		No Data						→
	1HP-24	Wm. Powell	←		No Data						→
	1HP-25	Wm. Powell	←		No Data						→
	1HP-26	Rock-Edw.	24993	8/24/77	Replaced packing	RC	EQ		8	0	No
			90197	10/25/74	Adjusted valve operator	RC	EQ		4	0	No
			90221	11/15/74	Replaced canopy ring	RC	EQ	2	10	0	No
			09197	11/15/74	Adjusted valve to close tighter	RC	EQ	2	2	0	No
			Duke	?	Repacked three times				12		
	1HP-27	Rock-Edw.	12090A	7/12/75	Repaired oper closing circuit	RC	EQ	2	2	100	No
			12072A	7/9/76	Replaced bent control on oper	RC	EQ	1	1	100	No
			Duke	?	Repacked four times				16		
			24994	8/26/77	Replaced packing with "1625"	RC	EQ		8	100	No
	1HP-31	Fisher	←		No Data						→
	1HP-43	Lonergan	24455	?	Replaced internal parts	RC	EQ		40	0	No
			Duke	?	Repacked two times				8		
	1HP-57	Velan	WA sheet	3/9/77	Replaced valve	RC	EQ	2	8	0	Yes
	1HP-60	Velan	Duke	3/25/77	Repaired reach rod	RC	EQ		6	0	No
	1HP-64	?	←		No Data						→
	1HP-71	?	22161	5/1/77	Lapped seat, replaced gasket	RC	EQ	2	6	100	No
			24107	8/10/77	Cleaned, new gasket on plug	RC	EQ		8	0	No
			22161	8/18/77	Lapped seat, repl spring, gasket	RC	EQ		40	0	No
			Duke	?	Repacked two times				8		
	1HP-79	Lonergan	20432	3/6/77	No description				6	0	No
			22418	9/2/77	Replaced spring, nozzle	RC	EQ		48	0	No
	1HP-98	Crane	22694	5/10/77	Repacked	RC	EQ		2		
	1HP-107	Crane	22097	5/10/77	Repacked	RC	EQ		4		
	1HP-118	Velan	EEI 77-24	5/10/77	Repacked	RC	EQ		2	0	No
	1HP-120	BMCo	11434	7/3/76	Fault in E/P controller	RC	EQ		1		
			05559	12/28/74	Repaired positioner on control	RC	EQ	2	2	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
			18591A	12/3/76	Replaced cotter pin on linkage	RC	EQ	1	1	0	No
			18744A	12/9/76	Lapped valve seat	RC	EQ		16	0	No
			24673	8/29/77	Lapped seat, replaced gasket	RC	EQ		12	0	No
			26230	9/30/77	Would not operate in automatic	RC	EQ		2	0	No
			Duke	?	Repacked four times				16		
	1HP-126	Velan	25922	9/19/77	Replaced springs, gasket, cap seats	RC	EQ		18	0	No
			24956	8/17/77	Cleaned seats, repacked	RC	EQ		16	0	No
			Duke	?	Repacked five times				20		
	1HP-127	Velan	25923	9/19/77	Replaced disc spring, gasket	RC	EQ		18	0	No
			Duke	9/8/77	Cleaned seats, replaced gaskets	RC	EQ		16	0	No
			Duke	?	Repacked one time				4		
	1HP-153	Velan	20399	3/6/77	Repaired body-to-bonnet leak	RC	EQ		4	0	No
	1HP-154	Velan	←		No Data						→
	1HP-249	Velan	EEI 77-24	5/10/77	Repacked one time	RC	EQ		4	0	No
Pump Motors	1HP-P1A	Ing-Rand	24813	9/2/77	Changed oil in motors, bearings	PM	EQ		5	0	No
	1HP-P1B	Ing-Rand	09333A	3/28/76	Swapped "B" motor to "C"	RC	EQ		16	0	No
			53051	5/30/77	Retorqued motor to stand	RC	EQ	3	2	87	No
			24813	9/2/77	Changed oil in motors, bearings	PM	EQ		5	0	No
			25946	9/26/77	Cleaned sight glass	RC	EQ		4	0	No
	1HP-P1C	Ing-Rand	08994A	4/8/76	Repaired, rebuilt "B" mtr for "C"	RC	EQ		40	70	No
			24813	9/2/77	Changed oil in motors, bearings	PM	EQ		5	0	No
			25947	9/26/77	Cleaned sight glass	RC	EQ		4	0	No
Pumps	1HP-P1A	Ing-Rand	25857	9/20/77	Replaced orifice	RC	EQ	2	4	0	No
	1HP-P1B	Ing-Rand	06206A	11/18/75	Tightened inlet ftg to reduce vibr	RC	EQ		2	98	No
			22703	5/23/77	Disassembly, insp'n, and repair vibration	RC	EQ	2	32	0	No
			53051	6/18/77	Removed, repaired, replaced pump	RC	EQ	4	32	100	No
			52615	9/12/77	Inspected seals	ITC	EQ		135	0	No
			52825	9/12/77	Replaced damaged pipe	RC	EQ	2	35	0	No
	1HP-P1C	Ing-Rand	52616	9/13/77	Pump seal flow test	ITC	EQ		117	0	No
Letdown Orifice	←				No Data						→
Letdown Coolers	1HPX-1A/B	Graham	10819A	9/6/77	Installed new cooler	NSM	EQ		970	0	No
Letdown Filters	1HP-F2A	Filtrite	17970A	11/17/76	Checked vent hose, seal	RC	EQ		2	0	No
			WA sheet	3/3/77	Cleaned RC pump seal ret filter	PM	EQ		2	0	No
			10700A	5/28/77	Replaced gasket in "A" filter	RC	EQ		12	0	No
	1HP-F2B	Filtrite	18383	11/29/76	Changed filter	RC	EQ		12	0	No
Seal Return Coolers	1HPC-1A/ -1B	←			No Data						→

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
Letdown Storage Tank	1HP-T1	← No Data →									
Purification Demineralizer	1HP-X1/ -X2	← No Data →									
RC Pump Seal Injection Filters	1HP-3A/ -3B	← No Data →									
2B DECAY HEAT/LP INJECTION											
Valves	LP-1	Walworth	09142	3/27/76	Replaced manual brk rel button	RC	EQ	2	2	0	No
	LP-2	Walworth	RADCAS	1/4/75	Replaced microswitch	RC	EQ	2	2	65	No
	LP-3	Wm. Powell	24118	8/9/77	Repacked valve	RC	EQ	2	2	0	No
	LP-9	Crane	24126	9/26/77	Repacked valve	RC	EQ	2	1	0	No
	LP-10	Crane	24127	9/5/77	Repacked valve	RC	EQ	2	2	0	No
	LP-11	Crane	24123	8/21/77	Repacked valve	RC	EQ	2	2	0	No
	LP-12	Crane	'77 Ref.	8/?/77	Replaced valve	RC	EQ	2	6	0	No
	LP-13	Crane	24123	8/21/77	Replaced valve	RC	EQ	2	6	0	No
	LP-14	Crane	'77 Ref.	8/?/77	Replaced valve	RC	EQ	2	3	0	No
	LP-15	Wm. Powell	17175	12/31/76	Set "open" limit switch	RC	EQ	3	1	96	No
	LP-16	← No Data →									
	LP-17	Walworth	06322A	2/27/75	Tightened packing	RC	EQ	2	1	0	No
			08156A	11/14/75	Adjusted torque switch	RC	EQ	2	1	96	No
	LP-18	Walworth	RADCAS	5/7/75	Cleaned corroded contacts	RC	EQ	2	½	98	No
			06128A	11/13/75	Electr operator not energized	RC	EQ	2	2	100	No
			08156A	11/14/75	Adjusted torque switch	RC	EQ	2	1	96	No
			Duke	2/27/75	Tightened packing	RC	EQ	2	1	0	No
			04996	8/25/77	Repacked	RC	EQ	2	4	0	No
	LP-19	Rockwell	025407	9/2/77	Investigated valve leak	RC	EQ	3	2	0	No
	LP-21	Rockwell	05479A	11/8/75	Tightened bolts	RC	EQ	2	8	30	No
	LP-22	Rockwell	04736	12/10/74	Rewound motor	RC	EQ	2	8	0	No
			05375	1/2/75	Operator stuck	RC	EQ	2	4	0	No
	LP-35	Chapman	06378	2/2/75	Repacked	RC	EQ	2	2	0	No
			03592	7/30/75	Installed new gasket	RC	EQ	2	4	0	No
	LP-45	Velan	Duke	2/28/76	Repacked valve	RC	EQ	2	4	0	No
	LP-79	Velan	RADCAS	1/2/75	Replaced valve	RC	EQ	2	6	0	No
	LP-94	Aloyco	18795	8/16/77	Repacked valve	RC	EQ	2	2	0	No
			24128	8/21/77	Repacked valve	RC	EQ	2	2	0	No
Pumps	LP-1A	Ing-Rand	← No Data →								
	LP-1B	Ing-Rand	← No Data →								
	LP-1C	Ing-Rand	0678LA	12/29/75	Replaced corroded shaft seals	RC	EQ	3	8	96	No
			10413A	5/4/76	Replaced pressure gage on pump	RC	EQ	2	8	0	No
			12487A	8/13/76	Replaced pressure gage on pump	RC	EQ	2	7	75	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
LPI Coolers	LP-C1A				No Data						
	LP-C1B				No Data						
BWST	LP-T1	PDM	LER	10/7/75	Aligned valves	RC	EQ	2	4	100	No
			10656A	?	Calibrated transmitter on level indicator	RC	EQ		6	0	No
2C CHEMICAL ADDITION AND BORON SAMPLING											
Pumps	CA-P3	LAPP	80137	8/21/74	Repaired elect insul'n breakdown	RC	EQ	2	3	100	No
	CA-P4	LAPP	80132	8/15/74	Repaired oil leak into pump	RC	EQ	2	6	100	No
			80144	8/26/74	Replaced diaphragm	RC	EQ	2	3	0	No
			80380	12/2/75	Replaced pump valve	RC	EQ	3	8	100	No
			80395	12/15/75	Replaced diaphragm	RC	EQ	2	2	96	No
			80444	1/11/76	Replaced diaphragm	RC	EQ	2	4	96	No
			80428	1/14/76	Replaced fittings	RC	EQ	2	4	96	No
			80450	1/19/76	Repaired oil leak	RC	EQ	2	4	96	No
			80460	1/20/76	Disassembled, cleaned pump	RC	EQ	2	4	96	No
			80716	9/9/76	Replaced diaphragm	RC	EQ	2	8	98	No
			81123	5/4/77	Repaired suction line	RC	EQ	2	4	96	No
			81260	8/29/77	Repaired oil leak	RC	EQ	2	1	0	No
	CA-P5	LAPP	02126	5/26/77	Adjusted relief valve	RC	EQ	2	2	100	No
			02571	6/11/77	Replaced diaphragm	RC	EQ	2	8	90	No
			02637	6/18/77	Motor won't start/valve closed	RC	EQ	3	8	100	No
			12682	8/7/76	Replaced check valve	RC	EQ	2	8	98	No
			16689	9/29/76	Repaired pump suction line	RC	EQ	2	4	96	No
			23728	7/4/77	Replaced gasket	RC	EQ	2	10	10	No
			23745	7/6/77	Replaced diaphragm, overhauled	RC	EQ	2	8	55	No
Tanks	WD-T22		18373	11/26/77	Adjusted relief valve	RC	EQ	1	2	100	No
	CA-T2		26245	10/2/77	Replaced shaft, coupling, pack'g	RC	EQ	3	7	100	No
Agitator	CA-M2		06406	2/15/75	Replaced broken shaft	RC	EQ	2	4	0	No
			18126	11/14/76	Replaced propeller	RC	EQ	3	4	100	No
Coolers	CA-C1, CA-C2A, CA-C2B				No Data						
2D SPENT FUEL COOLING											
Pumps	SF-P1A				No Data						
	SF-P1B	Ing-Rand	19183A	12/29/76	Repaired leaking sight glass	RC	EQ	2	2	96	No
	SF-P2				No Data						
Coolers	SF-C1A, SF-C1B				No Data						

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
Demineralizers	SF-X1				No Data						
Filters	SF-F1A				No Data						
	SF-F1B				No Data						
Valves	SF-19				No Data						
	SF-20				No Data						
	SF-33				No Data						
	SF-36				No Data						
	SF-61	Crane	24968	9/14/77	Replaced gaskets, repacked	RC	EQ	2	12	0	No
	SF-69				No Data						
2E REACTOR BUILDING SPRAY											
Pumps	BS-3A	Ing-Rand	RADCAS	9/22/75	Replaced gasket	RC	EQ	2	4	100	No
			24110		Repaired seal supply leak	RC	EQ	2	8	0	No
	BS-3B	Ing-Rand	21107	9/22/77	Inspected motor	RC	EQ	2	4	0	No
Valves	BS-1				No Data						
	BS-2	Aloyco	RADCAS	8/4/75	Replaced fuse	RC	EQ	1	1	100	No
	BS-3	Crane	RADCAS	3/14/75	Loosened packing gland	RC	EQ	2	2	40	No
	BS-13	Crane	24117	8/18/77	Repacked	RC	EQ	2	2	0	No
2F CORE FLOODING											
Valves	ICF-1	Walworth	05799	1/3/75	Temporary jumper-torque switch	RC	EQ	2	2	0	No
			50712	2/20/76	Replaced capacitor, tested relays	RC	EQ	2	2	0	No
	ICF-2	Walworth	17832A	9/5/77	Repacked	RC	EQ	2	5	0	No
	ICF-3				No Data						
	ICF-4	Rockwell	04958A	10/3/75	Adjusted open torque switch	RC	EQ	2	2	98	No
	ICF-5				No Data						
	ICF-6				No Data						
	ICF-10				No Data						
	ICF-15				No Data						
	ICF-17				No Data						
Flow Transmitter			09975A	4/13/76	Adjusted setpoint	RC	EQ		4	90	No
			12177A	7/19/76	Calibrated signal monitor alarm	ITC	EQ		4	100	No
Tanks					No Data						
2G LOW-PRESSURE SERVICE WATER											
Valves	LPSW-4	Walworth	26179	8/26/77	Stroke time corrected	RC	EQ	2	4	0	No
	LPSW-5	Walworth	91036	9/28/77	Adjusted limit switch	RC	EQ	2	2	0	No
	LPSW-6				No Data						
	LPSW-15	Walworth	90860	8/9/77	Set limits on operator	RC	EQ	3	10	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	LPSW-16 LPSW-18 LPSW-19 LPSW-21 LPSW-22 LPSW-24 LPSW-51	←			No Data						→
	LPSW-79	Crane	25405	9/6/77	Installed chain-oper handwheel	RC	EQ	2	8	0	No
	LPSW-356	←			No Data						→
Pumps/Motors	LPSW-P1A	←			No Data						→
	LPSW-P1B		21099	9/12/77	Inspected motor	RC	EQ	1	2	0	No
2H DEMINERALIZED WATER SYSTEM		←			No Data						→
2I COMPONENT COOLING SYSTEM											
Valves	CC-1	Walworth	04802	9/25/75	Adjusted limit switch	RC	EQ	2	4	100	No
			09996A	4/14/76	Adjusted limit switch	RC	EQ	2	6	0	No
	CC-7	Walworth	90862	8/31/77	Replaced gasket, repacked	RC	EQ	3	6	0	No
	CC-8	Wm. Powell	25920	9/21/77	Performed leak test	RC	EQ	2	4	0	No
			90861	9/7/77	Repaired seat leak, replaced gasket	RC	EQ	2	45	0	No
	CC-24	Crane	91065	8/29/77	Repaired seat leak, replaced gasket	RC	EQ	3	24	0	No
	CC-76	Crane	25041	9/6/77	Rpacked, changed gasket	RC	EQ	2	30	0	No
Coolers	CC-C1	Atlas	24422	9/9/77	Cleaned tubes w/air, water	RC	EQ	2	6	0	No
	CC-C2	Atlas	24422	9/9/77	Cleaned tubes w/air, water	RC	EQ	2	6	0	No
Pumps	ICC-P1A ICC-P1B	←			No Data						→
Pump Motor	ICC-P1A	Ing-Rand	04738	12/4/74	Replaced motor	RC	EQ	2	4	0	No
Pressure Switch			EEI-022	8/2/75	Repaired switch	RC	EQ	2	7	60-90	No
Filters	CC-F1 CC-F2	←			No Data						→
2J PENETRATION ROOM VENTILATION SYSTEM											
Valves	PR-2	Pratt	015530	8/10/76	Repaired valve	RC	EQ		10	70	No
			17804	11/1/76	Repaired hole in diaphragm	RC	EQ	2	2	0	No
			17830	11/3/76	Repaired blown diaphragm	RC	EQ	2	3	0	No
			21384	3/26/77	Replaced diaphragm	RC	EQ		7	0	No
			24660	8/5/77	Replaced diaphragm	RC	EQ	1	2	0	No
			24663	8/6/77	Replaced diaphragm	RC	EQ	2	3	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power,	Did event force or extend outage?
			25412	9/7/77	Repaired seat leak	RC	EQ		9	0	No
			19838	12/9/77	Replaced diaphragm	RC	EQ	2	5	100	No
			20831	3/1/77	Checked valve, found OK	RC	EQ	2	1	0	No
	PR-7	Grinnell	09997A	4/15/76	Replaced diaphragm, limit switch	RC	EQ	2	18	0	No
3 SECONDARY SYSTEMS											
3A MAIN TURBINE											
LP Turbine		Gen Electric	EEI-007	4/14/76	Checked vibr'n on No. 3 bearing	RC	EQ	4	17	75	Yes
			21110	9/12/77	Repaired diaphragm cracks	RC	EQ	10	210	0	No
HP and LP Turbine		Gen Electric	DPC	9/12/77	Overhauled turbine	RC	EQ	?	3241	0	No
Steam Seals	←				No Data						→
3B MAIN STEAM SYSTEM											
Valves	MS-1	Crosby	←		No Data						→
	MS-2	Crosby	07366	2/27/76	Lap disk and seats	RC	EQ	4	6	0	No
	MS-3	Crosby									
	MS-4	Crosby									
	MS-5	Crosby									
	MS-6	Crosby	←		No Data						→
	MS-7	Crosby									
	MS-8	Crosby									
	MS-9	Crosby	07366	2/27/76	Lap disk and seats	RC	EQ	4	6	0	No
	MS-10	Crosby	07366	2/27/76	Lap disk and seats	RC	EQ	4	6	0	No
	MS-11	Crosby	07366	2/27/76	Lap disk and seats	RC	EQ	4	6	0	No
	MS-19	Crane	25001	8/27/77	Tightened screw in stem	RC	EQ	2	4	0	No
	MS-22	Crane	←		No Data						→
	MS-24	Crane	18586	3/6/77	Repaired	RC	EQ	5	2	0	No
		Crane	23015	8/18/77	B/B leak, lapped seat, welded hole	RC	EQ	8	7	0	No
	MS-26	Crane	26227	9/29/77	Repaired	RC	EQ	2	4	0	No
	MS-28	Atw-Morr.	23186	8/27/77	Repacked	RD	EQ	2	16	0	No
	MS-31	←			No Data						→
	MS-39	Crane	3055A	8/11/75	Replaced B/B gasket, Furmanited	RC	EQ	2	16	90	No
	MS-42	Crane	4819A	2/15/76	Replaced gaskets, lapped seat	RC	EQ	2	8	0	No
			24175	8/24/77	Replaced gaskets	RC	EQ	2	2	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	MS-47	Crane	20010	8/16/77	Repacked	RC	EQ	2	4	0	No
	MS-58	Velan	22888	8/15/77	B/B leak, replaced gasket	RC	EQ	2	12	0	No
	MS-78	Edward	1670A	6/3/75	Furmanited	RC	EQ	2	4	100	No
			23923	8/16/77	Repacked	RC	EQ	2	4	0	No
	MS-79	Crane	23946	8/16/77	Repacked	RC	EQ	2	4	0	No
			10825	9/22/77	Dressed wedge, seats; repacked	RC	EQ	4	9	0	No
	MS-80	Edward	23945	8/16/77	Repacked	RC	EQ	2	4	0	No
	MS-81	Edward	27406	11/30/77	Furmanited B/B leak	RC	EQ	2	4	100	No
	MS-88	Velan	EEI 77-24	5/10/77	Repacked	RC	EQ	2	2	0	No
	MS-90	Crane	04050A	8/24/75	B/B leak furmanited	RC	EQ	2	4	100	No
	MS-92	Crane	03501A	7/29/75	Replaced gasket, reseated	RC	EQ	2	10	0	No
			19351A	8/16/77	Relapped seats, replaced gaskets	RC	EQ	3	16	0	No
	MS-93	Fisher	16993A	8/19/77	Cleaned, replaced seats, gasket	RC	EQ	2	5	0	No
	MS-94	?	24417	9/8/77	Reseated, replaced bearing	RC	EQ	2	16	0	No
	MS-96	Crane	24420	9/2/77	Improper operation	RC	EQ	2	8	0	No
	MS-97				No Data						
	MS-102	Gen Electric									
	MS-103	Gen Electric	23008	8/14/77	Replaced gland bolts, repacked	RC	EQ	2	8	0	No
	MS-104	Gen Electric			No Data						
	MS-105	Gen Electric									
	MS-106	Gen Electric	06367A	3/1/76	Disassembled, cleaned, insp'd, honed	RC	EQ	5	21	0	No
	MS-107	Gen Electric	06367A	3/1/76	Disassembled, cleaned, insp'd, honed	RC	EQ	5	21	0	No
	MS-108	Gen Electric	06367A	3/1/76	Disassembled, cleaned, insp'd, honed	RC	EQ	5	21	0	No
	MS-109	Gen Electric	06367A	3/1/76	Disassembled, cleaned, insp'd, honed	RC	EQ	5	21	0	No
	MS-115	Gen Electric			No Data						
	MS-116	Gen Electric									
	MS-119	Gen Electric	21111	8/25/77	Disassembled, cleaned, inspected	RC	EQ	6	30	0	No
	MS-120	Gen Electric	21111	8/25/77	Disassembled, cleaned, inspected	RC	EQ	6	30	0	No
	MS-123	Gen Electric	23691	8/29/77	Disassembled, cleaned, inspected	RC	EQ	5	47	0	No
	MS-124	Gen Electric			No Data						
	MS-139	Crane	23181	8/23/77	Disassembled, cleaned, inspected	RC	EQ	4	12	0	No
			24419	9/2/77	Repaired bonnet alignment pin	RC	EQ	4	8	0	No
	MS-140	Lonergan	24416	8/31/77	Adjusted guide pin, lapped seat	RC	EQ	4	6	0	No
			23812	8/18/77	Replaced seat, disk	RC	EQ	6	8	0	No
	MS-147	Velan	03445	8/1/75	Furmanited leak	RC	EQ	2	4	0	No
			5263A	3/15/76	Repaired	RC	EQ	2	8	0	No
Valve Test			EEI-024	8/8/75	Unit tripped during test	ITC	EQ	1	8	0	Yea
			WA sheet	3/19/77	MSSV test	ITC	EQ	1	4	55	Yes
Piping	Turb BP		24401	8/22/77	MT insp'n, rewelded headers	ITC	EQ	6 5	35 8	0	No
Sys-53	3-19EA		53700	9/26/77	Rewelded for PT	ITC	EQ	4	4	0	No
	AMS Lead		53699	9/26/77	Repaired weld	RC	EQ	2	4	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
Sys-01	2-X15BA		53960	9/27/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01	2-18BA		53973	9/27/77	Repaired weld	RC	EQ	2	4	0	No
Sys-53B	3-14FA		53974	9/27/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01	2-X2BF		53958	9/27/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01	1-X7A		53975	9/27/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01	2-X16A		53956	9/27/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01	2-X19B		53975	9/28/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01A	2-4BA		53961	9/29/77	Repaired defects	RC	EQ	9	6	0	No
Sys-01	1-X17A		53959	9/29/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01A	2-2BB		53970	9/29/77	Repaired defects	RC	EQ	9	6	0	No
Sys-01	1-X24		53955	9/29/77	Repaired weld	RC	EQ	2	4	0	No
Sys-01	1-X6A		53954	9/29/77	Repaired weld	RC	EQ	2	4	0	No
Sys-53	3-9AA		53694	9/29/77	Repaired weld	RC	EQ	2	4	0	No
	N/A		22122	9/30/77	Removed, replaced insulation	RC	EQ	3	60	0	No
3C FEEDWATER SYSTEM											
Valves	FDW-6	Crane	01128A	4/3/75	Repaired B/B leak	RC	EQ	4	4	65	No
	FDW-8	Sch-Koert	01608	10/30/74	Tightened flange	RC	EQ	2	18	0	No
			24176	8/16/77	Repacked	RC	EQ	4	6	0	No
	FDW-14	Crane	21573	8/25/77	Stuck on seat, cycled valve	RC	EQ	1	4	0	No
	FDW-16	Atw.-Morr.	03281	7/28/75	Repaired pin hole in body	RC	EQ	2	4	0	No
	FDW-21	Atw.-Morr.	05856	11/11/75	Furmanited leaky packing	RC	EQ	2	4	100	No
	FDW-23	Velan	23271	8/24/77	Replaced valve	RC	EQ	2	6	0	No
	FDW-31	Crane	25043	9/11/77	Limits out of adjustment	RC	EQ	1	8	0	No
	FDW-33	← No Data →									
	FDW-36	Crane	18790A	12/16/76	Trip on high delta-P	RC	EQ	2	3	0	No
			19835	2/8/77	Repaired motor	RC	EQ	2	2	0	No
			09335	3/9/76	Repaired actuator	RC	EQ	2	6	0	No
	FDW-37	Crane	20914	3/15/77	Hinge pin leak at gasket	RC	EQ	2	6	90	No
	FDW-40 FDW-42	← No Data →									
	FDW-45	Crane	00216	6/6/74	Adjusted limit switch	RC	EQ	1	2	0	No
			03761	7/30/75	Adjusted torque switch	RC	EQ	1	1	0	No
	FDW-46	Crane	00876A	3/18/75	Tightened packing	RC	EQ	2	2	75	No
			00458A	3/21/75	Furmanited hinge pin leak	RC	EQ	2	4	75	No
	FDW-47	Crane	03761	7/30/75	Replaced torque switch	RC	EQ	1	1	0	No
	FDW-48	Crane	22206	8/5/77	Tightened flange, bolts	RC	EQ	2	2	0	No
	FDW-51	Crane	01129A	1/4/75	Furmanited leak	RC	EQ	3	4	0	No
			03510	8/11/75	Furmanited leak	RC	EQ	3	16	90	No
			25423	9/7/77	Replaced hand wheel	RC	EQ	2	4	0	No
	FDW-53	Fisher	01864	10/3/74	Disassembled, inspected	RC	EQ	3	5	90	No
			00322	6/7/74	Repaired B/B leak	RC	EQ	2	4	0	No
			00946	6/26/74	Repaired body crack	RC	EQ	2	2	0	No
			00718	7/1/74	B/B leak repaired	RC	EQ	3	8	0	No
			24423	9/7/77	Inspected, repaired seat	RC	EQ	2	5	0	No
			25851	9/19/77	Replaced gaskets	RC	EQ	4	12	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	FDW-54	Crane	17086	8/24/77	Cleaned seats, seal rings	RC	EQ	4	12	0	No
			24423	9/7/77	Inspected, repaired seats	RC	EQ	2	4	0	No
			25851	9/20/77	Replaced gaskets	RC	EQ	4	12	0	No
	FDW-58	Velan	12813	11/25/76	Repaired leak, repacked replacement gasket	RC	EQ	2	3	0	No
	FDW-60	Velan	27409	12/7/77	Furmanited B/B leak	RC	EQ	3	4	100	No
	FDW-65	Fisher	00152	6/3/74	Replaced gasket — B/B leak	RC	EQ	2	4	0	No
			01868	10/30/74	Dissambled, inspected, reassembled	RC	EQ	3	5	0	No
			24423	9/7/77	Inspected seats	RC	EQ	2	2	0	No
			25851	9/20/77	Replaced B/B gaskets	RC	EQ	4	12	0	No
	FDW-66	Crane	00960	9/4/74	B/B leak repaired	RC	EQ	2	2	100	No
			18582	12/1/76	Valve removed to inspect	RC	EQ	2	2	0	No
			24423	9/7/77	Repaired valve seats	RC	EQ	4	10	0	No
			25851	9/20/77	Replaced gaskets	RC	EQ	2	12	0	No
			17086	9/24/77	Cleaned seats, seal rings	RC	EQ	2	4	0	No
	FDW-74	Crane	24438	9/30/77	Replaced seat rings, gaskets	RC	EQ	3	49	0	No
	FDW-75	Crane	24439	9/17/77	Inspected, cleaned seats, replaced ring	RC	EQ	3	20	0	No
	FDW-76	Crane	24440	9/17/77	Inspected, cleaned seats, repacked	RC	EQ	3	20	0	No
	FDW-77	Crane	24443	9/17/77	Inspected, cleaned seats, repacked	RC	EQ	3	20	0	No
	FDW-82	Fisher	24444	9/17/77	Replaced stem, plug, gasket; repacked	RC	EQ	3	20	0	No
	FDW-84	Cran-Chap	00960	7/3/74	Body/bonnet leak	RC	EQ	3	4	100	No
			01098	7/20/74	Won't open w/switch in "open"	RC	EQ	1	4	100	No
			01016	11/4/74	Won't operate, B/B leak	RC	EQ	2	14	100	No
			04576	12/24/74	Replaced gear in operator	RC	EQ	3	11	0	No
			01140A	3/30/75	Repaired B/B leak, Furmanited	RC	EQ	3	4	60	No
			02405A	7/26/75	Won't operate, repaired gear	RC	EQ	3	34	0	No
			06827A	12/11/75	Won't open	RC	EQ	2	4	0	No
			24442	9/19/77	Welded steam cut, replaced gasket	RC	EQ	4	30	0	No
	FDW-99	Crane	03270A	7/25/75	Furmanited leaking hinge pin	RC	EQ	2	8	100	No
			08670A	3/4/76	Replaced hinge pin gasket	RC	EQ	2	4	0	No
			12097A	8/3/76	Replaced hinge pin gasket	RC	EQ	2	3	100	No
			13764A	9/14/76	Furmanited leaking hinge pin	RC	EQ	2	4	100	No
			13774A	11/3/76	Replaced packing, gasket	RC	EQ	3	8	0	No
			53685	9/17/77	Retorqued flange halves	RC	EQ	1	1	0	No
	FDW-101	Crane	24198A	8/25/77	Replaced hinge pin gasket	RC	EQ	2	30	0	No
			27404A	11/29/77	Furmanited hinge pin leaks	RC	EQ	2	4	100	No
	FDW-103	Crane	01351A	4/3/75	Rough stem, won't cycle	RC	EQ	2	4	65	No
	FDW-104	Crane	00189A	2/21/75	Tightened packing, lubr. stem	RC	EQ	2	4	75	No
			00195A	2/22/75	Tightened packing, exercised	RC	EQ	2	2	0	No
			03770A	8/1/75	Tightened packing, exercised	RC	EQ	2	2	0	No
			05294A	10/14/75	Stuck limit switch	RC	EQ	2	4	100	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	FDW-104		13500A	9/3/76	Tightened packing	RC	EQ	2	2	0	No
	(cont'd)		10420A	11/7/76	Repacked	RC	EQ	2	4	75	No
	FDW-105				No Data						
	FDW-106	Rockwell	RADCAS	9/9/75	Plugged pinhole	RC	EQ	2	2	100	No
			52602	9/22/77	Repacked	RC	EQ	2	3	0	No
			24466A	7/30/77	Replaced faulty solenoid	RC	EQ	1	13	90	No
	FDW-107				No Data						
	FDW-108	Rockwell	07189	1/6/76	Removed plug to bleed	RC	EQ	1	1	100	No
			23267A	6/15/77	Lubr. stem, adjusted switch	RC	EQ	2	3	100	No
			53778	9/22/77	Repacked	RC	EQ	2	6	0	No
	FDW-113	Velan	20188	8/24/77	Replaced B/B gasket, repacked	RC	EQ	2	6	0	No
	FDW-208	Velan	03443A	7/30/75	Furmanited leak	RC	EQ	2	8	0	No
	FDW-232	?	24681	8/31/77	Replaced hinge pin gaskets	RC	EQ	2	36	0	No
	FDW-236	Crane	24441	9/17/77	Replaced seal, lapped	RC	EQ	3	20	0	No
			01420A	4/14/75	Furmanited B/B leak	RC	EQ	2	4	100	No
	FDW-247	?	Duke	3/26/77	Repaired valve	RC	EQ	2	6	0	No
	FDW-251	Velan	01770A	11/1/74	Lapped seat, repacked, replaced gasket	RC	EQ	1	5	0	No
	FDW-262				No Data						
	FDW-263	Velan	05497A	11/6/75	Furmanited packing leak	RC	EQ	2	4	0	No
	FDW-266	Velan	21862	8/24/77	Replaced B/B gasket, repacked	RC	EQ	2	4	0	No
	FDW-281	?	01733A	11/3/74	Lapped seat leak	RC	EQ	2	6	0	No
Heaters	HTR-1A1		17826A	11/4/76	Removed sight glass, plugged lines	RC	EQ	2	8	0	No
			24406	9/8/77	Reworked, added baffle	RC	EQ	2	240	0	No
			25908	9/9/77	Repaired pipe cap flange leak	RC	EQ	2	3	0	No
			23690	9/8/77	Plugged tube leaks	RC	EQ	2	100	100	No
			53908	9/19/77	Cleaned, installed gaskets	RC	EQ	2	3	0	No
			29191	12/1/77	Repaired weld leaks on manway	RC	EQ	2	4	100	No
	HTR-1A2		11659A	9/13/76	Welded heater outlet piping	RC	EQ	2	8	100	No
			17827A	11/5/76	Removed sight glass, plugged lines	RC	EQ	2	8	0	No
			26464	12/21/77	Plugged leaking tubes	RC	EQ	4	24	0	No
	HTR-1B1		06225	1/17/75	Repaired tube bundle leak	RC	EQ	2	24	0	No
	HTR-1B2		23690	9/8/77	Plugged tube leaks	RC	EQ	2	100	0	No
Pumps	N/A		95791	3/3/77	Blocked auxiliary flow nozzle	RC	EQ	3	8	0	No
	FDW-P1A	Deal	51859A	11/5/76	Shortened vent connection	RC	EQ	2	2	0	No
			24092	8/22/77	Inspected, cleaned bearings	RC	EQ	3	40	0	No
			EEI-041	10/18/77	Feedwater pump trip	RC	EQ	?	9	0	Yes
	FDW-P1B	Deal	Duke	7/11/75	Lost "B" feedwater pump	RC	EQ	1	1	70	Yes
			01076A	3/29/76	Pump trap won't work in automatic	RC	EQ	1	2	0	No
			12690A	8/12/76	Replaced gasket, flange leak	RC	EQ	2	4	75	No
			51859A	11/5/76	Shortened vent connection	RC	EQ	2	1	0	No
			?	1/26/77	Repaired pump drain	RC	EQ	3	12	60	Yes
	?	Deal	10395A	4/20/76	Repaired feedwater pump	RC	EQ	3	12	0	No
	FDW-P1B	Deal	27418	12/14/76	Repaired flange leak on casing	RC	EQ	3	4	0	No
	EFDW	Deal	24415	9/7/77	Overspeed control not working	RC	EQ	2	8	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
Turbines	FDW-P1A	Gen Electric	01364A	4/5/75	Trip on start; oil leak	RC	EQ	2	4	65	No
			23441	8/12/77	Level switch wired wrong	RC	EQ	2	5	0	No
			24177	8/22/77	Replaced gaskets on oil line flange	RC	EQ	2	8	0	No
			26238	10/1/77	Turning gear won't energize	RC	EQ	2	4	0	No
	FDW-P1B	Gen Electric	11655A	7/28/76	Furmanited leaking flange	RC	EQ	2	4	98	No
			EEI-011A	3/17/77	Reset FWPT	RC	EQ		2	60	Yes
	EFDW	Gen Electric			No Data						
Miscellaneous	N/A		EEI-001	1/10/76	Repaired pipe between FDW-281 and pump	RC	EQ	2	4	60	Yes
	N/A		WA sheet	1/?/77	Repaired auxiliary FW nozzle	RC	EQ	2	36	0	No
	N/A		27601	12/5/77	Furmanited heater instrument line	RC	EQ	2	8	100	No
3D CONDENSATE											
Valves	C-1	Rockwell	21853	4/7/77	Sealing water coupling leak	RC	EQ	2	4	100	No
	C-4	Rockwell	20513	3/17/77	Valve leaks through	RC	EQ	4	23	90	No
	C-10	Fisher	19976	8/21/77	Installed new flange gasket	RC	EQ	4	42	0	No
	C-11	Crane-Chap	19353	8/20/77	Repacked	RC	EQ	4	8	0	No
	C-16	Crane-Chap	81047	?	Replaced new seat, restroked	RC	EQ	2	17	?	No
	(Spare)	?	80319	10/24/75	Rebuilt, installed seat, O-rings	RC	EQ	2	6	100	No
	(Spare)	?	80457	1/28/76	Rebuilt, installed seat, O-rings	RC	EQ	2	4	100	No
	(Sapre)	?	80572	4/8/76	Rebuilt	RC	EQ	2	4	?	No
	C-17	Norris	81047	?	Replaced new seat, restroked	RC	EQ	2	17	?	No
			80396	12/19/75	Replaced valve	RC	EQ	2	5	68	No
	C-18	Norris	81047	?	Replaced seat, restroked	RC	EQ	2	17	?	No
			80103	5/16/75	Replaced shaft, flapper	RC	EQ	2	6	100	No
	C-19	Norris	81047	?	Replaced seat, restroked	RC	EQ	2	17	?	No
	C-20	Norris	81047	?	Replaced seat, restroked	RC	EQ	2	17	?	No
	C-22	Norris	21872	8/20/77	Replaced O-ring	RC	EQ	4	6	0	No
	C-24	Norris	22500	?	Replaced shaft seal	RC	EQ	2	8	0	No
	C-26	Norris	81047	?	Replaced seat, restroked	RC	EQ	2	17	0	No
			80282	2/11/74	Replaced valve	RC	EQ	2	4	?	?
	C-95	Crane	27403	11/29/77	Furmanited B/B leak	RC	EQ	2	8	100	No
			01137A	3/30/75	Body-bonnet leak	RC	EQ	2	2	60	No
			22492	8/26/77	Replaced B/B gasket, cleaned	RC	EQ	4	48	0	No
	C-97	Crane	01142A	3/28/75	Repaired B/B leak	RC	EQ	4	24	90	No
			26185	9/29/77	Tightened packing	RC	EQ	2	2	0	No
			10862	8/26/77	Replaced gaskets, cleaned	RC	EQ	4	48	0	No
			25862	9/24/77	Replaced B/B gasket	RC	EQ	4	24	0	No
	C-152	Allis-Chal	24203	7/21/77	Replaced gaskets, repacked	RC	EQ	4	24	90	No
	C-153	Allis-Chal	24204	7/21/77	Replaced gaskets, repacked	RC	EQ	4	24	90	No
	C-176	Fisher	02856	10/30/74	Seat leak, installed seal	RC	EQ	2	19	0	No
			07357A	2/13/76	Installed new valve	RC	EQ	3	8	0	No
			26729	10/20/77	Repaired air line	RC	EQ	2	2	40	No
	C-181	Fisher	01558	10/30/74	Replaced seat seal	RC	EQ	4	12	0	No
			07355A	2/15/76	Replaced rubber boot	RC	EQ	2	2	0	No
			09749	4/26/76	Replaced rubber boot	RC	EQ	4	8	0	No
			26729	10/20/77	Reconnected air line	RC	EQ	2	3	40	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	C-185	Crane	09123A	3/19/76	Cleaned, lapped seat	RC	EQ	2	8	0	No
	C-187	Fisher	07337A	2/16/76	Replaced valve	RC	EQ	3	8	0	No
	C-232	?	81087	9/6/77	Removed, inspected	RC	EQ	2	8	0	No
	C-233	Graver	80435A	1/19/76	Replaced seat	RC	EQ	2	4	100	No
			80581A	4/23/76	No Data	--	--	-	--	--	?
	C-237	Fisher	81087	9/6/77	Removed, inspected	RC	EQ	2	8	0	No
	C-240	Graver	80581A	4/23/76	Rebuilt	RC	EQ	2	4	0	No
			80629	6/7/76	Replaced valve	RC	EQ	2	6	100	No
	C-244	?	81087	9/6/77	Removed, inspected	RC	EQ	2	8	0	No
	C-247	Graver	80581A	4/23/76	Rebuilt	RC	EQ	2	4	0	No
	C-251	?	81087	9/6/77	Removed, inspected	RC	EQ	2	8	0	No
	C-254	Graver	80581A	4/23/76	Rebuilt	RC	EQ	2	4	0	No
	C-258	?	81087	9/6/77	Removed, inspected	RC	EQ	2	8	0	No
	C-261	Graver	80374	12/4/75	Replaced	RC	EQ	2	1	75	No
			080543	3/24/76	Rebuilt	RC	EQ	2	2	0	No
	C-262	Graver	80372	12/3/75	Replaced	RC	EQ	2	6	75	No
			80499	3/1/76	Repaired solenoid operator	RC	EQ	2	3	0	No
			80543	3/24/76	Rebuilt	RC	EQ	2	2	0	No
	C-263	Graver	80374	12/4/75	Replaced valve	RC	EQ	2	1	75	No
			80543	3/24/76	Rebuilt	RC	EQ	2	2	0	No
	C-264	Graver	80547	3/25/76	Replaced butterfly	RC	EQ	2	2	0	No
	C-267	Graver	80374	12/4/75	Replaced valve	RC	EQ	2	3	75	No
	C-269	Graver	80372	12/3/75	Replaced valve	RC	EQ	2	3	75	No
			80574	6/18/76	Replaced valve	RC	EQ	2	3	100	No
	C-270	Graver	80584	6/18/76	Replaced valve	RC	EQ	2	3	100	No
	C-271	Graver	80581	4/23/76	Replaced valve	RC	EQ	2	4	0	No
	C-272	Graver	80581	4/23/76	Replaced valve	RC	EQ	2	4	0	No
			26160	9/28/77	Tightened flanges, packing	RC	EQ	2	2	0	No
	C-273	Graver	80581	4/23/76	Replaced valve	RC	EQ	2	4	0	No
	C-274	Graver	80581	4/23/76	Replaced valve	RC	EQ	2	4	0	No
			80597	6/3/76	Repaired shaft	RC	EQ	2	4	0	No
	C-275	Graver	80581	4/23/76	Replaced valve	RC	EQ	2	4	0	No
	C-333	Fisher	23871	7/21/77	Adjusted, repacked	RC	EQ	2	17	90	No
	C-339	Velan	23871	7/21/77	Adjusted, repacked	RC	EQ	2	17	90	No
Pumps and Motors — Hotwell	1A	Ing-Rand	23294	8/12/77	Changed oil in motor	PM	EQ	2	34	0	No
			52295	9/8/77	Replaced expansion joint	RC	EQ	3	24	0	No
			21100	8/31/77	Inspected, test, changed oil	ITC/PM	EQ	2	5	0	No
	1B	Ing-Rand	EEI-032	10/9/76	Lower bearing oil temp high	RC	EQ	2	21	45	Yes
			20861	3/15/77	Replaced wiring, modified motor	RC	EQ	2	39	90	Yes
			23295	8/12/77	Changed oil	PM	EQ	2	33	0	No
			23200	8/18/77	Replaced gasket	RC	EQ	2	18	0	No
	1C	Ing-Rand	23298	8/12/77	Changed oil	PM	EQ	2	8	0	No
			21100	8/31/77	Changed filters, test, inspected	ITC/PM	EQ	2	4	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
Pumps and Motors — Condensate	1A	?	26226	9/29/77	Replaced nipple on vent	RC	EQ	2	8	0	No
	1B	?			← No Data →						
	1C	?									
Air Ejectors	1A	?	24209	9/23/77	Cleaned nozzles, repl gasket	RC	EQ	2	9	0	No
	1B	?	24210	9/23/77	Cleaned nozzles, repl gasket	RC	EQ	2	6	0	No
	1C	?	24211	9/23/77	Cleaned nozzles, repl gasket	RC	EQ	2	6	9	No
Condensate Coolers	1A	?			← See 3E CONDENSER CIRCULATING WATER →						
	1B	?									
Condenser	1A	?									
	1B	?			← See 3E CONDENSER CIRCULATING WATER →						
	1C	?									
3E CONDENSER CIRCULATING WATER											
Valves	CCW-8	?									
	CCW-21	Pratt			← No Data →						
	CCW-22	Pratt									
	CCW-23	Pratt	24803	9/14/77	Solenoid air bypass leaking	RC	EQ	2	2	0	No
	CCW-24	Pratt			← No Data →						
	CCW-25	Pratt									
	CCW-26	Allis-Chal	22710	9/23/77	Valve cycling	RC	EQ	2	16	0	No
	CCW-27	Allis-Chal			← No Data →						
	CCW-28	Allis-Chal									
Pumps	CCW-P1A	?	12898	8/39/77	Changed oil in motor	RC	EQ	2	9	0	No
	CCW-P1B	?	19412	8/39/77	Changed three heaters	RC	EQ	2	3	0	No
	CCW-P1B	?	24274	8/12/77	Changed oil	PM	EQ	2	12	0	No
	CCW-P1C	?	06395	8/30/77	Changed three heaters	RC	EQ	2	4	0	No
	CCW-P1C	?	23293	8/30/77	Changed oil	PM	EQ	2	18	0	No
Condenser	1A										
	1B				← No Data →						
	1C										
Coolers — RCW	A										
	B				← No Data →						
	C										
	D										
Coolers — Condensate	1A				← No Data →						
	1B										
Heaters	HTR-1C1		53577	9/6/77	Repaired tube leaks	RC	EQ	4	30	0	No
	HTR-1C2		04254	11/7/74	Plugged leaking tube	RC	EQ	2	8	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	HTR-1D1, HTR-1D2, HTR-1E1, HTR-1E2, HTR-1F1, HTR-1F2, HTR-1F3	No Data									
3F RECIRCULATED COOLING WATER (RCW)											
Pumps	RCW-P1A	No Data									
	RCW-P1B	Ing-Rand	25924	9/28/77	Replaced bearing	RC	EQ	3	8	0	No
	RCW-P1C	Ing-Rand	24970	8/22/77	Replaced cooling water line	RC	EQ	2	1	0	No
Coolers	No Data										
Valves	No Data										
3G AUXILIARY STEAM SYSTEM											
Steam Seals	No Data										
3H MOISTURE SEPARATORS AND REHEATERS											
Reheaters	1A1	Gen Electric	10687A	5/25/76	Vacuum leak, tightened manways	RC	EQ	2	2	0	No
			16960	11/23/76	Replaced manway gasket	RC	EQ	2	6	0	No
	1A2	Gen Electric	04647A	3/8/76	Repaired leaking manways	RC	EQ	2	9	0	No
			06352A	3/8/76	Replaced arrangement baffles	RC	EQ	12	18	0	No
			09551A	5/10/76	Welded manways to stop leakage	RC	EQ	4	4	0	No
			10687A	5/26/76	Tightened manway covers	RC	EQ	2	2	0	No
			19175	1/20/77	Seal-welded manway	RC	EQ	2	8	0	No
			22134	9/16/77	Seal-welded manway	RC	EQ	2	12	0	No
	1B1	Gen Electric	07164A	3/8/76	Repaired leaking manway	RC	EQ	3	6	0	No
			Duke	5/26/76	Tightened manway cover	RC	EQ	2	8	0	No
			20194	9/16/77	Replaced gasket, seal-welded	RC	EQ	2	12	0	No
	1B2	Gen Electric	06448	2/11/75	Replaced high-level pot. gasket	RC	EQ	2	2	0	No
			04648	12/15/75	Repaired weld, leak in manway	RC	EQ	2	2	100	No
			06353A	3/8/76	Replaced impingement baffles	RC	EQ	12	18	0	No
			09551A	5/10/76	Welded manway	RC	EQ	2	2	0	No
			10687A	5/26/76	Tightened all manways	RC	EQ	2	2	0	No
			17033A	11/23/76	Welded up manway	RC	EQ	2	4	0	No
			20195	9/11/77	Replaced gasket, welded manway	RC	EQ	2	12	0	No
Pumps	A, B	No Data									

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
3I GENERATOR STATOR COOLING											
Stator Cooling Water											
Pumps	SC-P1A	Gen Electric	24109A	8/22/77	Replaced seal	RC	EQ	2	19	0	No
	SC-P1B	Gen Electric	22779	8/18/77	Replaced gasket	RC	EQ	2	4	0	No
Coolers	1A	Basco	21130	8/24/77	Replaced gasket, O-ring	RC	EQ	3	16	0	No
	1B	Basco	26186	9/30/77	Repaired flange leak	RC	EQ	2	6	0	No
Valves	← No Data →										
Hydrogen Cooling											
Valves	H-7	Rego	18791A	2/17/76	Installed new union seals	RC	EQ	2	4	0	No
	H-14	Fisher	24976	9/9/77	Tightened leaking fitting	RC	EQ	2	5	0	No
3J HEATER DRAIN SYSTEM											
Valves	HD-27	Fisher	04233	11/18/74	Reseated, new gaskets	RC	EQ	2	4	0	No
			02044	8/18/75	Furmanited B/B leak	RC	EQ	2	20	100	No
			EEI-021	4/24/77	Adjusted actuator	RC	EQ	1	6	0	Yes
	HD-28	Fisher	04233	11/18/74	Reseated, new gaskets	RC	EQ	2	4	0	No
			23255	6/15/77	Welded casing pinhole	RC	EQ	2	2	100	No
	HD-30	?	22497	9/13/77	Installed new internals	RC	EQ	2	15	0	No
	HD-33	Fisher	24168	8/15/77	Installed flexible gasket	RC	EQ	2	16	0	No
	HD-42	Velan	09900	4/20/76	Lapped seat, replaced gaskets	RC	EQ	2	16	0	No
	HD-50	Crane	03817	8/25/75	Furmanited leaking hinge pin	RC	EQ	2	8	100	No
	HD-55	Crane	24186	8/10/77	Cleaned, replaced gasket	RC	EQ	2	8	0	No
	HD-69	Crane	01823	5/30/75	Furmanited B/B leak	RC	EQ	2	8	100	No
			01439	8/11/75	Furmanited B/B leak	RC	EQ	2	8	90	No
			04633	10/7/75	Furmanited B/B leak	RC	EQ	2	8	100	No
	HD-71	Velan	09982	4/20/76	Lapped seat, replaced B/B gasket	RC	EQ	3	8	0	No
	HD-83	Crane	24185	8/10/77	Repacked	RC	EQ	3	4	0	No
	HD-86	Velan	01613	10/25/74	Repaired seat leak, gasket; repacked	RC	EQ	2	4	0	No
	HD-91	Crane	10835	9/15/76	Tightened flange	RC	EQ	2	2	0	No
			24181	8/11/77	Repacked	RC	EQ	2	2	100	No
	HD-92	Fisher	01131	4/11/75	Furmanited B/B leak	RC	EQ	2	8	90	No
			02042	5/18/75	Repaired bonnet, stem	RC	EQ	3	8	0	No
			02711	8/11/75	Furmanited B/B leak	RC	EQ	2	8	100	No
			01656A	2/13/76	Broken operator, bent stem	RC	EQ	2	45	0	No
	HD-93	Crane	25427	6/29/77	Repacked	RC	EQ	3	2	100	No
	HD-95	Fisher	10811	7/21/76	Furmanited B/B leak	RC	EQ	2	8	100	No
			10835	9/15/76	Tightened leaking flange	RC	EQ	2	2	100	No
			11015	9/30/76	Replaced gasket	RC	EQ	2	8	100	No
			16694	11/25/76	Flange leak, replaced gasket	RC	EQ	2	16	0	No
			17195	1/18/77	Replaced flexible gasket	RC	EQ	2	4	0	No
			23199	8/16/77	Replaced gaskets in body	RC	EQ	2	2	0	No
			23911	8/16/77	Cut off flange, refaced; gaskets	RC	EQ	2	20	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	HD-100	Velan	09779	4/21/76	Replaced	RC	EQ	2	4	0	No
	?	?	24287	8/11/77	Replaced B/B gasket	RC	EQ	2	9	0	No
	?	?	24284	8/12/77	Replaced B/B gasket, repacked	RC	EQ	2	4	0	No
	?	?	24169	8/12/77	Replaced B/B gasket, repacked	RC	EQ	2	10	0	No
	?	?	3790A	8/13/75	Furmanited to stop leak	RC	EQ	2	4	0	No
	?	?	20843	3/2/77	Added packing	RC	EQ	2	4	0	No
	HD-105	Crane	08965	3/18/76	Lapped seat	RC	EQ	2	8	0	No
			27410	12/7/77	Furmanited flange	RC	EQ	2	8	0	No
	HD-106	Crane	08966	3/18/76	Lapped seat, disk	RC	EQ	2	8	0	No
	HD-109	Crane	03276	8/11/75	Furmanited leaking flange	RC	EQ	2	9	90	No
	HD-113	Fisher	03811	8/13/75	Furmanited leaking flange	RC	EQ	2	8	100	No
	HD-123	Crane-Chap	02013A	8/11/75	Furmanited leaking B/B	RC	EQ	2	14	90	No
			04636	11/7/75	Furmanited leaking flange	RC	EQ	2	8	30	No
	HD-137	Crane	05494	11/6/75	Furmanited leaking B/B	RC	EQ	2	8	0	No
	HD-142	Crane	03810	8/14/75	Changed out gasket	RC	EQ	3	3	100	No
	HD-143	Crane	03810	8/14/75	Changed out gasket	RC	EQ	3	3	100	No
	HD-145	Velan	29148	11/29/77	Replaced B/B gasket, repacked	RC	EQ	2	6	0	No
	HD-150	Crane-Chap	20516	3/17/77	Replaced B/B gasket, repacked	RC	EQ	2	36	90	No
			20654	3/6/77	Repacked	RC	EQ	4	6	0	No
	HD-156	Crane-Chap	19170	9/12/77	Replaced B/B gasket, repacked	RC	EQ	2	6	0	No
	HD-164	Crane-Chap	26192	9/29/77	Replaced B/B gasket	RC	EQ	2	40	0	No
	HD-165	Crane-Chap	26192	9/29/77	Replaced B/B gasket	RC	EQ	2	40	0	No
	HD-178	Velan	27113	12/22/77	Lapped seat	RC	EQ	2	2	0	No
	HD-190	Fisher	04543	2/14/76	Replaced seat	RC	EQ	2	67	0	No
	HD-192	Crane	04543	2/14/76	Lapped seat, disk	RC	EQ	2	67	0	No
	HD-205	Crane-Chap	02030A	5/22/75	Furmanited leak	RC	EQ	2	8	100	No
	HD-208	Fisher	21848	9/14/77	Replaced internals	RC	EQ	2	8	0	No
	HD-224	Fisher	21848	9/14/77	Replaced internals	RC	EQ	2	8	0	No
			24807	8/18/77	Replaced solenoid coil	RC	EQ	2	2	0	No
	HD-227	Crane-Chap	01054A	5/7/75	Furmanited	RC	EQ	2	8	100	No
	HD-269	Fisher	19153	8/16/77	Repacked	RC	EQ	3	8	0	No
			23178	8/16/77	Repaired positioner, repacked	RC	EQ	6	8	0	No
	HD-319	Lonergan	09096A	3/23/76	Replaced gate	RC	EQ	2	3	0	No
	HD-351	Velan	02333	10/28/74	Lapped seat, repacked	RC	EQ	2	4	0	No
	HD-401	Velan	05498	11/10/75	Furmanited	RC	EQ	2	8	90	No
	HD-404	?	22508	8/11/77	Replaced gasket	RC	EQ	2	6	0	No
	HD-422	Velan	23251	6/15/77	Rewelded leak	RC	EQ	2	2	100	No
Pumps											
Heater Drain	1D1	Byron-Jackson	23899	8/29/77	Changed oil in pump motor	PM	EQ	2	6	0	No
Tank Pumps	1D2	Byron-Jackson	23900	8/26/77	Changed oil in pump motor	PM	EQ	2	6	0	No
Heater Drain	1E1	Byron-Jackson	04575	3/27/75	Installed repaired pumps	RC	EQ	3	6	90	No
Pumps			17268	10/14/76	Cleaned, regalanced, realigned	RC	EQ	3	38	97	No
			17272	10/19/76	Changed oil, strainer	RC	EQ	2	4	97	No
			19382	1/28/77	Rep. auxiliary oil pump	RC	EQ	2	8	100	No
			20910	3/15/77	Rep. auxiliary oil pump	RC	EQ	3	8	90	No
			23431	8/29/77	Changed oil, grease in motor	PM	EQ	2	6	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	1E2	Byron-Jackson	04575	3/27/75	Installed repaired pump	RC	EQ	3	6	90	No
			17272	10/19/76	Inspected, chainged oil, strainer	RC	EQ	2	4	97	No
			52177	1/5/77	Installed, aligned new motor	RC	EQ	3	10	100	No
			22334	8/23/77	Bearings using excessive oil	RC	EQ	2	1	0	No
			23431	8/29/77	Changed oil in motor	PM	EQ	2	6	0	No
			29189	12/13/77	Auto. oiler leak	RC	EQ	2	4	0	No
Moisture Separator Dr. Pumps	1A 1B	Ing-Rand Ing-Rand	No Data								
1st Stage Reheat Dr. Tank Pumps	1A 1B	Ing-Rand Ing-Rand	No Data								
Coolers	?	B-L-H	No Data								
Tanks											
1st Stage Reheat	1A	?	03818A	8/22/75	Furmanited leaking flange	RC	EQ	2	4	100	No
			04123A	8/26/75	Furmanited leaking flange	RC	EQ	2	4	100	No
			05966A	11/21/75	Furmanited leak	RC	EQ	2	4	100	No
			24118	8/12/77	Repacked tank level det. valve	RC	EQ	2	8	0	No
			24179	8/12/77	Upper level det. flange leak	RC	EQ	2	4	0	No
			05705	2/19/76	Repaired flange leak	RC	EQ	2	8	0	No
			08983	3/26/76	Replaced gaskets	RC	EQ	2	6	0	No
	1B	?	05967	11/21/75	Replaced gaskets	RC	EQ	2	4	0	No
			24286	?	Replaced gasket	RC	EQ	2	2	0	No
2nd Stage Reheat	1A	?	05968A	11/21/75	Furmanited leaks	RC	EQ	2	4	100	No
			23196	7/1/77	Repaired sight glass valve	RC	EQ	2	6	100	No
			24180	7/21/77	Repaired leaks	RC	EQ	2	2	90	No
			17198	8/14/77	Remove inst. valve	RC	EQ	2	8	100	No
			24285	8/12/77	Replaced gasket	RC	EQ	2	4	90	No
			24282	8/12/77	Replaced gasket	RC	EQ	2	4	90	No
			10180	8/26/77	Lapped sight glass valve seat	RC	EQ	2	6	100	No
	1B	?	05723	10/24/75	Furmanited flange leak	RC	EQ	2	4	100	No
			13006A	11/27/76	Replaced flange, gaskets	RC	EQ	3	12	0	No
			17198	5/24/77	Furmanited flange	RC	EQ	2	4	0	No
			23915	8/10/77	Replaced gasket	RC	EQ	2	16	0	No
			19305	8/12/77	Replaced gasket	RC	EQ	2	6	0	No
			24281	8/20/77	Repaired flange leak	RC	EQ	2	20	0	No
			24280	8/20/77	Repaired flange leak	RC	EQ	2	8	0	No
			05677	1/10/75	Replaced manway gasket	RC	EQ	2	4	0	No
Moisture Separator Drain	1A 1B	No Data									
Feedwater Flash	1C1	?	04049	8/22/75	Furmanited flange leak	RC	EQ	2	4	0	No
			23007	6/2/77	Repaired sight glass leak	RC	EQ	2	2	100	No
			24187	7/21/77	Level detector flange leak	RC	EQ	2	8	100	No
	1C2	?	No Data								
	1D1	?	05246	11/7/76	Leaking sight glass	RC	EQ	2	2	0	No
			25427	9/6/77	Replaced broken sight glass	RC	EQ	2	3	0	No
	1D2	?	25427	9/6/77	Replaced broken sight glass	RC	EQ	2	3	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
Piping, Drain Line	N/A	?	EEI-77-23	4/26/77	Repaired leak	RC	EQ	?	10	80	Yes
3K INSTRUMENT AIR	←				No Data						→
3L TURBINE LUBE OIL SYSTEM		Sharples	01125	7/18/74	Reinstalled purif'n bowl brakes	RC	EQ	2	8	100	No
		Sharples	02316	9/1/74	Replaced bearing	RC	EQ	2	4	100	No
		Sharples	03250	2/1/75	Checked out purification system	RC	EQ	2	4	0	No
		Sharples	04105	2/1/75	Replaced bowl	RC	EQ	2	4	0	No
		Sharples	06450	2/7/75	Installed new oiler pot	RC	EQ	2	8	0	No
		Sharples	00468A	3/18/75	Cleaned suction strainer	RC	EQ	2	8	75	No
		Sharples	00838A	3/21/75	Replaced heater fuses	RC	EQ	2	2	0	No
		Sharples	01032A	3/24/75	Installed brk lng on oil pur.	RC	EQ	2	8	75	No
		Sharples	01017A	3/27/75	Adjusted heater thermostat	RC	EQ	2	2	90	No
		Sharples	01412A	4/16/75	Replaced bowl sleeve, bushing	RC	EQ	2	6	100	No
		Sharples	Duke	4/21/75	Pumps trip — improper valving	RC	EQ	2	4	100	No
		Sharples	02398A	6/2/75	Replaced bearing, clutch	RC	EQ	2	6	100	No
		Sharples	02619A	6/18/75	Tightened leaking sight glass	RC	EQ	2	4	100	No
		Sharples	02865A	6/25/75	Replaced bushing	RC	EQ	2	3	100	No
		Sharples	03084A	7/7/75	Replaced clutch bearings	RC	EQ	2	8	100	No
		Sharples	03567	7/31/75	Replaced bearings	RC	EQ	2	6	0	No
		Sharples	03771A	8/4/75	Replaced purif'n CR relay	RC	EQ	2	2	90	No
		Sharples	04036A	8/25/75	Replaced coupling, clutch, belt	RC	EQ	2	4	100	No
		Sharples	04030A	8/29/75	Replaced blown fuses	RC	EQ	2	2	100	No
		Sharples	04360A	8/30/75	Installed clutch bearings	RC	EQ	2	8	100	No
		Sharples	03854A	9/17/75	Replaced bearings, coupling	RC	EQ	2	4	100	No
		Sharples	04955A	9/23/75	Replaced sleeve, drive belt	RC	EQ	2	8	100	No
		Sharples	05295A	10/16/75	Replaced bushing, spring, gasket	RC	EQ	2	2	100	No
		Sharples	05632A	10/23/75	Installed new brake assembly	RC	EQ	2	8	100	No
		Sharples	05649A	10/27/75	Installed new spring	RC	EQ	2	2	100	No
		Sharples	06121A	11/13/75	Replaced brg assy, clutch, coup'g	RC	EQ	2	4	100	No
		Sharples	06210A	11/21/75	Replaced fuses, adjust thermostat	RC	EQ	3	3	100	No
		Sharples	06497A	12/9/75	Repaired oil leak	RC	EQ	2	4	100	No
		Sharples	06897A	12/16/75	Replaced flexible coupling	RC	EQ	2	8	100	No
		Sharples	06966A	12/22/75	Realigned pump	RC	EQ	2	4	100	No
		Sharples	06990A	12/26/76	Replaced clutch	RC	EQ	2	4	100	No
		Sharples	07797A	4/3/76	Replaced clutch, adj. cylinder	RC	EQ	1	2	100	No
		Gen Electric	EEI 76-15	6/21/76	Repaired oil leak	RC	EQ	1	3	12	Yes
		Gen Electric	EEI 77-16	3/30/77	Insp brg, rep. oil pump	RC	EQ	?	74	0	Yes
		Gen Electric	EEI 77-28	6/6/77	Low oil pressure — turbine trip	RC	EQ	2	2	0	Yes
3M TURBINE EHC SYSTEM		Gen Electric	Gray Book	3/15/75	Repaired oil leak	RC	EQ	2	3	20	Yes
		Gen Electric	EEI 011	6/8/75	Low oil pressure	RC	EQ	2	16	0	Yes
		Gen Electric	23747	8/12/77	Calibrated pressure transmitter	RC	EQ	2	9	0	No
		Gen Electric	95867	8/12/77	Repl turbine relays, add switch	RC	EQ	4	42	0	No
		Gen Electric	95758	8/21/77	Add time delay to valve circuit	RC	EQ	2	4	0	No
		Gen Electric	95763	8/21/77	Add EHC interlock	RC	EQ	2	15	0	No
		Gen Electric	95888	9/24/77	Add time delay to EHC	RC	EQ	2	7	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
		Gen Electric	53416	9/12/77	Inspected EHC	RC	EQ	4	15	0	No
		Gen Electric	52610	9/12/77	Inspected HP fluid system	RC	EQ	2	3	0	No
		Gen Electric	53609	9/15/77	Inspected thrust wear detector	RC	EQ	2	4	0	No
3N HIGH-PRESSURE SERVICE WATER											
Pump Motor	HPSW-P1A	?	21098	8/23/77	Inspected, tested motor	RC	EQ	4	3	0	No
Pump Motor Cooler	HPSW-P1B	?	24988	8/23/77	Repaired tubing	RC	EQ	3	12	0	No
3O (Not Used)											
3P NITROGEN SUPPLY											
Valves	N-116	?	22721	5/15/77	Welded held wheel to stem	RC	EQ	2	4	0	No
		?	95999	9/3/77	Replaced valve	NSM	EQ	2	24	0	No
	N-130	Williams	24470	8/29/77	Lapped seat, replaced gasket	RC	EQ	2	4	0	No
	N-137	Crane	09570	4/29/76	Repaired seat leak, repacked	RC	EQ	2	8	0	No
			21148	5/12/77	Lapped gate, replaced bushing	RC	EQ	2	6	0	No
			25409	9/9/77	Replaced valve	RC	EQ	2	24	0	No
	N-128	Velan	95978	9/5/77	Replaced valve	NSM	EQ	2	20	0	No
3Q STEAM DRAINS											
Valves	SD-7	?	01720	10/24/74	Repaired seat leak	RC	EQ	?	18	0	No
			07844	2/14/76	Repaired seat leak	RC	EQ	2	2	0	No
			09899	4/25/76	Replaced seats, gaskets; repacked	RC	EQ	?	8	0	No
	SD-8	?	01720	10/24/74	Repaired seat leak	RC	EQ	?	18	0	No
			07845	2/14/76	Repaired seat leak	RC	EQ	2	2	0	No
			09899	4/25/76	Replaced seats, gaskets; repacked	RC	EQ	?	8	0	No
	SD-19	?	22365	8/10/77	Replaced valve	RC	EQ	2	4	0	No
	SD-20	?	22365	8/19/77	Replaced valve	RC	EQ	2	4	0	No
	SD-23	?	06466	2/14/76	Lapped seats, replaced gasket	RC	EQ	2	2	0	No
	SD-24	?	06466	2/14/76	Lapped seats, replaced gaskets	RC	EQ	2	2	0	No
	SD-28	?	24182	8/17/77	Lapped seats	RC	EQ	2	2	0	No
	SD-37	?	01379A	8/17/77	Furmanited leak	RC	EQ	?	?	0	No
	SD-39	Velan	27300	11/18/77	Replaced B/B gaskets	RC	EQ	2	3	0	No
	SD-40	Velan	37300	11/18/77	Replaced B/B gaskets	RC	EQ	2	3	0	No
	SD-47	Velan	09789	4/26/76	Lapped seats, replaced gasket	RC	EQ	2	4	0	No
	SD-53	Velan	01706	10/25/74	Seat leak lapped	RC	EQ	2	2	0	No
			07899	2/14/76	Repaired seat leak	RC	EQ	?	4	0	No
			09962	5/21/76	Replaced valve	RC	EQ	?	12	0	No
	SD-54	Velan	01706	10/25/74	Seat leak lapped	RC	EQ	?	4	0	No
			07900	2/14/76	Repaired seat leak	RC	EQ	?	4	0	No
			09962	5/21/76	Replaced valve	RC	EQ	?	12	0	No
	SD-70	?	07172	2/13/76	Replaced seat	RC	EQ	2	6	0	No
			22481	8/11/77	Replaced valve	RC	EQ	2	5	0	No
	SD-72	?	19309	8/17/77	Replaced B/B gasket	RC	EQ	2	5	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	SD-73	?	24461	8/14/77	Repaired, installed gasket	RC	EQ	?	4	0	No
	SD-78	?	10173	4/25/76	Repacked	RC	EQ	?	4	0	No
			19303	8/6/77	Repaired	RC	EQ	?	10	0	No
	SD-80	?	20869	8/16/77	Installed new seat, spring	RC	EQ	?	6	0	No
	SD-82	?	22493	8/17/77	Replaced B/B gasket	RC	EQ	2	4	0	No
	SD-84	?	02887	7/27/75	Lapped seat leak	RC	EQ	2	6	0	No
			09787	4/30/76	Replaced B/B gasket	RC	EQ	2	8	0	No
			19156	1/29/77	Repaired seat leak	RC	EQ	2	8	100	No
	SD-85	?	02887	7/27/75	Lapped seat leak	RC	EQ	2	6	0	No
			09787	4/30/76	Replaced B/B gasket	RC	EQ	2	8	0	No
			19156	1/29/77	Repaired seat leak	RC	EQ	2	8	100	No
	SD-89	?	24104	8/17/77	Replaced valve	RC	EQ	2	10	0	No
	SD-114	Kerotest	03511	7/29/75	Lapped seat, repacked	RC	EQ	?	10	0	No
	SD-115	Kerotest	03511	7/29/75	Lapped seat, repacked	RC	EQ	?	10	0	No
	SD-118	Kerotest	01732	10/25/74	Lapped seat leak	RC	EQ	2	2	0	No
	SD-119	Kerotest	01732	10/25/74	Lapped seat leak	RC	EQ	2	2	0	No
	SD-126	?	01124	7/29/75	Lapped, repacked	RC	EQ	2	6	0	No
	SD-127	Velan	01124	7/29/75	Lapped, repacked	RC	EQ	2	6	0	No
	SD-135	?	01127A	4/3/75	Furmanited leak	RC	EQ	?	?	65	No
	SD-146	Velan	09789	4/26/76	Lapped seats, repacked	RC	EQ	2	4	0	No
	SD-188	?	24183	8/17/77	Lapped seats, repacked	RC	EQ	?	4	0	No
	SD-199	?	22336	4/25/77	Replaced valve	RC	EQ	2	6	0	No
	SD-204	?	04847	1/5/76	Replaced valve	RC	EQ	2	4	0	No
	SD-235	?	03576	7/29/75	Replaced valve	RC	EQ	2	4	0	No
			09085	3/23/76	Replaced valve	RC	EQ	2	4	0	No
	SD-240	?	25864	9/27/77	Replaced valve	RC	EQ	2	6	0	No
	SD-241	?	01731	10/25/74	Lapped, replaced seat	RC	EQ	2	7	0	No
			02890	7/27/75	Lapped seat	RC	EQ	2	4	0	No
			09083	3/23/76	Replaced valve	RC	EQ	2	4	0	No
	SD-273	Rock-Edw.	00374	7/25/75	Lapped seat leak	RC	EQ	?	20	0	No
	SD-288	Velan	53325	8/8/77	Replaced control, wiring	RC	EQ	2	8	0	No
	SD-307	Velan	02894	7/29/75	Lapped seat leak	RC	EQ	?	8	0	No
	SD-419	Edward	04621	11/21/75	Furmanited leak	RC	EQ	?	?	100	No
3R VACUUM											
Valves	V-84	Crane	Duke	8/18/77	Repacked	RC	EQ	2	8	0	No
	V-85	Crane	Duke	8/18/77	Repacked	RC	EQ	2	8	0	No
	V-86	Crane	Duke	8/18/77	Repacked	RC	EQ	2	8	0	No
	V-132	?	Duke	8/21/77	Cleaned, replaced sponge balls	RC	EQ	2	8	0	No
	V-136	?	Duke	8/21/77	Cleaned, replaced sponge balls	RC	EQ	2	8	0	No
	V-148	?	Duke	8/21/77	Cleaned, replaced sponge balls	RC	EQ	2	8	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
4 AUXILIARY MECHANICAL EQUIPMENT											
4A CONTROL ROD DRIVES											
4A1 Drives	?	Diamond	EOF	12/22/74	Galled leadscrew	RC	EQ	?	48	0	No
	CRD M-3	Diamond	50613	1/1/75	Rethreaded drives	RC	EQ	2	8	0	No
	CRD K-9	Diamond	08873A	4/21/76	Replaced torque taker, gaskets	RC	EQ	3	24	0	No
	CRD E-11, N-8	Diamond	11159A	6/9/76	Jogged debris from roller nut	RC	EQ	2	3	0	Yes
	CRD M-5	Diamond	EEI-028	8/31/76	Jogged debris from roller nut	RC	EQ	2	2	0	Yes
	CRD M-11, N-8	Diamond	13193A	9/1/76	Exercised leadscrew	RC	EQ	2	3	0	Yes
	Group 6	Diamond	12843	8/15/76	Exercised leadscrew	RC	EQ	2	4	0	Yes
	CRD O-7	Diamond	13474A	9/5/76	Exercised leadscrew	RC	EQ	2	3	0	Yes
	?	Diamond	20218	2/21/77	Replaced motor	RC	EQ	2	5	100	No
4A2 Stators	CRD M-13	Diamond	02485	8/30/74	Replaced stator	RC	EQ	2	12	0	No
	?	Diamond	EOF	10/7/74	Stuck rod	RC	EQ	2	12	0	Yes
	?	Diamond	EOF	11/6/74	Replaced six stators	RC	EQ	4	24	0	No
	?	Diamond	EOF	12/22/75	Replaced two stators	RC	EQ	2	8	0	No
	CRD D-10	Diamond	06465	1/13/75	Replaced stator	RC	EQ	4	8	0	No
	?	Diamond	05664	1/19/75	Replaced stator	RC	EQ	4	4	0	No
	?	Diamond	EOF	1/31/75	Replaced stator	RC	EQ	4	4	0	No
	?	Diamond	EOF	2/2/75	Replaced stator	RC	EQ	4	4	0	No
	CRD L-10	Diamond	00037A	2/24/75	Replaced stator	RC	EQ	3	8	0	No
	CRD G-13	Diamond	00210A	3/5/75	Replaced stator	RC	EQ	4	4	0	No
	CRD O-7	Diamond	06830A	12/11/75	Replaced stator	RC	EQ	1	3	0	Yes
	CRD G-3	Diamond	50105	12/11/75	Replaced stator	RC	EQ	2	10	0	Yes
	CRD N-6	Diamond	09337A	3/29/76	Replaced stator	RC	EQ	2	12	0	No
	CRD K-9	Diamond	10682A	5/26/76	Lifted, turned rotor	RC	EQ	1	3	0	No
	CRD F-12	Diamond	WA sheet	2/2/77	Replaced stator	RC	EQ	2	2	0	No
	CRD K-5	Diamond	20828	2/28/77	Replaced stator	RC	EQ	2	4	0	Yes
4A3 Position Indicators	CRD L-6	Diamond	03372	11/9/74	Replaced PI tube	RC	EQ	4	2	0	No
	CRD G-3	Diamond	020474	5/18/75	Replaced PI tube	RC	EQ	4	8	0	Yes
	?	Diamond	EOF	8/1/75	Replaced three PI tubes	RC	EQ	2	24	0	No
	CRD M-9	Diamond	18399A	11/19/76	Replaced PI tube	RC	EQ	4	2	0	Yes
	CRD F-12	Diamond	18353	11/19/76	Replaced PI tube	RC	EQ	6	2	0	Yes
	CRD B-6	Diamond	17521A	11/22/76	Replaced PI tube	RC	EQ	4	2	0	Yes
	CRD G-9	Diamond	11317A	12/30/76	Repaired API card	RC	EQ	2	2	100	No
	CRD E-11	Diamond	17181A	1/2/77	Loose API card	RC	EQ	2	2	100	No
			19354A	1/7/77	Replaced API card	RC	EQ	1	1	100	No
	CRD K-11	Diamond	19425A	1/13/77	Replaced API card	RC	EQ	2	3	100	No
	GR-3	Diamond	19741A	1/24/77	Replaced PI tube	RC	EQ	3	4	0	No
	CRD K-9	Diamond	19736A	1/25/77	Replaced PI tube	RC	EQ	3	7	0	Yes
	CRD N-10	Diamond	19841A	1/30/77	Cleaned connection	RC	EQ	2	8	100	No
	CRD K-7	Diamond	19843	1/30/77	Reset reed switch	RC	EQ	2	2	50	Yes
	?	Diamond	50105A	2/3/77	Replaced PI tube	RC	EQ	4	3	0	Yes

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	?	Diamond	50105A	2/3/77	Cleaned, repaired 20 tubes	RC	EQ	4	40	0	Yes
	?	Diamond	50105A	2/3/77	Repaired 20 PI tube cables	RC	EQ	4	30	0	Yes
	CRD L-14	Diamond	20523	3/29/77	Replaced PI tube switch	RC	EQ	4	1	0	No
4A4 Power-T/C Cables											
Power Cables	?	Diamond	19966	2/2/77	Replaced 10 power cables	RC	EQ	4	40	0	Yes
	CRD K-5	Diamond	19855	2/28/77	Replaced power cable	RC	EQ	2	4	0	Yes
T/C Cables	?	Diamond	50105A	2/2/77	Replaced all T/C cables	RC	EQ	4	32	0	Yes
4A5 Closure Vent Assy	?	Diamond	06241	1/19/75	Replaced O-rings on center vent	RC	EQ	2	1	0	No
	CRD 54	Diamond	19110A	1/19/77	Replaced vent valve	RC	EQ	2	1	0	No
	CRD K-7	Diamond	52305	2/2/77	Repaired leak	RC	EQ	2	4	0	Yes
	?	Diamond	20856	3/3/76	Replaced CRDM gasket	RC	EQ	4	4	0	No
	Group 7	Diamond	EOF	4/23/77	Pulled CRDM for repair	RC	EQ	2	7	0	Yes
4A6 Control System	Groups 6-8	Diamond	02405	9/1/74	Replaced switches	RC	EQ	2	8	100	No
	CRD E-9	Diamond	00147A	2/2/75	Calibrated meter	RC	EQ	2	4	0	No
	Group 3	Diamond	00172A	2/20/75	Replaced selector switch	RC	EQ	3	3	0	No
	CRD H-8	Diamond	00226A	2/20/75	Replaced vacuum/pressure gage	RC	EQ	2	6	0	No
	CRD G-11	Diamond	00590A	3/11/75	Replaced switch	RC	EQ	4	1	0	No
	CRD L-2	Diamond	00674A	3/12/75	Replaced reed switch	RC	EQ	4	1	0	Yes
	?	Diamond	00841A	3/22/75	Replaced, repaired switches	RC	EQ	1	13	75	Yes
	Groups6,7	Diamond	01418A	4/11/75	Replaced d-c brake board	RC	EQ	2	6	90	Yes
	CRD L-6	Diamond	01917A	5/2/75	Replaced bad switch	RC	EQ	4	8	100	No
	?	Diamond	EEI-010	5/17/75	Crontrol rod repatch	RC	EQ	?	30	0	Yes
	?	Diamond	EOF	5/17/75	Control rod interchange	RC	EQ	?	21	0	Yes
	?	Diamond	02418A	6/2/75	Replaced switches	RC	EQ	2	3	100	No
	?	Diamond	04399A	9/3/75	Balanced ICS	RC	EQ	1	1	90	Yes
	?	Diamond	94826A	9/19/75	Replaced statalarm card	RC	EQ	2	1	100	No
	CRD E-11, H-4	Diamond	11158A	6/8/76	Repaired power supply wire	RC	EQ	1	9	0	Yes
	Groups 6,7	Diamond	12834A	8/14/76	Replaced 3-2 hold module	RC	EQ	1	8	0	Yes
	?	Diamond	EEI-030	9/4/76	Control rod reptach	RC	EQ	?	26	0	Yes
	?	Diamond	17252A	10/6/76	Replaced "T" handle switch	RC	EQ	2	2	100	No
	Group 5	Diamond	17628A	10/26/76	Checked signals, recalibrated	RC	EQ	2	7	0	Yes
	?	Diamond	18134A	11/15/76	Replaced "T" handle switch	RC	EQ	2	1	0	No
	?	Diamond	18533A	12/6/76	Replaced breaker	RC	EQ	2	6	0	No
	?	Diamond	Duke	3/3/77	Repaired alarm	RC	EQ	1	3	0	No
	?	Diamond	21532	3/30/77	Replaced screws	RC	EQ	1	1	0	No
4B FUEL HANDLING BRIDGES											
Bridges		S-R	EOF	10/28/74	SF bridge cable problem	RC	EQ	?	48	0	Yes
		S-R	EOF	11/1/74	Electrical problems	RC	EQ	?	48	0	Yes
		S-R	EOF	11/4/74	CR mast-MFHB not engaging	RC	EQ	?	24	0	Yes
		S-R	04191	11/4/74	Replaced MHHB Dillon load meter	RC	EQ	2	17	0	No
		S-R	EOF	11/6/74	MFHB electrical cable support	RC	EQ	?	12	0	Yes

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
		S-R	EOF	11/6/74	CR mast not engaging, electrical problem	RC	EQ	?	708	0	Yes
		S-R	07477A	2/9/76	Replaced grapple up-limit switch	RC	EQ	?	4	0	No
		S-R	08389A	2/19/76	Freed roller on MFHB	RC	EQ	?	8	0	No
		S-R	08243A	2/19/76	Replaced reel — MFHB	RC	EQ	?	6	0	No
		S-R	08244A	2/19/76	Adjusted limit on aux FHB	RC	EQ	?	2	0	No
		S-R	07393	2/29/76	Replaced temp hose on CR mast	RC	EQ	?	6	0	No
		S-R	08890	3/6/76	Rewired geared limit switch	RC	EQ	?	4	0	No
		S-R	08833	3/9/76	Hyd. leak -- aux bridge takeup reel	RC	EQ	?	6	0	No
		S-R	08780	3/11/76	Replaced hose — CR mast, MFHB	RC	EQ	?	6	0	No
		S-R	08779	3/11/76	Tightened tube fttg on CR mast	RC	EQ	?	16	0	No
		S-R	08488	3/26/76	Replaced grapple tube light SFP br.	RC	EQ	?	12	0	No
		S-R	10388A	4/27/76	Replaced switch, MFHB	RC	EQ	?	13	0	No
		S-R	10517A	5/12/76	Adjusted limit switch, FHB	RC	EQ	?	2	0	No
		S-R	10443A	5/16/76	Replaced grapple tube switch act.	RC	EQ	?	12	0	No
		S-R	10666A	5/17/76	Tightened wires, SFB hoist trm'ls	RC	EQ	?	4	0	No
		S-R	10674A	5/22/76	Replaced hose, MFHB	RC	EQ	?	6	0	No
		S-R	10678A	5/22/76	Replaced job switch, MFHB	RC	EQ	?	3	0	No
		S-R	10761A	5/23/76	Replaced switch, SFB	RC	EQ	?	8	0	No
		S-R	55003A	8/11/77	PM on fuel handling crane	PM	EQ	4	8	0	No
		S-R	20377	8/12/77	Inspected main, aux FH bridges	ITC	EQ	?	40	0	No
		S-R	23648	8/11/77	Installed swivel on CR mast	RC	EQ	3	12	0	No
		S-R	22115	8/12/77	Installed CRD mast	RC	EQ	4	12	0	No
		S-R	52480	8/14/77	Replaced grapple cams M, AFHB	RC	EQ	4	10	0	No
		S-R	22097	8/16/77	Calibrated load cell	RC	EQ	2	4	0	No
		S-R	EOF	8/17/77	Repaired MFHB orifice rod circuit	RC	EQ	?	32	0	Yes
		S-R	EOF	8/18/77	CR mast repairs	RC	EQ	?	36	0	Yes
		S-R	EOF	8/18/77	Raplaced grapple underload switch on MFHB	RC	EQ	?	2	0	Yes
		S-R	95596	8/18/77	Revised bridge circuit, bumpers	NSM	EQ	4	6	0	Yes
		S-R	EOF	8/20/77	Hose leak, MFHB	RC	EQ	?	6	0	Yes
		S-R	EOF	8/20/77	Repaired hyd. hose, AFHB	RC	EQ	?	2	0	Yes
		S-R	EOF	8/20/77	Repaired CR grapple, MFHB	RC	EQ	?	2	0	Yes
		S-R	EOF	8/26/77	Repaired hose leak on CR cylinder	RC	EQ	?	2	0	Yes
		S-R	EOF	8/30/77	Telescop. cylinder problem	RC	EQ	?	1	0	Yes
		S-R	EOF	8/31/77	Could not latch orifice rod	RC	EQ	?	2	0	Yes
		S-R	EOF	9/1/77	CR mast rotated 1 inch CCW	RC	EQ	?	3	0	Yes
		S-R	EOF	9/1/77	Could not engage fuel assembly	RC	EQ	?	2	0	Yes
		S-R	EOF	9/16/77	Replaced valve on CR mast	RC	EQ	5	20	0	No
4C FUEL TRANSFER SYSTEM											
Tranfer Mechanisms		S-R	EOF	2/7/76	Upender	RC	EQ	?	72	0	Yes
		S-R	08783	3/12/76	Adjusted air pressure, SFP W. upender	RC	EQ	?	2	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
		S-R	10375A	4/23/76	Adjust frame, up-limit switch, E. upender	RC	EQ	?	4	0	No
		S-R	10377A	4/28/76	Replaced hydraulic hose, E. upend.	RC	EQ	?	6	0	No
		S-R	10393A	4/28/76	Installed limit switch, W. upender	RC	RQ	2	4	0	No
		S-R	21089	8/10/77	Removed fuel transfer tube covers	RC	EQ	2	2	0	No
		S-R	23625	8/11/77	Replaced air motor	RC	EQ	3	8	0	No
		S-R	95945	8/11/77	Changed drain plug on hydr. tank	NSM	EQ	2	6	0	No
		S-R	EOF	9/16/77	Repaired transfer mechanism	RC	EQ	?	8	8	Yes
		S-R	EOF	8/21/77	Loose scres, upender motor shaft	RC	EQ	?	2	0	Yes
		S-R	EOF	8/22/77	W. upender would not raise	RC	EO	?	17	0	Yes
		S-R	26413	10/13/77	Revised wiring in SFP carriage control	RC	EQ	?	9	0	No
4D CRDM SERVICE STRUCTURE											
Fans	NA		18376A	11/29/76	Replaced eight fans	RC	EQ	4	10	0	No
Ductwork	NA		EEI 77-15/ -16	3/25/77	Altered ductwork	NSM	EQ	2	4	0	No
4E SUPPRESSORS AND HANGERS											
Hydraulic Suppressors		Itt-Grinell	Gray Book	12/20/76	Repl. eight hydraulic suppressors	RC	EQ	?	37	0	Yes
		Itt-Grinell	WA sheet	1/16/77	Inspected safety-related equipm.	ITC/RC	EQ	?	238	0	No
		Itt-Grinell	EEI 77-15, -16	3/25/77	Changed out 34 suppressors	RC	EQ	?	102	0	No
		Itt-Grinell	52577	5/10/77	Inst. new supp., aux building	RC	EQ	2	4	0	No
		Itt-Grinnel	52592	5/10/77	Inst. new supp., turbine bldg	RC	EQ	2	3	0	No
		Itt-Grinell	52593	5/10/77	Inst. new supp., turbine bldg	RC	EQ	2	3	0	No
		Itt-Grinell	52597	5/10/77	Inst. new supp., turbine bldg	RC	EQ	2	3	0	No
		Itt-Grinell	52598	5/10/77	Inst. new supp., turbine bldg	RC	EQ	2	2	0	No
		Itt-Grinell	52599	5/10/77	Inst. new supp., turbine bldg	RC	EQ	2	3	0	No
		Itt-Grinell	52600	5/11/77	Inst. new supp., turbine bldg	RC	EQ	2	3	0	No
		Itt-Grinell	52743	5/12/77	Inst. new supp., RC pump 1A1	RC	EQ	2	14	0	No
		Itt-Grinell	52744	5/12/77	Inst. new supp., RC pump 1A1	RC	EQ	2	14	0	No
		Itt-Grinell	52745	5/13/77	Inst. new supp., RC pump 1A1	RC	EQ	2	14	0	No
		Itt-Grinell	52746	5/14/77	Inst. new supp., RC pump 1A1	RC	EQ	2	14	0	No
		Itt-Grinell	52747	5/15/77	Inst. new supp., RC pump 1A2	RC	EQ	2	14	0	No
		Itt-Grinell	52748	5/16/77	Inst. new supp., RC pump 1A2	RC	EQ	2	14	0	No
		Itt-Grinell	52463	8/7/77	Insp 10% - safety-related equip	ITC	EQ	2	32	0	No
		Itt-Grinell	52727	8/10/77	Inst. seal kit, 2½x5 supp.	RC	EQ	2	16	0	No
		Itt-Grinell	52729	8/10/77	Inst. seal kit, 2½x5 supp.	RC	EQ	2	10	0	No
		Itt-Grinell	52733	8/10/77	Inst. seal kit, 2½x5 supp.	RC	EQ	2	15	0	No
		Itt-Grinell	52734	8/10/77	Inst. seal kit, 2½x5 supp.	RC	EQ	2	15	0	No
		Itt-Grinell	52890	8/10/77	Repl. thread-seal, washer	RC	EQ	2	5	0	No
		Itt-Grinell	53410	8/10/77	Repaired 2½x5 suppressor	RC	EQ	2	10	0	No
		Itt-Grinell	52735	8/11/77	Inst. seal kit, 2½x5 supp.	RC	EQ	2	14	0	No
		Itt-Grinell	52889	8/12/77	Insp. reactor bldg hydr. supp.	RC	EQ	4	10	0	No
		Itt-Grinell	52859	8/12/77	Inst. seal kit, 6x5 supp.	RC	EQ	2	20	0	No
		Itt-Grinell	52853	8/13/77	Inst. seal kit, 5x5 supp.	RC	EQ	4	16	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time		Actual plant power, %	Did event force or extend outage?
								No. men	Clock hours		
		Itt-Grinell	52854	8/13/77	Installed new 5x5 supp.	RC	EQ	4	16	0	No
		Itt-Grinell	52856	8/13/77	Install seal kit, 5x5 supp.	RC	EQ	4	16	0	No
		Itt-Grinell	52852	8/13/77	Install seal kit, 5x5 supp.	RC	EQ	4	16	0	No
		Itt-Grinell	52888	8/17/77	Inspect hydraulic supp.	ITC	EQ	4	15	0	No
		Itt-Grinell	52857	8/17/77	Install seal kit, 8x5 supp.	RC	EQ	2	30	0	No
		Itt-Grinell	95861	8/17/77	Repl. orifice plug, RCP motor supp.	RC	EQ	4	12	0	No
		Itt-Grinell	53418	8/18/77	Install new 3¼x5 supp.	RC	EQ	4	6	0	No
		Itt-Grinell	52858	8/21/77	Install seal kit, 6x5 supp.	RC	EQ	2	15	0	No
		Itt-Grinell	52728	9/2/77	Install seal kit, 2½x5 supp.	RC	EQ	2	8	0	No
		Itt-Grinell	52730	9/12/77	Install seal kit, 2½x5 supp.	RC	EQ	2	8	0	No
		Itt-Grinell	52732	9/12/77	Install seal kit, 2½x5 supp.	RC	EQ	2	8	0	No
		Itt-Grinell	53532	9/12/77	Inspected hydraulic supp.	RC	EQ	4	10	0	No
		Itt-Grinell	53404	9/13/77	Calib. funct. test machine	RC	EQ	2	8	0	No
		Itt-Grinell	52731	9/17/77	Install seal kit, 2½x5 supp.	RC	EQ	2	8	0	No
		Itt-Grinell	53533	9/21/77	Inspect hydr. supp., reactor bldg	RC	EQ	4	8	0	No
Pipe Hangers		Itt-Grinell	20386	3/3/77	Insp., adjusted 25 hangers	ITC	EQ	4	9	0	No
		Itt-Grinell	53697	9/22/77	Modified pipe hangers	NSM	EQ	4	13	0	No
5 ELECTRICAL SYSTEMS											
5A GENERATOR		Gen Electric	17100	1/1/76	Repl., calibr. load follower	RC	EQ	2	6	95	No
		Gen Electric	13484	9/7/76	Repaired brekaers	RC	EQ	1	2	0	No
		Gen Electric	26490	10/18/76	Repl. limit sw, adjust breakers	RC	EQ	2	7	15	No
		Gen Electric	27625	12/10/77	Replaced breakers	RC	EQ	3	4	15	Yes
5B SWITCHGEAR		← No Data →									
5C CONTROLS		← No Data →									
5D EXCITER		Gen Electric	EEI-003	1/22/76	Realigned, excessive vibration	RC	EQ	4	60	0	Yes
		Gen Electric	EEI-038	11/18/76	Repl. bearings, grout base plate	RC	EQ	4	436	0	Yes
		Gen Electric	WA sheet	1/19/77	Realigned, excessive vibration	RC	EQ	4	39	0	No
5E TRANSFORMERS		?	17043A	12/9/76	Repaired oil leak	RC	EQ	7	19	0	No
		?	19145A	12/28/76	Changed oil in cooler pump	RC	EQ	5	4	0	No
		?	20886	3/11/77	Replaced circuit breaker	RC	EQ	3	3	0	No
		?	52607	5/12/77	Rewelded leak	RC	EQ	1	3	0	No
		?	22512	5/12/77	Repaired oil pump leak	RC	EQ	1	1	0	No
		?	22140	8/12/77	Repaired cooler oil leak	RC	EQ	2	9	0	No
		?	26285	10/7/77	Replaced oil pump	RC	EQ	4	3	0	No
5F SUBSTATION		← No Data →									
5G ISOLATED PHASE BUS		← No Data →									
5H BATTERIES		← No Data →									

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
5I BATTERY CHARGER	?	?	08211A	2/16/76	Repaired battery charger	RC	EQ	2	4	0	No
	SY-5	?	21398	3/28/77	Cleaned fins, firing module	RC	EQ	2	3	0	No
	?	?	22813	5/25/77	Repaired wire, checkout	RC	EQ	1	3	70	No
	SY-1	?	23168	6/8/77	Loose card in control circuit	RC	EQ	2	4	90	No
	SY-1	?	23287	6/20/77	Cleaned contacts	RC	EQ	2	6	100	No
	SY-5	?	23432	6/29/77	Replaced module, cleaned cont's	RC	EQ	2	6	100	No
	?	?	25027	8/31/77	Replaced voltage relay card	RC	EQ	3	2	0	No
	ICB	?	25945	9/24/77	Adjusted voltage	RC	EQ	2	2	30	No
	ICB	?	29188	12/19/77	Replaced firing module	RC	EQ	2	2	100	No
6 CONTROLS AND INSTRUMENTATION SYSTEMS											
6A CONTROL AND MONITORING EQUIPMENT											
6A1 ICS	?	?	Gray Book	3/13/75	Repaired instrument	RC	EQ	2	6	0	Yes
	?	?	EEI-008	4/22/75	Corrected control malfunction	RC	EQ	2	7	0	Yes
	?	?	EEI-009	4/23/75	High pressure, FW swing	RC	EQ	1	9	0	Yes
	?	?	EEI-012	6/9/75	High RC pressure during restart	RC	EQ	1	10	0	Yes
	?	?	EEI-918	7/13/75	Could not increase load	RC	EQ	1	12	0	Yes
	?	?	EEI-025	8/9/75	Trip during power escalation	ITC	EQ	1	13	0	Yes
	1HP23-DPT1	?	05650A	10/23/75	Recalibrated	ITC	EQ	1	2	100	No
	1WD/80-DPT	?	05646A	10/23/75	Recalibrated	ITC	EQ	1	2	100	No
	?	?	09349A	3/30/76	Checked "A" SG SU valve	ITC	EQ	2	4	0	No
	?	?	EEI-021	7/14/76	Reactor trip during maintenance	RC	EQ	1	4	0	Yes
6A2 NNI	?	?	Gray Book	3/12/75	Repaired steam leak	RC	EQ	2	13	0	Yes
	1FT26-P2	?	05219A	11/25/75	Replaced transmitter	RC	EQ	2	4	100	No
	1RC14A-DPT1	?	06248A	11/26/75	Replaced transmitter amplifier	RC	EQ	2	75	100	No
	1WD64-DPT1	?	06964A	12/17/75	Recalibrated transmitter	ITC	EQ	1	2	100	No
	?	?	Duke	1/8/76	Replaced summer module	RC	EQ	2	10	95	No
	LT16A	?	09134A	3/19/76	Calibrated transmitter	ITC	EQ	1	3	0	No
	?	?	09331A	3/28/76	Checked on A1, A2, and B1 RCP	RC	EQ	2	4	0	No
	?	?	09626A	4/1/76	Replaced static multiplier	RC	EQ	2	8	0	No
	?	?	09628A	4/1/76	Replaced thermocouple leads	RC	EQ	2	8	0	No
	?	?	01584A	4/21/76	Recalibrated transmitter	RC	EQ	1	5	100	No
	?	?	EEI-017	6/27/76	Repaired "E" channel indicator	RC	EQ	2	5	0	Yes
	?	?	EOF	6/30/76	Repaired recorder	RC	EQ	2	3	75	No
	?	?	EEI-019	7/7/76	Valved out during test	RC	EQ	1	8	0	Yes
	?	?	12467A	7/29/76	Added new thermocouple	ITC	EQ	2	6	95	No
	1FT14P	?	10832A	8/11/76	Replaced, calibrated transmitter	RC	EQ	2	9	73	No
	?	?	13468A	9/4/76	Replaced Channel "A" flow xmtr	RC	EQ	2	25	0	No
	?	?	17068A	11/3/76	Replaced line "A" RC pump	RC	EQ	2	6	0	No
	?	Foxboro	20428A	3/26/77	Repaired meter	RC	EQ	2	6	0	No
	?	?	WA sheet	12/28/77	Replaced channel "A" transmitter	RC	EQ	2	13	40	Yes

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
6A3 Incore	?	?	09511A	3/13/76	Repaired monitor	RC	EQ	2	5	0	No
Detectors	?	?	WA sheet	1/18/77	Repaired incore instr. tube leak	RC	EQ	2	102	0	No
			53147	8/29/77	Replaced 32 incore detectors	RC	EQ	4	73	0	No
			53148	9/9/77	Cleaned flanges, inst. seals	RC	EQ	4	8	0	No
6A4 Computers	← Computers →										

6B PLANT PROTECTION EQUIPMENT

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
6B1 NI/RPS	NI-4	?	02321A	5/24/75	Replaced log amplifier	RC	EQ	2	20	100	No
	NI-1	?	08355	2/20/75	Changed/calibrated preamplifeir	RC	EQ	2	60	0	No
	NI-1	?	11846A	7/2/76	Repaired loose wire	RC	EQ	2	21	0	No
	?	?	12466A	7/29/76	Calibrated	ITC	EQ	1	2	100	No
	?	?	WA sheet	6/6/77	NI calib - 25% power hold	ITC	EQ	1	2	25	Yes
	?	?	WA sheet	12/28/77	Ni calib - 60% power hold	ITC	EQ	1	2	60	Yes
6B2 SRCI	← No Data →										
6B3 ESFAS	← No Data →										

7 WASTE HANDLING SYSTEMS

7A *LIQUID WASTE DISPOSAL*

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
Valves	LWD-1	Grinnell	90190	10/22/74	Reset torque switch	RC	EQ	2	4	0	No
			90253	12/24/74	Replaced diaphragm, gasket	RC	EQ	?	6	0	No
	LWD-6	Velan	Duke	2/20/76	Lapped seat, replaced disk	RC	EQ	?	8	0	No
	LWD-22	Grinnell	EEI 77-15/ -16	3/25/77	Repaired valve	RC	EQ	?	4	0	No
	LWD-57	Grinnell	05380	12/30/74	Replaced diaphragm	RC	EQ	2	4	0	No
	LWD-59	Grinnell	05541	12/19/74	Replaced diaphragm, gasket	RC	EQ	2	6	0	No
			07015	12/27/75	Replaced diaphragm	RC	EQ	2	4	100	No
			Duke	5/6/76	Installed new bonnet assembly	RC	EQ	2	2	0	No
	LWD-66	Grinnell	01801	9/9/74	Replaced diaphragm	RC	EQ	2	4	60	No
	LWD-68	Fish-Gov	00361	6/10/74	Clogged valve — cleaned	RC	EQ	2	4	0	No
			05244	10/10/75	Disassembled, cleaned	RC	EQ	2	2	0	No
	LWD-78	Grinnell	25402	9/30/77	Repaired valve	RC	EQ	?	8	0	No
	LWD-89	Fish-Gov	05247	10/15/75	Cleaned, lapped seat	RC	EQ	2	3	100	No
	LWD-106	?	Duke	2/20/76	Lapped seat, replaced disc	RC	EQ	4	6	0	No
	LWD-107	Velan	23430	9/22/77	Lapped seat, cleaned, repl gaskets	RC	EQ	?	8	0	No
	LWD-109	Velan	26157	9/27/77	Lapped seat, replaced gaskets	RC	EQ	?	6	0	No
	LWD-110	Grinnell	26157	9/27/77	Replaced bonnet assembly	RC	EQ	?	6	0	No
	LWD-119	Velan	23430	9/22/77	Lapped seat, cleaned, replaced	RC	EQ	?	8	0	No
	LWD-125	Grinnell	23748	9/1/77	Replaced bonnet assembly	RC	EQ	?	4	0	No
	LWD-129	Grinnell	24457	8/30/77	Repaired rubbing hand wheel	RC	EQ	?	4	0	No
			27604	12/12/77	Replaced stem, diaphragm assembly	RC	EQ	2	4	100	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	LWD-130	Grinnell	25010	8/29/77	Replaced diaphragm	RC	EQ	?	6	0	No
	LWD-132	Velan	LER	7/8/75	Valve failed to close	RC	EQ	?	4	100	No
	LWD-137	Velan	01104A	4/17/75	Inspected, replaced cover plate	RC	EQ	2	8	100	No
	LWD-230	Velan	05286A	3/4/76	Seat leak, replaced valve	RC	EQ	2	4	0	No
			11898A	11/12/76	Lapped, honed disc	RC	EQ	2	5	0	No
	LWD-354	Grinnell	Duke	3/3/76	Replaced diaphragm	RC	EQ	2	4	0	No
	LWD-387	Grinnell	04012	11/2/74	Replaced roll pin	RC	EQ	?	3	0	No
	LWD-755	Grinnell	25403	9/8/77	Replaced bonnet assembly	RC	EQ	2	2	0	No
Evaporator	LWD-EV1	Aqua-Chem	03210A	7/9/75	Installed gasket, cleaned sight glass	RC	EQ	2	3	99	No
			05063	9/16/77	Cleaned strainer, checked valve	RC	EQ	2	2	0	No
			24472	9/27/77	Replaced gasket, flanges	RC	EQ	?	8	0	No
	?	?	24472	8/29/77	Installed new valve	RC	EQ	?	4	0	No
Pumps	LWD-P7	Ing-Rand	06422	1/31/75	Cleaned lines, impeller	RC	EQ	2	8	0	No
			06464	2/3/75	Valved in HP and LP sw.	RC	EQ	1	2	0	No
			06418	2/5/75	Pulled pump, checked bearings	RC	EQ	2	4	0	No
			01115A	4/4/75	Cleaned pump, replaced seals	RC	EQ	2	8	75	No
			01113A	4/7/75	Cleaned pump, replaced seals	RC	EQ	2	8	60	No
			01496A	4/18/75	Installed new gaskets	RC	EQ	2	4	99	No
	LWD-P2A/B	Ing-Rand	23892	8/25/77	Repl. impeller, gaskets, seals	RC	EQ	4	6	0	No
	LWD-P2B	Ing-Rand	80041A	4/2/75	Repacked lower packing gland	RC	EQ	2	8	62	No
	LWD-P3A	Sydnor	12180	7/22/76	Replaced gaskets under "A" pump	RC	EQ	4	16	96	No
			17262	10/21/76	Complete pump overhaul	RC	EQ	4	16	96	No
			17547	10/25/76	Tightened leaking union	RC	EQ	2	2	96	No
			17615	10/26/76	Corrected running rotation	RC	EQ	3	8	96	No
	LWD-P3B	Sydnor	12180	7/22/76	Replaced gaskets under "B" pump	RC	EQ	4	16	96	No
			01030A	4/8/75	Replaced impeller, shaft, bushing	RC	EQ	2	16	63	No
	LWD-P5A	Sydnor	03646A	9/29/75	Welded pump baseplate leaks	RC	EQ	2	4	0	No
	WD-P42	?	01705A	8/2/74	Replaced coupling	RC	EQ	?	4	100	No
			02243A	8/24/74	Installed new coupling	RC	EQ	?	4	0	No
			04953A	9/23/75	Replaced mechanical seals	RC	EQ	2	8	100	No
			24960	8/18/77	Replaced coupling	RC	EQ	2	2	0	No
			24969	8/19/77	Repaired mechanical seal	RC	EQ	3	6	0	No
	WD-P44	Aqua-Chem	01690A	8/1/74	Breaker tripped after 15-min. run	RC	EQ	?	4	100	No
			06485A	12/4/75	Trip on thermal overload	RC	EQ	2	2	75	No
	?	?	25411	9/3/77	Replaced coupling	RC	EQ	2	1	0	No
	?	?	25411	9/3/77	Replaced coupling	RC	EQ	2	1	0	No
7B GASEOUS WASTE DISPOSAL											
Valves	1GWD-2	Grinnell	00880A	3/19/75	Adjusted limit switch	RC	EQ	2	2	75	No
			02588A	6/18/75	Adjusted limit switch	RC	EQ	2	4	98	No
			02579A	6/13/75	Adjusted limit switch	RC	EQ	2	2	98	No
			03608	7/2/75	Adjusted limit switch	RC	EQ	?	4	98	No
			10398A	5/5/76	Replaced solenoid valve	RC	EQ	?	2	0	No
			10901A	5/30/76	Adjusted limit switch	RC	EQ	3	2	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	1GWD-2	Grinnell	10907A	5/9/76	Adjusted limit switch	RC	EQ	?	6	0	No
	(cont'd)		17185A	1/4/77	Installed diaphragm, gasket	RC	EQ	5	6	0	No
	1GWD-3	Grinnell	04934	12/13/74	Repaired solenoid	RC	EQ	2	4	0	No
	1GWD-21	?	13698	9/28/76	Replaced	RC	EQ	2	3	100	No
	1GWD-23	Grinnell	09974	4/14/76	Replaced disc	RC	EQ	?	6	75	No
	GWD-24	Grinnell	04836	9/30/75	Replaced bonnet assembly	RC	EQ	?	2	100	No
	GWD-26	Velan	Duke	4/13/75	Leaking after top pulled	RC	EQ	2	4	100	No
	GWD-27	Grinnell	13175A	8/28/76	Replaced bonnet assembly	RC	EQ	2	4	100	No
	GWD-58	Velan	95240	9/10/75	Replaced w/ Kerotest valve	RC	EQ	2	12	100	No
	GWD-59	Velan	95240	9/10/75	Replaced w/ Kerotest valve	RC	EQ	2	12	100	No
	GWD-78	Fisher	21557	4/7/77	Reset to open at 80 psi	RC	EQ	2	4	100	No
	GWD-79	Fisher	20002	2/11/77	Adjusted controller	RC	EQ	2	2	100	No
			Duke	2/23/77	Checked for normal operation	RC	EQ	2	1	100	No
			22339	4/21/77	Adjusted pressure setting	RC	EQ	2	1	100	No
	GWD-84	Fisher	02240A	6/11/75	Replaced diaphragm	RC	EQ	2	2	90	No
	GWD-85	Fisher	24499	8/4/77	Tightened flanges	RC	EQ	2	2	85	No
	GWD-87	?	24499	8/4/77	Tightened flanges	RC	EQ	2	2	85	No
	GWD-90	Grinnell	02593A	6/17/75	Replaced diaphragm	RC	EQ	2	4	100	No
	GWD-100	Grinnell	09861	4/10/76	Replaced diaphragm	RC	EQ	?	8	72	No
	GWD-153	Velan	24669	8/18/77	Lapped seat, replaced gasket	RC	EQ	?	8	0	No
	GWD-228	Grinnell	19107	1/1/77	Replaced diaphragm	RC	EQ	?	8	100	No
Gas Compressor	WDP-67A	Nash	05742	1/3/75	Checked oil	RC	EQ	?	4	0	No
			03763	8/11/75	Rebuilt pump	RC	EQ	2	8	90	No
			04169	11/21/75	Repaired pump	RC	EQ	?	16	100	No
			10194	4/19/76	Replaced 1/2" nipple	RC	EQ	2	4	0	No
			09846	4/21/76	Tightened leaky flange	RC	EQ	2	8	0	No
			11873	7/9/76	Rebuilt compressor	RC	EQ	?	16	100	No
			12073	7/23/76	Factor rep. check	RC	EQ	?	2	95	No
			25031	9/3/77	Checked erratic cycling	RC	EQ	4	7	0	No
	WDP-67B	Nash	06260	2/1/75	Lapped seat, hydro-set	RC	EQ	2	8	0	No
			01925A	5/21/75	Replaced flange bushing, gasket	RC	EQ	2	4	90	No
			02574	6/7/75	Installed new diaphragm	RC	EQ	2	8	100	No
			06631	12/7/75	Adjusted breaker	RC	EQ	?	4	0/50	No
			11874	7/8/76	Rebuilt compressor	RC	EQ	2	8	90	No
			17623	10/26/76	Installed new seal, adjusted	RC	EQ	2	8	100	No
			17185	1/4/77	Rebuilt relief valve	RC	EQ	2	18	100	No
			19367	1/11/77	Adjustments	RC	EQ	?	4	100	No
			19410	1/12/77	?	RC	EQ	2	2	100	No
			22339	4/21/77	Adjusted pressure setting	RC	EQ	2	1	100	No
			52408	2/23/77	Checkout - nothing found	RC	EQ	2	3	100	No
			23275	6/17/77	Adjusted unload controller	RC	EQ	2	2	100	No
			24685	8/29/77	Repaired controller	RC	EQ	2	2	0	No
Transmitters	1WD183-DPT	Foxboro	09969A	4/13/76	Repaired flow transmitter	RC	EQ	?	4	90	No
			13176A	8/28/76	Replaced diaphragm, recalibrated	RC	EQ	?	19	100	No
	1WD180-DPT	Foxboro	10179A	4/18/76	Repaired pressure transmitter	RC	EQ	?	3	0	No
			08847A	4/21/76	Bleed line to GWD transmitter	RC	EQ	?	3	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
Gas Analyzer		Hays	27641	12/14/77	Reset sicruit breaker	RC	EQ	2	1	100	No
Vent Header		?	12661	8/6/76	Checked pressure gages	RC	EQ	2	1	100	No
		?	23013	5/27/77	Checked pressure gages	RC	EQ	2	3	70	No
		?	23291	6/23/77	Checked pressure gages	RC	EQ	3	3	100	No
		?	23416	6/28/77	Checked water trap petcock	RC	EQ	3	4	100	No
		?	27176	11/4/77	Tightened transmitter nut	RC	EQ	3	2	25	No
7C SOLID WASTE DISPOSAL		←			No Data						→
7D COOLANT STORAGE											
Valves	CS-11	Velan	25042	9/2/77	Cleaned seat, replaced gasket	RC	EQ	?	10	0	No
	CS-20	Velan	22695	9/14/77	Repacked	RC	EQ	2	2	0	No
	CS-46	Grinnell	00186A	2/21/75	Tightened body-to-bonnet bolts	RC	EQ	2	4	90	No
	CS-56	Grinnell	00110A	2/15/75	Replaced diaphragm	RC	EQ	?	4	0	No
			00184	2/21/75	Replaced flexible connector	RC	EQ	?	3	0	No
			12189A	7/26/76	Replaced diaphragm	RC	EQ	2	2	96	No
			22874	5/18/77	Replaced diaphragm	RC	EQ	2	1	0	No
	CS-62	Grinnell	12169A	6/20/76	Replaced diaphragm	RC	EQ	2	2	98	No
	CS-65	Grinnell	23285	8/17/77	Replaced diaphragm	RC	EQ	?	4	0	No
	CS-66	Grinnell	EEI-77-31	7/5/77	Replaced bonnet, diaphragm	RC	EQ	?	4	0	Yes
	CS-69	Grinnell	24129	8/25/77	Replaced diaphragm	RC	EQ	?	4	0	No
	CS-70	Grinnell	25934	9/20/77	Replaced diaphragm	RC	EQ	?	6	0	No
	CS-72	Grinnell	20430	3/6/77	Repaired	RC	EQ	?	6	0	No
	CS-85	Grinnell	EEI-77-15/ -16	3/25/77	Repaired valve	RC	EQ	?	4	0	No
	CS-89	Grinnell	23851	9/8/77	Replaced diaphragm	RC	EQ	?	4	0	No
	CS-100	Grinnell	25410	9/14/77	Replaced diaphragm	RC	EQ	2	2	0	No
	CS-173	Grinnell	24283	9/8/77	Repalced diaphragm	RC	EQ	?	4	0	No
Pumps	1WD-P21A	Ing-Rand	22691	9/7/77	Replaced seal	RC	EQ	?	12	0	No
			26244	10/2/77	Corrected motor wiring	RC	EQ	2	4	0	No
			23959	7/15/77	Replaced mechanical seals	RC	EQ	3	4	86	No
7E COOLANT TREATMENT											
Valves	CT-20	VAREC	24472	8/2/77	Repaired leaking valve	RC	EQ	2	4	80	No
	CT-22	Grinnell	05543	12/28/74	Removed, replaced compressor pin	RC	EQ	?	4	0	No
			01308	4/8/75	Replaced rubber diaphragm	RC	EQ	?	8	60	No
	CT-28	Fish-Gov	00517	6/18/74	Disassembled, cleaned	RC	EQ	2	4	0	No
			02411	9/6/74	Removed trash from gate	RC	EQ	?	8	100	No
			02963	9/13/74	Removed, unclogged	RC	EQ	?	8	90	No
			03520	10/11/74	Removed, cleaned	RC	EQ	?	8	85	No
			04346	11/26/74	Cleaned valves, lines	RC	EQ	?	8	0	No
			04499	11/25/74	Removed, cleaned	RC	EQ	?	8	0	No
			05365	12/31/74	Unclogged, replaced gasket	RC	EQ	?	8	0	No
			06359	1/29/75	Removed foreign metal from valve	RC	EQ	?	4	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time: No. men	Repair time: Clock hours	Actual plant power, %	Did event force or extend outage?
	CT-28 (cont'd)	Fish-Gov	06673	2/16/75	Unclogged, cleaned	RC	EQ	2	2	0	No
			00432A	3/6/75	Cleaned, replaced gasket	RC	EQ	2	4	0	No
			00821A	3/18/75	Removed foreign material	RC	EQ	?	4	75	No
			01305A	4/3/75	Unclogged lines, valve	RC	EQ	?	8	65	No
			02003A	5/11/75	Unclogged valve	RC	EQ	2	4	100	No
			02371A	5/28/75	Removed trash from valve	RC	EQ	?	8	100	No
			03200A	7/9/75	Pulled valve, cleaned	RC	EQ	?	8	100	No
			03380A	7/16/75	Removed trash, cleaned	RC	EQ	?	8	100	No
			04380A	9/1/75	Cleaned trash from valve	RC	EQ	?	8	100	No
			04951A	9/23/75	Replaced diaphragm, cleaned	RC	EQ	2	2	100	No
			03251A	10/9/75	Disassembled, cleaned	RC	EQ	2	2	100	No
			05291A	10/15/75	Removed trash	RC	EQ	?	8	100	No
			05617A	10/17/75	Cleaned valve, nozzle	RC	EQ	2	2	100	No
			05906A	10/31/75	Cleaned valve	RC	EQ	?	4	100	No
			03072	7/6/75	Problem with instrument PT-28	RC	EQ	3	1	100	No
	CT-39	Ladish	24083	7/22/77	Removed boron	RC	EQ	2	1	90	No
	CT-40	Fish-Gov	05558	10/20/75	Cleaned, unclogged lines	RC	EQ	?	4	90	No
	CT-46	Aqua-Chem	24263	7/25/77	Cleaned, reassembled	RC	EQ	2	3	90	No
	CT-48	Velan	04381	9/1/75	Replaced diaphragm	RC	EQ	?	4	100	No
	CT-49	Grinnell	04494	12/1/74	Repaired flange leak	RC	EQ	?	4	0	No
			05366	1/1/75	Replaced bad diaphragm	RC	EQ	?	8	0	No
			02586A	7/4/75	Replaced diaphragm	RC	EQ	2	4	90	No
			04381A	9/1/75	Replaced diaphragm	RC	EQ	?	8	100	No
			05553A	10/19/75	Replaced diaphragm	RC	EQ	2	2	85	No
	CT-52	Grinnell	04849A	9/23/75	Replaced diaphragm, stem	RC	EQ	2	4	100	No
			?	10/1/74	Disassembled, inspected, will not close	RC	EQ	?	4	45	No
			00688A	3/13/75	Replaced diaphragm	RC	EQ	?	8	40	No
			01074A	3/28/75	Replaced diaphragm	RC	EQ	?	8	90	No
	CT-53	Grinnell	01962	8/7/74	Won't operate	RC	EQ	2	3	100	No
			00688A	3/13/75	Replaced diaphragm	RC	EQ	?	8	40	No
			00921A	3/26/75	Replaced diaphragm	RC	EQ	?	8	90	No
			02464A	6/7/75	Replaced compression pin	RC	EQ	?	4	100	No
			04164A	8/19/75	Repl. bushings, stem, diaphragm,	RC	EQ	?	8	100	No
			03896A	8/11/75	Rebuilt valve	RC	EQ	?	8	90	No
			?	9/23/75	Replaced bonnet, diaphragm	RC	EQ	2	2	100	No
			?	4/23/76	Installed new valve	RC	EQ	?	8	0	No
	CT-54	Grinnell	00522	6/18/74	Disassembled, repaired	RC	EQ	2	2	0	No
			00984	6/28/74	Replaced seat set screw	RC	EQ	?	4	0	No
			04500A	9/23/75	Replaced diaphragm	RC	EQ	?	8	100	No
	CT-55	Grinnell	02391	9/1/74	Changed valve	RC	EQ	?	2	100	No
			02438A	6/7/75	Replaced diaphragm	RC	EQ	?	8	0	No
			04904A	9/23/75	Replaced diaphragm	RC	EQ	?	8	100	No
			05180A	10/4/75	Replaced compression pin	RC	EQ	?	4	100	No
			24291	7/28/77	Replaced stem pin, bonnet	RC	EQ	2	2	90	No
	CT-65	Grinnell	00123A	?	Replaced diaphragm	RC	EQ	?	8	?	No
			06256A	?	Replaced diaphragm	RC	EQ	2	4	?	No
	CT-75	Grinnell	03826	11/7/74	Installed new stem, diaphragm	RC	EQ	2	4	0	No
			03528	11/13/74	Wouldn't operate	RC	EQ	2	2	0	No

System/component	Mark No.	Manufacturer	Source of info	Date	Repair	Work category	Cause category	Repair time No. men	Repair time Clock hours	Actual plant power, %	Did event force or extend outage?
	CT-87	Grinnell	00240A	2/28/75	Put counterweight on hand wheel	RC	EQ	?	8	0	No
			00247A	3/3/75	Replaced diaphragm	RC	EQ	?	8	0	No
	CT-89	Grinnell	06606	2/12/75	Repaired seat leak	RC	EQ	2	3	0	No
Piping	NA	?	25065	9/16/77	Opened clogged line	RC	EQ	5	2	0	No
	NA	?	80332	11/4/75	Opened clogged line	RC	EQ	2	8	100	No
Evaporator	?	Aqua-Chem	02060	5/16/75	Pressure indicator plugged	RC	EQ	2	4	100	No
			25948	9/27/77	Welded pin hole in tank	RC	EQ	?	6	0	No
			80332A	11/4/75	Opened clogged sample line	RC	EQ	2	8	100	No
			05746A	12/17/75	Cleaned evaporator	RC	EQ	9	40	100	No
			25065	9/16/77	Inspected for blockage	RC	EQ	5	2	0	No
			05550A	10/20/75	Unclogged cooler	RC	EQ	?	8	90	No
Pumps	WD-P41	?	05550A	10/20/75	Installed new mechanical seal	RC	EQ	?	8	90	No
			25060	9/16/77	Replaced shaf, seal, brg, cplg	RC	EQ	5	16	0	No
			01741A	4/27/75	Replaced coupling	RC	EQ	?	8	100	No
			05641A	10/22/75	Replaced coupling	RC	EQ	?	8	100	No
			06122A	11/12/75	Replaced coupling	RC	EQ	?	8	100	No
			?	11/13/75	Replaced coupling, aligned pump	RC	EQ	?	8	100	No
	?	?	07349A	1/10/75	Replaced impeller	RC	EQ	?	24	0	No
	?	?	06662A	12/14/75	Replaced motor, pump	RC	EQ	2	8	100	No
	?	?	07707A	1/29/76	Checked motor - was OK	RC	EQ	2	2	100	No
	?	?	?	2/7/76	Repaired broken leads	RC	EQ	?	8	100	No
	?	?	08127A	2/11/76	Replaced shorted leads	RC	EQ	?	8	0	No
	?	?	?	2/19/76	Replaced motor, pump	RC	EQ	2	4	0	No
	?	?	?	3/22/76	Motor was grounded	RC	EQ	?	8	0	No
	?	?	?	3/22/76	Replaced motor, pump	RC	EQ	?	16	0	No
	?	?	?	4/29/76	Rebuilt pump	RC	EQ	2	8	0	No
8 OTHER											
8A POLAR CRANE	NA	Whiting	Duke	3/26/77	PM on hoist, crane	PM	EQ	2	64	0	No
			52612-1	8/8/77	Preventive maintenance	PM	EQ	2	8	0	No
			52612	8/8/77	Replaced speed control power supp.	RC	EQ	2	20	0	Yes

APPENDIX F

Refueling Work Activities

The following tables give the detailed refueling activities data that were collected from Oconee 1, Oconee 3, Rancho Seco, and Three Mile Island Unit 1. A discussion of these data is given in section 4.3 along with recommendations for improving availability. Table 4-4 summarizes the data in order of the limiting factor for refueling values that were calculated. The methodology for these calculations is discussed in Appendix C, paragraph 2.3.


UTILITY: METROPOLITAN [redacted]SON COMPANY
STATION: TMI-1
LOCATION: THREE MILE [redacted] PENN
APPROX PERFORMANCE CLOCK TIME (HOURS)
MAR 18, 1977
1. SHUTDOWN STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING 28
5. REACTOR COOLANT PUMP SEAL REMOVAL 174
6. REMOVE SHIELD BLOCKS
7. CLEAN TRANSFER CANAL 28
8. REMOVE/INSTALL INCORE DETECTORS 60*
9. REMOVE/INSTALL RV HEAD INSULATION 12
10. INSTALL/REMOVE CANAL SEAL PLATE 20
11. DETENTION/RETENTION RV HEAD 34
12. SECURE/REINSTALL CRDM's 52
13. RV HEAD REMOVAL/REINSTALL 18
14. PLENUM REMOVAL/REINSTALL 28
15. INSTALL/REMOVAL STUD HOLE PLUGS 18
16. FILL/DRAIN TRANSFER CANAL 50
17. CHECKOUT FUEL HANDLING EQUIPMENT* 30*
18. REFUELING OPERATIONS 144
19. STEAM GENERATOR TUBE INSPECTION & REP. 316
20. EXERCISE VENT VALVES 1 5
21. CONTAINMENT LEAK TESTS 102
22. PHYSICS TESTS
[redacted]
DURING THE CHECK OUT OF THE CIRCULATING WATER SYSTEM, THAT SYSTEM's "D" PUMP CASING RUPTURED CREATING A FLOOD CONDITION IN THIS PUMP ROOM. FROM APRIL 19 THRU MAY 4, MAJOR EMPHASIS WAS PLACED ON REPAIR AND DRYING OUT OF THIS EQUIPMENT.
UTILITY: DUKE POWER COMPANY
STATION: OCONEE III
LOCATION: SENECA, S.C.
APPROX PERFORMANCE CLOCK TIME (HOURS)
OCT 22, 1977
1. SHUTDOWN STARTUP 200
2. REACTOR BUILDING PURGE n. dat
3. HEALTH PHYSICS SURVEY n. dat
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING 24
5. REACTOR COOLANT PUMP SEAL REMOVAL 208
6. REMOVE SHIELD BLOCKS 26
7. CLEAN TRANSFER CANAL -
8. REMOVE/INSTALL INCORE DETECTORS 90*
9. REMOVE/INSTALL RV HEAD INSULATION 10
10. INSTALL/REMOVE CANAL SEAL PLATE 18
11. DETENTION/RETENTION RV HEAD 64
12. SECURE/REINSTALL CRDM's 62
13. RV HEAD REMOVAL/REINSTALL 28
14. PLENUM REMOVAL/REINSTALL 20
15. INSTALL/REMOVAL STUD HOLE PLUGS 11
16. FILL/DRAIN TRANSFER CANAL 28
17. CHECKOUT FUEL HANDLING EQUIPMENT* 15*
18. REFUELING OPERATIONS 204
19. STEAM GENERATOR TUBE INSPECTION & REP. 96*
20. EXERCISE VENT VALVES n. dat
21. ARIS INSPECTION 44
22. PHYSICS TESTS INCOMPLETE
23. TURBINE
*NON-CRITICAL ACTIVITIES
FILL & VENT RCS
ACTIVITY
1. SHUTDOWN STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL
6. REMOVE SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENTION/RETENTION RV HEAD
12. SECURE/REINSTALL CRDM's
13. RV HEAD REMOVAL/REINSTALL
14. PLENUM REMOVAL/REINSTALL
15. INSTALL/REMOVAL/STUD HOLE PLUGS
16. FILL/DRAIN TRANSFER CANAL
17. CHECKOUT FUEL
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP.
20. EXERCISE VENT VALVES
21. CONTAINMENT LEAK TESTS
22. PHYSICS TESTS
DWN BY UNDERWOOD
CHKD BY
PASSED BY
APPVD BY
STD. SPL. DATE
SHEET 1 OF 2
DO NOT SCALE USE DIMENSIONS ONLY
REFUELING ACTIVITY CHART
THREE MILE ISLAND I
OCONEE III
SCALE
THIS DRAWING IS THE PROPERTY OF THE BABCOCK & WILCOX CO POWER GENERATION GROUP
DWG NO 1004375 F
REV 3-2 79

UTILITY SMUD
STATION RANCHO SECO
LOCATION SACRAMENTO, CALIFORNIA

ACTIVITY	ACTIVITY	DELAYS DIRECT	DELAYS INDIRECT	TOTAL
1. SHUTDOWN / STARTUP		4	70	275
2. REACTOR BUILDING PURGE		-		74
3. HEALTH PHYSICS SURVEY		-		2
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING	1.		59	184
5. REACTOR COOLANT PUMP SEAL REMOVAL/REPLACE	191	53	-	244
6. REMOVE/INSTALL SHIELD BLOCKS	5	-	-	5
7. CLEAN TRANSFER CANAL	63	31.5		94.5
8. REMOVE/INSTALL INCORE DETECTORS	24	0	0	24
9. REMOVE/INSTALL RV HEAD INSULATION	12.5		26	38.5
10. INSTALL/REMOVE CANAL SEAL PLATE	14			14
11. DETENTION/RETENTION RV HEAD	48.5	15	27.5	91
12. SECURE/REINSTALL CRDM's	88	34.5	40	162.5
13. RV HEAD REMOVAL/REINSTALL	21.5	8.5	147	177
14. PLENUM REMOVAL/REINSTALL	17.5		7	24.5
15. INSTALL/REMOVAL/STUD HOLE PLUGS	8		-	8
16. FILL/DRAIN TRANSFER CANAL	24.5		0	24.5
17. CHECKOUT FUEL HANDLING EQUIPMENT	137	159	6	302
18. REFUELING OPERATIONS	225.5	19	12.5	257
19. STEAM GENERATOR TUBE INSPECTION & REP.	29		-	29
20. EXERCISE VENT VALVES	2	-	-	2
21. CONTAINMENT LEAK TESTS	131	40	-	171
22. PHYSICS TESTS	189	43	-	232

* NON-CRITICAL PATH ACTIVITIES
** ONLY CRITICAL PATH TIME

AUG 20 1977

EVENTS
a. START RX SHUTDOWN FROM 91% POWER
b. RX AT 15% END OF SHUTDOWN PERIOD
c. ESTABLISHED RX COOLDOWN
d. STARTED DECAY HEAT SYSTEM
e. RCP'S SECURED

DELAYS
1. DELAY IN COOLDOWN DUE TO

REFUELING ACTIVITIES

LEGEND

CRITICAL OR NEAR CRITICAL PATH — DELAYS / EQUIPMENT / OPERATING PRACTICE / REGULATORY

EVENTS a. FUEL LOADING

DWN BY UNDERWOOD
CHKD BY
PASSED BY
APPVD BY
STD SPL DATE
SHEET 1 OF 3
DO NOT SCALE. USE DIMENSIONS ONLY
SCALE

REFUELING ACTIVITY CHART
OCONEE I
PROJECTED STANDARD
RANCHO SECO

DWG NO 1004376 F REV 32 79

[redacted]

[redacted]

Date	Delays	Events
12		a. RCS HEATUP
13	1. BALANCE RCP-1B2	a. RCS HEATUP
14	1. CRD TRIP TEST DELAY 2. VALVE MISALIGNMENT	a. COMMENCED APPROACH TO CRITICALITY b. REACTOR CRITICAL c. INSTRUMENT CALIBRATIONS
15	1. ALIGNMENT OF VALVES DELAYED BORON ADDITION	a. BORATING RCS FOR ALL REGULATING RODS OUT TESTS. b. BORON CONCENTRATION DETERMINED FOR ALL RODS OUT. c. ROD WORTH TESTS d. INSTRUMENT CALIBRATIONS
16	1. VERIFIED CRD GRP 6 WIRING CONNECTION.	a. CONTINUE ROD WORTH TESTS b. VERIFIED CRD GRP 6 WIRING CONNECTION
17	1. HEAT BALANCE ERROR CONNECTION. 2. GENERATOR EXCITER BREAKER (COULD NOT CLOSE).	a. POWER ESCALATION b. NI CAL c. POWER @ 13% d. EXERCISE RODS TO VERIFY COUPLING
18	1. LOSS FW PUMP-HIGH REACTOR PRESSURE TRIP	a. ELEC. GENERATOR ON-LINE SMALL TILT NOTED. b. REACTOR TRIP, LOSS FW PUMP c. REACTOR CRITICAL-INCREASING POWER TO 40%
19		a. RECORDING TILT DATA WHILE XENON BUILDS IN. b. REACTOR @ 40% POWER
20	1. WAITING FOR XENON STABILITY	
21	1. WAITING FOR XENON STABILITY	a. ROD GROUP TESTS b. POWER REDUCTION TO 30%
22	1. WAITING FOR XENON STABILITY & PREPARING FOR TILT TESTS.	
23	1. PREPARING FOR TILT TESTS	a. TILT TESTS
24		a. TILT TESTS
25		a. TILT TESTS b. POWER ESCALATION TO 40% F.P. c. 40% POWER TESTS
26		a. 40% POWER TESTS
27		a. 40% POWER TESTS
28		a. 40% POWER TESTS b. POWER ESCALATION TO 75% F.P.
29		a. POWER ESCALATION TO 75% FP b. REACHED 75% F.P.

UTILITY DUKE POWER COMPANY
STATION OCONEE I
LOCATION SENECA

ACTIVITY

1. SHUTDOWN - STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL/REPLACE
6. REMOVE/INSTALL SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENSION/RETENSION RV HEAD
12. SECURE/REINSTALL CRDM'S
13. RV HEAD REMOVAL/REINSTALL
14. PLENUM REMOVAL/REINSTALL
15. INSTALL/REMOVAL STUD HOLE PLUGS
16. FILL/DRAIN TRANSFER CANAL
17. CHECKOUT FUEL HANDLING EQUIPMENT
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP
20. EXERCISE VENT VALVES
21. AREA INSPECTION
22. PHYSICS TESTS

PROJECTED STANDARD

ACTIVITY

1. SHUTDOWN - STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL/REPLACE
6. REMOVE/INSTALL SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENSION/RETENSION RV HEAD
12. SECURE/REINSTALL CRDM'S
13. RV HEAD REMOVAL/REINSTALL
14. PLENUM REMOVAL/REINSTALL
15. INSTALL/REMOVAL STUD HOLE PLUGS
16. FILL/DRAIN TRANSFER CANAL
17. CHECKOUT FUEL HANDLING EQUIPMENT
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP
20. EXERCISE VENT VALVES
21. CONTAINMENT LEAK TESTS (3rd YEAR)
22. PHYSICS TESTS

NON CRIT

STATION RANCHO SECO
LOCATION SACRAMENTO, CALIFORNIA

ACTIVITY

1. SHUTDOWN - STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL/REPLACE
6. REMOVE/INSTALL SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENSION/RETENSION RV HEAD
12. SECURE/REINSTALL CRDM'S
13. RV HEAD REMOVAL/REINSTALL
14. PLENUM REMOVAL/REINSTALL
15. INSTALL/REMOVAL STUD HOLE PLUGS
16. FILL/DRAIN TRANSFER CANAL
17. CHECKOUT FUEL HANDLING EQUIPMENT
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP
20. EXERCISE VENT VALVES
21. CONTAINMENT LEAK TESTS
22. PHYSICS TESTS

Date	Events	Delays
27	TO RECHECK ... PUMP VENDOR ... TO RE-BALANCE ...	
28	a. FILLED OTSG-B DRAIN WATER b. INSTALLED ... RCP OIL ... c. REPLACED 2 CRDM PI TUBES	
29	a. CONTINUED OIL PAN INSTALLATION	
30	a. SYSTEM HEATUP UNDERWAY	
31	a. RCS HEATUP b. REACTOR CRITICAL c. CONTROL ROD DRIVE TESTS	1. DELAYS DUE TO HIGH RCS LEAKAGE & STICKING CRDM'S
NOV 1		1. DELAYS DUE TO HIGH RCS LEAKAGE AND STICKING CRDM'S
4	a. ZERO POWER TESTS	
5	a. ZERO POWER TESTS	
6	a. COMPLETED ZERO PHYSICS TESTS b. ESCALATING POWER TO 40% c. CLOSED GRID BREAKER d. REACHED 15% F.P.	
7	a. CONTINUING POWER ESCALATION TO 40% b. REACHED 40% F.P. c. HOLD FOR XENON STABILITY	
8	a. 40% POWER TESTS b. HOLD FOR XENON STABILITY	
9	a. 40% FP TESTS	
10	a. 40% FP TESTS b. ESCALATING POWER TO 75%	
11	a. ESCALATING TO 75% F.P. b. 75% F.P. TESTS	
12	a. ESCALATING TO ... F.P. b. 75% F.P. ...	1. STEAM EXTRACTION LINE LEAK 2. DECREASE POWER TO 15% TO REPAIR LEAK
13	a. 92% F.P. b. 5 HOUR HOLD c. ESCALATING POWER d. REACTOR @ 100% F.P.	

REFUELING ACTIVITIES

LEGEND

	DELAYS	EQUIPMENT	OPERATING PRACTICE
CRITICAL OR NEAR CRITICAL PATH			
NON-CRITICAL PATH			

REFUELING ACTIVITY CHART
OCONEE I
PROJECTED STANDARD
RANCHO SECO


UTILITY METROPOLITAN EDISON COMPANY
STATION TMI-[1]
LOCATION THREE MILE ISLAND PENN
1. SHUTDOWN / STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL
6. REMOVE SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE INSTALL INCORE DETECTORS
9. REMOVE INSTALL RV HEAD INSULATION
10. INSTALL REMOVE CANAL SEAL PLATE
11. DETENTION RETENTION RV HEAD
12. SECURE REINSTALL CRDM's
13. RV HEAD REMOVAL REINSTALL
14. PLENUM REMOVAL REINSTALL
15. INSTALL REMOVAL STUD HOLE PLUGS
16. FILL DRAIN TRANSFER CANAL
17. CHECKOUT FUEL HANDLING EQUIPMENT
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP
20. EXERCISE VENT VALVES
21. [CONTAINMENT] LEAK TESTS
22. PHYSICS TESTS
DURING THE CHECK OUT OF THE [CIRCULATING] WATER SYSTEM THAT SYSTEM'S "D" PUMP [CASING] RUPTURED CREATING A FLOOD CONDITION IN THIS PUMP ROOM FROM APRIL 19 THRU MAY 4. MAJOR EMPHASIS WAS PLACED ON REPAIR AND DRYING OUT OF THIS EQUIPMENT
MAY 4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
JUNE 1
2
[redacted]
6
SNUBBER REPAIR DELAY
RCS VENTED
SNUBBER REPAIR & INSPECTIONS
DELAY
POWER TO 15%
POWER TO 40%
XENON HOLD
POWER TO 75%
XENON HOLD
90% FP
XENON HOLD
POWER TO 100%
SECURING LOCA RESTRAINTS
DELAY
ZERO POWER TESTS
T/G
XENON HOLD
XENON HOLD
IMBALANCE TESTS
XENON HOLD
UTILITY DUKE POWER COMPANY
STATION OCONEE III
LOCATION SENECA S C
1. SHUTDOWN STARTUP
2. REACTOR BUILDING PURGE
3. HELATH PHYSICS SURVEY
4. MOVE EQUIPMENT IN OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL
6. REMOVE SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENTION/RETENTION RV HEAD
12. SECURE/REINSTALL CRDM's
13. RV HEAD REMOVAL/REINSTALL
14. PLENUM REMOVAL/REINSTALL
15. INSTALL/REMOVAL/STUD HOLE PLUGS
16. FILL/DRAIN TRANSFER CANAL
17. CHECKOUT FUEL HANDLING EQUIPMENT
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP
20. EXERCISE VENT VALVES
21. ARIS INSPECTION
22. PHYSICS TESTS
NOV 25
26
27
28
29
30
DEC 1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
RCS HEATUP
RCS CLEAN UP
RCS HEATING
DEBORATE
SEC CHEMISTRY
HOLD
NI OVERLAP
PE TO 15% FP
PR
COOL DOWN
DRAIN
COUPLED APSR ROD
REPLACED 'O' RINGS
POWER 36% TO 40%
POWER ESC TO 75% FP
XENON HOLD
POW CAC TO 90% FP
ESCALATION TO 100% POWER
ROD DROP TEST
ZERO POWER TEST
REPLACED STATIC
DELAY
ZPPT
ROD LATCH
VERIFICATION
ROD FOUND UNCOUPELED
MANUAL TRIP TEST
ZERO POWER PHYSICS TEST
40% EDRE
PW DIST TEST
PIDC TEST
75% FP TEST
ACTIVITY
1. SHUTDOWN / STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL
6. REMOVE SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENTION/RETENTION RV HEAD
12. SECURE/REINSTALL CRDM's
13. RV HEAD REMOVAL/REINSTALL
14. PLENUM REMOVAL/REINSTALL
15. INSTALL/REMOVAL/STUD HOLE PLUGS
16. FILL/DRAIN TRANSFER CANAL
17. CHECKOUT FUEL
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP.
20. EXERCISE VENT VALVES
21. CONTAINMENT LEAK TESTS
22. PHYSICS TESTS
DWN BY UNDERWOOD
CHKD BY
PASSED BY
APPVD BY
STD
SPL
DATE
SHEET 2 OF 2
DO NOT SCALE USE DIMENSIONS ONLY
REFUELING ACTIVITY CHART
THREE MILE ISLAND
OCONEE III
SCALE
THE BABCOCK & WILCOX CO
POWER GENERATION GROUP
DWG NO
1004375 F
3-2
79


[redacted]
UTILITY DUKE POWER COMPANY
STATION OCONEE I
LOCATION SENECCA
ACTIVITY
1. SHUTDOWN / STARTUP
2. REACTOR BUILDING PURGE
3. HEALTH PHYSICS SURVEY
4. MOVE EQUIPMENT IN/OUT REACTOR BUILDING
5. REACTOR COOLANT PUMP SEAL REMOVAL/REPLACE
6. REMOVE/INSTALL SHIELD BLOCKS
7. CLEAN TRANSFER CANAL
8. REMOVE/INSTALL INCORE DETECTORS
9. REMOVE/INSTALL RV HEAD INSULATION
10. INSTALL/REMOVE CANAL SEAL PLATE
11. DETENTION/RETENTION RV HEAD
12. SECURE/REINSTALL CRDM's
13. RV HEAD REMOVAL REINSTALL
14. PLENUM REMOVAL REINSTALL
15. INSTALL REMOVAL STUD HOLE PLUGS
16. FILL DRAIN TRANSFER CANAL
17. CHECKOUT FUEL HANDLING EQUIPMENT
18. REFUELING OPERATIONS
19. STEAM GENERATOR TUBE INSPECTION & REP
20. EXERCISE VENT VALVES
21. ARIS INSPECTION
22. PHYSICS TESTS
g. MOVING IN LAST ROLL OF INCORES TO TANK
h. INSTALL LP SHELL BOLTS & COUPLING BOLTS
PROJECTED STANDARD
DAY 35
DAY 36
EVENTS
a. RCS PRESSURIZED TO 225 PSI FOR SYSTEM LEAK TESTS
b. TEST RUN RCP TO CHECK VIBRATION C PUMP NOT RUN DUE TO MISSING INSTRUMENTATION
EVENTS
a. SYSTEM PRESSURIZED TO CHECK LEAKS
b. RETAN RCP'S A B &D TO CHECK VIBRATION
c. TESTED CRDM-FOUND THREE STUCK RODS
d. USED AUX POWER TO FREE STUCK RODS
EVENTS
a. SYSTEM PRESSURIZED LOW PRESSURE FOR TESTING
b. BINGHAM PUMP CHECKED VIBRATION AND CONCLUDED RCP BALANCING OK
c. CRDM TESTING CONTINUING
DWN BY UNDERWOOD
CHKD BY
PASSED BY
APPVD BY
STD SPL DATE
REFUELING ACTIVITY CHART
OCONEE I
PROJECTED STANDARD
RANCHO SECO
THE BABCOCK & WILCOX CO
POWER GENERATION GROUP
DWG NO
REV
32

APPENDIX G

Valve Repair Data

The following tables give the detailed key valve vendor information for Oconee 1 and Rancho Seco and summarize the work request information obtained at these two plants. This information is discussed in section 4.4 along with recommendations for improving availability (Table 4-4).

Table G-1. Valve Failure/Repair Data From Rancho Seco

B&W MARK #	SMUD MARK #	VALVE NAME	MANUF.	TYPE	SIZE IN	TYPE OPERATOR	OUTAGES DUE TO THIS VALVE: DATE	DURATION	CAUSE	WORK REQUEST: COMPLETE DATE	WR#	MNHR	CREW	WORK PERFORMED
RC-V1	PV-21509	PRESSURIZER SPRAY CONTROL	Velan	Gate	2½	Limitorque SMB-2-40	5/21/77	22	S.D. For Rep.	2/5/77	001727	5	2	Temp. Repl. Opr.
							SMUD Reports that			3/31/75	002234	4	3	Install New Opr.
							Pressurizer spray control			4/3/75	001684	40	4	Rewound Motor
							valves have delayed start-			6/15/75	003310	1	2	Install New Drive Bushing
							ups due to Body to Bonnet			1/2/76	003320	8	2	Rewound Motor For OPR.
							leakage and motor opr.			8/31/76	014438	1	2	Exercised Valve
							problems. No records were			6/1/77	019634	27	3	Retested Valve
							found to confirm this.			6/7/77	019027	30	3	Rebuilt Opr.
										10/21/77	020323	8	2	Repaired Valve Reset Timing
										11/1/77	026387	8	2	Repl. Bonnet Gasket
RC-V6	PV-21520	PRESSURIZER SPRAY CONTROL	Velan	Gate	2½	Limitorque SMB-2-40	5/21/77	23	S.D. For Rep.	9/24/75	006403	16	2	Comp. DCN A-777
							≈5/9/78	Shutdown to re-		3/14/77	019208	3	2	DCN-A1652, Rem. Interlocks
								pair spray		6/1/77	019635	9	3	Retest Valve
								control valve		8/30/77	025862	1	2	EID-13099, Valve OPR. Normly
										10/8/77	020321	1	1	Tighten Nut, Insp. Opr.
										10/18/77	021639	2	1	Chk. Opr., Stroked
										10/31/77	021960	1	1	Stroked & Chk out
N.A.	RCS-005	PRESSURIZER SPRAY CONTROL	Control	Globe	½	Manual	SMUD reports			5/25/77	020856	160	10	Install Cap over valve
		BYPASS	Components				that a Body to Bonnet leak							
							delayed a startup							
							about 5/21/77							
RC-V3	HV-21510	PRESSURIZER SPRAY CONTROL	Velan	Gate	2½	Limitorque SMB-00-10	No evidence that this			NO WORK REQUEST FOUND FOR THIS VALVE				
		BLOCK					Valve has caused an outage							
							or delayed a startup.							

Table G-1. (Cont'd)

B&W MARK #	SMUD MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPERATOR	OUTAGES DUE TO THIS VALVE: DATE	DURATION	CAUSE	WORK REQUEST DATA: DATE	WR#	MNHR	CREW	WORK PERFORMED
RC-RV2	PSV-21511	PZR POWER RELIEF	Dresser	Power Actuated	2½"	Electromatic Actuated	Leakage Past Seat has			11/5/74	060393	12	2	Lapped Seats
				Relief			caused Valve to be iso-			4/10/75	03120	30	3	Lapped Seats, Instl. Solenoid
							lated during oper. This			4/21/75	03363	2	1	Recalibrate
							has resulted in RX trip			9/12/75	06160	8	2	Repaired Solenoid
							during some transients			11/4/75	06184	6	2	EID #22283
							due to high RCS Press			12/12/75	03411	20	2	Machined Seat
										12/22/75	07817	38	4	Lapped Seat
										9/7/76	04774	1	1	Tested
										10/8/77	024826	1	1	Install Man. Dump Control
										10/11/77	018739	20	2	Valve Repaired
										10/15/77	022088	6	2	Disconnect for Maintenance
RC-V2	HV-21505	PZR POWER RELIEF BLOCK	Velan	Gate	2½"	SMB-00-10	No Evidence that this			2/18/75	002148	1	2	Tighten Packing
							Valve has caused a			2/17/75	003227	1	1	Adj. Packing
							Shutdown or Delayed a			9/8/76	014458	1	1	Adj. Packing
							Startup.			11/8/76	015669	4	1	Adj. Packing, Restroked
RC-RV 1A	PSV-21506	PZR CODE RELIEF	Dresser	Relief	3"	--	No Evidence that this			8/25/75	005709	120	3	Plug & Seal Lapped
				VLV			Valve has caused a shut-			10/11/77	015564	60	2	Repair & Adj.
							down or Delayed a Start							Setpoint
							up.							
RC-RV 1B	PSV-21507	PZR CODE RELIEF	Dresser	Relief	3"	--	No Evidence that this			NO WORK REQUEST RECORDED				
				VLV			Valve has cause a shut-			AGAINST THIS ITEM.				
							down or Delayed a Start							
							up.							

Table G-1. (Cont'd)

B&W MARK #	SMUD MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPERATOR	OUTAGE DUE TO THIS VALVE			WORK REQUEST DATA				
							DATE	DURATION	CAUSE	DATE	WR#	MNHR	CREW	WORK PERFORMED
---	(Gas) HV-21515	PZR SAMPLE BLOCK	Velan	Globe	1"	Limitorque SMB-000-5	Has not caused an outage			6/17/75	003228	1	1	Adj. Packing
							or delay			11/25/75	007793	2	1	Adj. Packing & Stroked
										1/12/76	008083	1	1	Adj. Packing & Stroked
										1/22/76	010376	6	2	Adj. Packing & Stroked
										4/26/76	012766	1	1	Adj. Packing & Stroked
										8/25/76	014306	2	2	Stroked VLV
										9/8/76	014459	1	1	Adj. Packing
										11/19/76	15915	1	1	Insp. & Clean OPR, Valve overheated
														by location
										9/2/77	018086	2	1	Adj. & Added Packing
										9/6/77	025929	10	1	Cln. Torque SW & Stroked
										10/25/77	22145	2	2	Lub. Level Checked
---	(Liquid) HV-21516	PZR SAMPLE BLOCK	Weston	Globe	1"	Limitorque SMB-000-5	Has not caused an outage			1/12/76	008015	1	1	Adj. Packing, Restroked
							or delay.							

Table G-1. (Cont'd)

MARK #	MARK #	VALVE NAME	MANUF.	TYPE	SIZE	OPERATOR	OUTAGES DUE TO THIS VALVE			WORK REQUEST DATA				
							DATE	DURATION	CAUSE	DATE	WR#	MNHR	CREW	WORK PERFORMED
MU-RV2	PSV-22012	LETDOWN LINE RELIEF	Dresser	Relief	2½"	N.A.	It was reported by SMUD			2/10/76	010298	1	2	Tighten Flange
	(Upstream						that this Valve delayed			10/10/77	22621	16	2	Hydrotest, Set
	of Filters)						a Startup. Date(s)							Pressure
							are not identified or							
							confirmed by SMUD Records							
N.A.	PSV-22203	LETDOWN LINE RELIEF	Dresser	Relief	2½"	N.A.	It was reported by SMUD			11/15/74	000747	3	1	Reworked DISC & Seat
	(Downstream						that this Valve delayed			7/16/76	013350	10	1	Lapped Seat & Plug, Set pressure
	of Filters)						a Startup, Date(s)			8/31/76	014390	1	1	Tighten B/D Ring to stop leak
							are not identified or			9/15/76	014814	1	1	Tighten Bolts
							confirmed by SMUD Records			11/8/76	015814	4	2	Repl. Gasket
										12/6/76	016560	3	2	Set Press to 410PSIG
MU-V17	LV-21503	MAKEUP FLOW CONTROL	Fisher/CCI	Control	2½"	BMCO	It was reported by SMUD			12/20/74	001283	1	1	Adj. Packing
							that this Valve delayed							
							a Startup Date(s)							
							are not identified or							
							confirmed by SMUD Records							
----	SIM-037	HPI CHECK ISOL. FROM RCS	Velan	Swing Ck.	2½"	N.A.	No Outages identified			No Failure Data @ SMUD Identified				
----	SIM-041	HPI CHECK ISOL. FROM RCS	Velan	Swing Ck.	2½"	N.A.	or Reported							
----	SIM-049	HPI CHECK ISOL. FROM RCS	Velan	Swing Ck.	2½"	N.A.								
----	SIM-050	HPI CHECK ISOL. FROM RCS	Velan	Swing Ck.	2½"	N.A.								

Table G-1. (Cont'd)

B&W MARK #	SMUD MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPERATOR	OUTAGES DUE TO THIS VALVE			WORK REQUEST DATA				
							DATE	DURATION	CAUSE	DATE	WR#	MNHR	CREW	WORK PERFORMED
DH-V5A	SFV-25003	LPI PUMP ISOL. FROM BWST	Aloyco	Gate	16"	Limitorque SMB-2	10/1/77		Failed Opr.	11/6/74	000617	1	1	Added Packing
										8/26/75	005909	1	1	Adj. Packing & Restrok
										10/1/77	022582			Repl. HW Clutch & Clutch Pinion
										11/30/77	022147	2	1	Tested Torque SW Set
DH-V5B	SFV-25004	LPI PUMP ISOL. FROM BWST	Aloyco	Gate	16"	Limitorque SMB-2				5/16/75	003598	1	1	Adj. Packing
										11/6/75	007610			Reinstall Plastic Tube Around Pos. Ind
										2/23/76	10324	1	2	FAB & Inst. Ind.
										4/30/76	12568	4	2	Inspect Motor Opr.
										6/3/76	12843	16	2	Disassm. for Repair
										6/3/76	12566	16	3	FAB. new Stem
										6/3/76	12567	3	1	Insp. Body & Seat
										11/23/76	015812	4	1	Repl. Motor with Conc.
										2/28/77	019005	1	1	Mark for Temp. Pos. Until New Stem
										3/25/77	019395	8	2	Weld Rep. Stem. & Mach.
										10/27/77	26545	2	2	Repl. Damag. Gears
										11/2/77	26598	2	8	Insp. & Install Gears
DH-V1	HV-20001	DECAY HEAT LETDOWN ISOL.	Velan	Gate	12"	SMB-3-80	Has Not	Caused an	Outage	3/30/75	002905	1	2	Adj. Packing
							Or Delayed	Startup.		8/18/75	005054	1	1	Adj. Packing & Restrok
										5/6/76	012772	1	1	Adj. Packing & Restrok
										10/9/76	015226	1	1	Cleaned Valve
										8/30/77	018575	2	2	Clean Valve & Adj. Packing
DH-V2	HV-20002	DECAY HEAT LETDOWN ISOL.	Velan	Gate	12"	SMB-3-80	Has Not	Caused an	Outage	3/30/75	002904	1	2	Adj. Packing
							Or Delayed	Startup.		5/6/76	012773	1	1	Adj. Packing & Clean
										10/9/76	015227	1	1	Cleaned
										10/31/77	022243	24	1	Repl. Transmitter With Rosemont

Table G-1. (Cont'd)

B&W MARK #	SMUD MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPERATOR	OUTAGE DUE TO THIS VALVE DATE	DURATION	CAUSE	WORK REQUEST DATE	WR#	MNHR	CREW	
	PSV-20533	MAIN STEAM CODE SAFETY	Dresser	Relief	6"	N/A	One Valve has stuck open				34 WR's are for lapping			
				Valve			and delayed a return-to-				the seat and plug.			
							power operation.							
	PSV-20534	MAIN STEAM CODE SAFETY	Dresser	Relief	6"	N/A								
				Valve										
	PSV-20544	MAIN STEAM CODE SAFETY	Dresser	Relief	6"	N/A					7 WR's for setting the set-			
				Valve							point.			
	PSV-20545	MAIN STEAM CODE SAFETY	Dresser	Relief	6"	N/A					3 Misc. WR's			
				Valve							44 WR's total			
	PSV-20559	MAIN STEAM CODE SAFETY	Dresser	Relief	6"	N/A								
				Valve										

Table G-1. (Cont'd)

B&W MARK #	SMUD MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPERATOR	OUTAGE DUE TO THIS VALVE DATE	DURATION	CAUSE	WORK REQUEST DATE	WR#	MNHR	CREW	
---	TV-1	MAIN STEAM THROTTLE (STOP)	West.	Gate	26"	Hydraulic Operated				8/16/76	012514	80	6	Completed Linkage Mod.
										10/11/77	025494	224	1	
---	TV-2	MAIN STEAM THROTTLE (STOP)	West.	Gate	26"	Hydraulic Operated				7/16/76	13475	3	2	Rep'd Conduit
---	TV-2	MAIN STEAM THROTTLE (STOP)	West.	Gate	26"	Hydraulic Operated	NO INFO AVAILABLE ON			8/16/76	12515	80	6	Comp. Linkage Mod.
							OUTAGES DUE TO VALVES.			10/11/77	025495	224	1	
---	TV-3	MAIN STEAM THROTTLE (STOP)	West.	Gate	26"	Hydraulic Operated	SEE TEXT.			8/16/76	12516	80	6	Comp. Linkage Mod.
										10/11/77	025496	224	1	
---	TV-4	MAIN STEAM THROTTLE (STOP)	West.	Gate	26"	Hydraulic Operated				4/14/75	002559	1	1	Comp. NCR-112, Rem. Strnr.
---	TV-4	MAIN STEAM THROTTLE (STOP)	West.	Gate	26"	Hydraulic Operated				8/16/76	012517	80	6	Comp. Linkage Mod.
										10/11/77	025497	224	1	
---	GV-1	TURBINE GOVERNOR	West.	Gate	26"	Hydraulic Operated	DELAYED RETURN TO POW-			4/19/77	020522	2	2	No work - Ind. OK
							ER AS A RESULT OF EHC			10/18/77	025498	147	1	
							OIL PROB.							
---	GV-2	TURBINE GOVERNOR	West.	Gate	26"	Hydraulic Operated				4/26/77	020610	1	1	Reposition SW
										10/21/77	025499	147	1	Seatring broke & Repl. (Refuel)
---	GV-3	TURBINE GOVERNOR	West.	Gate	26"	Operated				11/7/77	22942	4	2	Meter Tracked OK
														While Stroking
---	GV-3	TURBINE GOVERNOR	West.	Gate	26"	Hydraulic Operated				10/18/77	025500	147	1	Disasem. & Inspect.
---	GV-4	TURBINE GOVERNOR	West.	Gate	26"	Hydraulic Operated				2/23/77	018602	3	1	Reinstall Actuator Rod
---	GV-1 to 4						12-3-77	1½ days	(*)	10/21/77	025501	147	1	Weld Repaired Body & Install New Seat,
								@ 85%						Lapped

* On 12-3-77 Perfromed Turbine Gov. Valve Test, one of the Valves stuck close requiring holding power @ 85% for 1½ days to repair valve.

Table G-2. Valve Failure/Repair Data From Oconee 1

B&W MARK #	DUKE MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPR.	OUTAGES DUE TO THIS VALVE			WORK REQUEST DATA			
							DATE	DURATION	CAUSE	COMPL. DATE	WR #	MNHR CREW	WORK PERFORMED
RC-V2	RC-2	PZR. SPRAY CONTROL BYPASS	Velan	Globe	½"	Manual	12-21-77	92 Hrs.	Packing Leak	--	ECI 77-52	2 x 4	REPLACED VALVE
								(Part of	RC-16				
									outage)				
RC-V5	RC-3	PZR. SPRAY CONTROL BLOCK	Rockwell	Globe	2½"	SMB-0015				5-3-74	07308		RESET TORQUE SW.
										5-8-77	22718	2 X 2	WON'T OPEN, ADJ. TORQUE SWITCH
RC-RV3	RC-66	PZR. POWER RELIEF	Dresser	R/V	2½"	EM				2-3-75	RADCAS	2 x 12	LAPP SEAT TO STOP LEAK
										1-9-76	07361	2 x 2	REP'L LIMIT BOX
										3-19-76	--	2 x 4	REP'L LIMIT BOX LEAK PAST STEM
										1-19-77	18953	3 x 16	LAPP SEAT TO STOP LEAK
										7-21-77	24208	2 x 12	REP DISK + GASKETS
									VALVE	10-4-77	26298	3 x 16	LAPP MAIN & PILOT VALVES
RC-V2	RC-4	PZR. POWER RELIEF BLOCK	Dresser	Gate	2½"	Limit.				4-19-76	10189	32	WON'T OPEN, RESET
		(CHANGED TO W)								11-11-76	17071	(2 x 2)	FLANGE LEAK REP'D
										8-26-77	95525	(2 x 4)	MACHINED
RC-RV4A+B	RC-67 & 68	PZR. CODE RELIEF	Dresser	R/V	2½"	N/A				9-26-77	25861	2 x 8	REPL. W/DRESSER
										9-26-77	25860	4 x 6	REPL. W/DRESSER
NA	RC-15	PZR. SAMPLE BLOCK (STEAM)	Velan	Globe	½"	Manual				NO DATA			
NA	RC-16	PZR SAMPLE BLOCK (LIQUID)	Velan	Gate	1"	Manual	12-1-77	22 Hrs.	Packing Leak	8-26-77	24957	2 x 6	REPACKED
							12-21-77	92 Hrs.	Packing Leak				

Table G-2. (Cont'd)

B&W Mark #	DUKE Mark #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPR.	OUTAGES DUE TO THIS VALVE: DATE	DURATION	CAUSE	WORK REQUEST: COMPL. DATE	WR #	MNHR CREW	WORK PERFORMED
LP-V1	LP-1	D.H. LETDOWN ISOL.	Walworth	Gate	12"					2-26-76	08493	2 x 2	REPL PACKING
LP-V1	LP-1	D.H. LETDOWN ISOL.	Walworth	Gate	12"					3-27-76	09142	2 x 2	REPL MAN. BRAKE BUTTON
LP-V2	LP-2	D.H. LETDOWN ISOL.	Walworth	Gate	12"					1-4-75	RADCAS	2 x 2	REPL. MICRO SW.
LP-V5A	LP-21	LPI PUMP SUCTION - BWST	Rockwell	Gate	14"					11-8-75	5479	2 x 8	TIGHTEN BOLTS
LP-V5B	LP-22	LPI PUMP SUCTION - BWST	Rockwell	Gate	14"					12-10-74	4736	2 x 8	REWOUND MOTOR
LP-V5B	LP-22	LPI PUMP SUCTION - BWST	Rockwell	Gate	14"					1-2-75	5375	2 x 4	OPERATOR STUCK
NA	HP-43	LETDOWN LINE RELIEF VALVE	Lonergan	R/V	2 x2½"					8-26-77			REPL. INTERNAL PT.
													REPACKED
													REPACKED
HP-V23	HP-120	MAKEUP FLOW CONTROL	Leslie	Globe/ Throttle	2½"					12-28-74		2 x 2	REPD' POS. ON CONTROL R
										4-30-76	18394		REPACKED
										7-3-76		1 x 1	WON'T CLOSE - BAD E/P CONV
										12-3-76		1 x 1	REPL. COTTER PIN ON LINK
										12-9-76		16	LAPPED LEAKING ST.
										8-29-77		12	LAPPED SEAT & REPL. GASKET
										9-30-77		2	OPER. IN HAND POS.
													NOT AUTO. REPACKED
													4 TIMES

Table G-2. (Cont'd)

B&W MARK #	DUKE MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPR.	OUTAGES DUE TO THIS VALVE			WORK REQUEST			
							DATE	DURATION	CAUSE	COMPL. DATE	WR #	MNHR CREW	WORK PERFORMED
HP-V5	HP-7	LETDOWN FLOW CONTROL	Leslie	Globe	2½"	BMCo				4-26-74	7236		STUFF IN BOX LK-TIGHTEN
HP-V5	HP-7	LETDOWN FLOW CONTROL								8-29-74	--		REPAIRED CONTR'R
HP-V5	HP-7	LETDOWN FLOW CONTROL								9-6-74	2492		REPLACED REGULATOR
HP-V5	HP-7	LETDOWN FLOW CONTROL								10-12-74	3755		ADJ. AIR SUPPLY REG
HP-V5	HP-7	LETDOWN FLOW CONTROL								7-10-77			TIGHTEN PACKING
HP-V5	HP-7	LETDOWN FLOW CONTROL								8-6-77			CHK'D POSITIONER &
													DIAPHRAM - RESET
													REPACKED 4 TIMES
--	HP-64	RCP SEAL INJ. THROTTLE	Velan	Globe: Needle	½"					NO DATA			
--	HP-65	RCP SEAL INJ. THROTTLE											
--	HP-66	RCP SEAL INJ. THROTTLE											
--	HP-67	RCP SEAL INJ. THROTTLE											

Table G-2. (Cont'd)

B&W MARK #	DUKE MARK #	VALVE NAME	MANUF.	TYPE	SIZE	TYPE OPR.	OUTAGES DUE TO THIS VALVE: DATE	DURATION	CAUSE	WORK REQUEST: COMPL. DATE	WR #	MNHR. CREW	WORK PERFORMED
NA	MS-103	MS THROTTLE (STOP)	G.E.							8-14-77			REPL PACKING GLAND
													BOLTS & REPACKED
NA	MS-2	MS CODE SAFETY	Crosby							2-27-76		4 x 6	LAPP DISC. + SEATS
	MS-9	MS CODE SAFETY	Crosby							2-27-76		4 x 6	LAPP DISC. + SEATS
	MS-10	MS CODE SAFETY	Crosby							2-27-76		4 x 6	LAPP DISC. + SEATS
	MS-11	MS CODE SAFETY	Crosby							2-27-76		4 x 6	LAPP DISC. + SEATS
NA	MS-106	TURBINE GOVERNOR	G.E.							3-1-76		105	DISASS, CLN & INSP
	MS-107	TURBINE GOVERNOR	G.E.							3-1-76		105	DISASS, CLN & INSP
	MS-108	TURBINE GOVERNOR	G.E.							3-1-76		105	DISASS, CLN & INSP
	MS-109	TURBINE GOVERNOR	G.E.							3-1-76		105	DISASS, CLN & INSP
NA	MS-19	BYPASS TO CONDENSER	Crane							8-27-77		2 x 4	LOOSE STEM-TIGHT
	MS-22	BYPASS TO CONDENSER								NO INFO.			SOR.
	MS-28	BYPASS TO CONDENSER											
	MS-31	BYPASS TO CONDENSER											

APPENDIX H

List of Abbreviations

AFHB	Auxiliary fuel handling bridge
AFW	Auxiliary feedwater
API	Absolute position indicator
APSR	Axial power shaping rod
ARIS	Automatic reactor inservice inspection
B&W	Babcock & Wilcox
B/B	Body to bonnet
BOP	Balance of plant
BF	Butterfly
BWST	Borated water storage tank
c	Critical path
CBAST	Concentrated boric acid storage tank
CCW	Condenser circulating water, counter-clockwise
CELF	Combined equipment limiting factor
CF	Core flooding
CLT	Containment leak test
cplg	Coupling
CR	Crystal River (Florida Power Company)
CRD	Control rod drive
CRDM	Control rod drive mechanism
CSA	Core support assembly
CSAE	Condensate steam air ejector
cse	Cause
ck	Check
DB	Davis Besse (Toledo Edison Company)
DH	Decay heat
DOE	Department of Energy
E	East
ECCS	Emergency core cooling system
ECP	Estimated critical position
EEI	Edison Electric Instritue
EFPD	Effective full-power days
EFPH	Effective full-power hours
EHC	Electro hydraulic control
EOCL	End of core life
EOF	Equipment outage factor

E/P	Electric/piston
EPRI	Electric Power Research Institute
ESFAS	Engineered safety features actuation system
FDW, FW	Feedwater
FHG	Fuel handling bridge
FOAK	First of a kind
FP	Full power
FTC	Fuel transfer canal
FWPT	Feedwater pump turbine
GE	General Electric Company
gpm	Gallons per minute
GWD	Gaseous waste disposal
h	Hours
HP	Health physics, high pressure
HPI	High-pressure injection
HPSW	High-pressure service water
HTR	Heater
ICS	Integrated control system
ID	Inside diameter
I&E	Instrument and electrical
Inst	Instrument
IP	Instrument procedures
LWD	Liquid waste disposal
LBP	Lumped burnable poison
LER	License event report
LF	Limiting factor
LFM	Limiting factor for maintenance
LFO	Limiting factor for operation
LFR	Limiting factor for refueling
lkg	Leaking
LOCA	Loss-of-coolant accident
LP	Low pressure
LPI	Low-pressure injection
LPSW	Low-pressure service water

malf	Malfunction
MFHB	Main fuel handling bridge
mhrs, mh	Manhours
mrem	millirem
MS	Main steam
MSIV	Main steam isolation valve
MSSV	Main steam stop (throttle) valve
MST	Main steam transmitter
MT	Magnetic particle
mtr	Meter
MTTR	Mean time to repair
NI	Nuclear instrumentation
NA	Not applicable
NAV	Not available
nc	Non-critical path
NDE	Nondestructive examination
NNI	Non-nuclear instrumentation
NPRDS	Nuclear plant reliability data systems
NR	Narrow range (transmitter)
NRC	Nuclear Regulatory Commission
NSS	Nuclear steam system
OTSG	Once-through steam generator
OD	Outside diameter
PWR	Pressurized water reactor
PCI	Pellet-cladding interaction
PI	Position indicator
ppb	Parts per billion
PT	Physics tests, dye penetrant
RADCAS	Reliability and availability data collection and analysis system
RCP	Reactor coolant pump
RCS	Reactor coolant system
RCW	Recirculating cooling water
R&D	Research and development
repl	Replace
rem/h	Rem/hour
rpr	Repair

RPS	Reactor protection system
RV	Reactor vessel
RX	Reactor
SF	Spent fuel
SFB	Spent fuel bridge
SFP	Spent fuel pool
SMUD	Sacramento Municipal Utility District
SOAK	Second of a kind
SRCI	Safety-releated controls and instrumentation
SSHT	Surveillance specimen holder tube
TMI	Three Mile Island (General Public Utilities Co.)
TC	Thermocouple
W	West
W	Westinghouse
WA	Work authorization
ΔP	Pressure drop

APPENDIX I

Definitions

Additional outage time: The time required to bring the plant to a condition that permits work to be done and to return to power after work is completed.

Availability: The amount of time that the plant was available for power production and which is represented as a percentage of the time that the plant could be available.

B&W: The Babcock & Wilcox Company, a wholly owned subsidiary of the J. Ray McDermott Company, 1010 Common Street, New Orleans, Louisiana 70112.

Combined equipment limiting factor (CELF): The loss of plant availability in effective full-power hours per unit-year for a given system/component. The CELF is a normalized, one-reactor-unit-averaged value which includes LFO historical and current data, and the actual outage extension portion of the LFR for those systems/components which are not directly a part of the refueling activity.

It is the single figure-of-merit factor for availability evaluation.

The CELF is determined from the following formula:

$$\text{CELF} = \frac{\text{LFO}_H + \text{LFO}_C}{2} + \text{refueling outage extension (EFPH/unit-year)}$$

See appendix C for definition of these terms.

Component: A part within the unit or system that performs a specific function, such as a pump, motor, valve, or heat exchanger.

Control rods: Clusters of core-length poison rods which are moved in and out of the reactor core to control reactor power and reactor power distribution.

Current data: Non-refueling data on Oconee Units 1, 2, and 3 for 1977.

EPRI: The Electric Power Research Institute, 3412 Hillview Avenue, Palo Alto, California 94304.

Failure: The termination of the ability of an item to perform its required function. Failures may be unannounced and not detected until the next test (unannounced failure) or they may be announced and detected by any number of methods at the instant of occurrence (announced failure).

Feed and bleed: A method of reactor control wherein major reactivity changes are made by adding (feeding) or removing (bleeding) borated water concentrations to and from the reactor coolant system.

Historical data: Data collected on Oconee Unit 1 from July 1, 1974, through December 31, 1977.

Key activity/key component: A component (or activity) whose failure or malfunction caused or could have caused a plant shutdown or power reduction, whose failure or malfunction extended or could have extended a plant shutdown or power reduction; whose maintenan-e or use during the refueling/maintenance outage was on or could reasonably have been on the critical path; whose maintenance would cause workers to receive high doses of radiation; whose maintenance frequency or manhour requirements was deemed to be excessive.

Key valve: Valve and valve operators with associated instrumenation which have had a negative impact on plant availability and/or valves which have had excessively high maintenance requirements.

Limiting factor for maintenance (LFM): The manhours of labor for maintenance or repair per unit-year for a given system or component. The LFM is determined for the Oconee 1 historical data from the following formula:

$$\text{LFM} = \begin{matrix}\text{No. of}\\ \text{events}\end{matrix}\ (\text{No. of men} \times \text{MTTR})\ \frac{1}{3.5} \qquad (\text{mh/unit-year})$$

Refer to Appendix C for definition of these terms.

Limiting factor for operation (LFO): The loss of plant availability in EFPH per unit-year due to failure or malfunction of a given system or component. This factor includes power (EFPH) losses due to reactor shutdown and startup and component access as well as the power losses during the actual maintenance or repair work.

The LFO is determined for the Oconee 1 historical data from the following formula:

$$\text{LFO} = \begin{matrix}\text{No. of}\\ \text{events}\end{matrix} \left[\begin{matrix}\text{power loss}\\ \text{factor}\end{matrix} \times \left(\text{MTTR} + \begin{matrix}\text{additional}\\ \text{outage time}\end{matrix}\right)\right] \frac{1}{3.5} \qquad (\text{EFPH/unit-yr})$$

Refer to Appendix C for definition of these terms.

Limiting factor for refueling (LFR): The difference between the actual time (clock hours) to perform a given refueling outage activity and the B&W-projected standard time to perform that activity. Also, LFRs have been determined for certain components that undergo maintenance during the refueling activities. The LFR is expressed in equivalent full-power hours and is determined from the following formula:

$$\text{LFR} = (P - S)Fp \qquad (\text{EFPH/unit-year})$$

Refer to Appendix C for definition of these terms.

Lost capacity days (LCD): LCD is based on equipment outage factor data and indicates lost capacity days in terms of full-power production. This may result from full shutdown/artial load. The LCD can be determined for a single component/system in a given plant/combination of plants. It may also be determined for non-equipment/non-system items such as refueling, high radioactivity, system design problems, human error, balance of plant, and load dispatching.

Lumped burnable poison rods (LBP rods): Clusters of core-length poison rods located in fixed core positions and used to control core power distribution in new or reload cores.

Mean time between failures (MTBF): The arithmetic average of operating times between failures of an item.

Mean time to repair (MTTR): The arithmetic average of time required to complete a repair activity.

Oconee: The Oconee Nuclear Power Station, owned and operated by the Duke Power Company, P. O. Box 1278, Charlotte, North Carolina 28242.

Orifice rods (OR): Clusters of "short" non-poison rods used to limit reactor coolant flow through fuel element spaces voided by the absence of control rods or LBP rods.

Outage cause: A component failure, preventive maintenance, or other condition that requires that the unit or a component be taken out of service or run at reduced capacity.

Power loss factor: A multiplier in the LFO equation used to account for power generation capacity lost if the work event caused a plant shutdown or power reduction.

Project team: Representatives from B&W as the NSS supplier and Duke as both the owner/operator and the architect-engineer of the reference plant.

Rancho Seco: The Rancho Seco Nuclear Generating Station, owned and operated by the Sacramento Municipal Utility District, 6201 S. Street, P. O. Box 15830, Sacramento, California 95813.

Ratchet trips: An intermittent loss of power to the CRD stator which causes the roller nut to disengage, allowing the leadscrew to start to fall. If power is restored before the control rod is fully inserted, the roller nut will attempt to re-engage with the falling leadscrew.

Refueling outage: The scheduled outage to accomplish core refueling, plant maintenance, and plant modification. In this study it includes the period from breaker trip (15% power) to 75% power.

Refueling outage data: 1977 refueling outage data from Rancho Seco, Oconee 1, and TMI-1.

Reliability: The characteristic of an item expressed by the probability that it will perform a required mission under stated conditions for a stated mission time.

Standard projected performance time: An estimated "normal" time to complete an individual refueling outage task.

Station/plant: One or more electrical energy-producing facilities located at a common site and in close proximity to each other.

System: An arrangement of parts within the unit or a work activity that performs a specific function, such as the feedwater system, control rod drives, or the core physics tests.

TMI: The Three Mile Island Nuclear Power Station, owned and operated by the Metropolitan Edison Company (a subsidiary of General Public Utilities Corp.), P. O. Box 542, Reading, Pennsylvania 19603.

Unit: The set of equipment uniquely with the reactor, including turbine generators and ancillary equipment, considered as a single electrical energy production facility.

Upender: A device in the fuel transfer equipment that moves fuel assemblies from a horizontal to a vertical position, or vice versa.

Xenon hold: A hold at steady-state power (usually near 90%) to wait for transient xenon conditions to reach near-equilibrium conditions.

APPENDIX J

References

REFERENCES

1. B&W Refueling Outage Availability Study, Phase 1 Final Report, COO-4068-14, Babcock & Wilcox, Lynchburg, Virginia, November 1977.

2. Oconee Nuclear Station Unit 1 Cycle 2 Refueling Shutdown Primary System Radiation Levels, EPRI-NP-340, Electric Power Research Institute, Palo Alto, California, May 10, 1977.

3. Improvement of Availability of PWR Nuclear Plants Through the Reduction of Time Required for Refueling/Maintenance Outages — Interim Report, COO-2962-5, Westinghouse, April 1978.

4. Reduction of Nuclear Refueling Outage Time for Increased Plant Availability — Phase 1 Summary Report, COO-2426-138, Combustion Engineering, April 1978.

5. Refueling Outage Trends in Light Water Reactors — Interim Report, EPRI-NP-842, Electric Power Research Institute, Palo Alto, California, Project 705-1, August 1978.

6. Assessment of Industry Valve Problems, EPRI-NP-241, Electric Power Research Institute, Palo Alto, California, November 1976.

7. "Power Engineers Valve Manual," reprinted from Power Engineering Magazine.

8. "Solenoid-Operated Valves for Nuclear Service," Power Engineering, June 1978.

9. "Turbine Cycle," Section VI, Nuclear Power Experience, Vol. PWR-2, Part A — Turbine, April 1976.

10. "Turbine Cycle," Section VI, Nuclear Power Experience, Vol. PWR-2, Part E — Condensate and Feedwater, September 1975.